A Guide to PFR Operational Data

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1. Scope Work Plan for FRKC Technical Area "Reactor Operations"

PFR (prototype fast reactor) was sanctioned as a project in 1966 and by October 1967 all of the major contracts were let. Delays during construction meant that sodium filling did not take place until September 1973, 33 months later than the original programme intended. The reactor then went critical in March 1974 and full power was achieved in February 1977. The reactor continued to operate until 1994.

1.1. What we are aiming to produce – "A Guide to PFR Operational Data"

The objective is to produce an annotated index to documents containing data on the operation of PFR, in a form that will allow the data and the experience gained to be used in support of the design and operation of future fast reactors and sodium-cooled reactors.

The ultimate intention is that the index will be to primary sources of data (measurements, first-hand descriptions, etc.) if they still exist and can be found, but this is unlikely to be achievable within the present phase of work. The present interim intention, therefore, is that the index will lead to secondary sources and interpretations as close as possible to the original data.

The index will cover the nuclear plant including the reactor, primary circuit and fuel handling equipment; and the secondary coolant circuits including the steam generators and auxiliary equipment. The non-nuclear "conventional" steam plant (turbine, alternator, condenser, feed pumps, feed heaters, steam drums, etc.) will be excluded.

The index will consist, for each document, of its title, reference number, authorship, location and (where known) ownership. In the present phase of work the search for relevant documents has been confined almost entirely to the archive in the possession of Wood plc. Some 20,000 documents from this microfiche archive were identified (by keyword search on title, series and author) as relating to the UK Fast reactor programme and digitised in the form of PDF files. These PDFs are referred to as the "Wood Archive" in what follows. The titles, authors and original reference numbers of the documents are listed in a series of Excel files which form a catalogue of the "Wood Archive". The reference number quoted below is the number of the entry in these Excel files.

The annotation will consist of a list of components (control rods, pumps, etc.) and operational issues, together with a brief overview of the operation of the reactor, arranged so as to direct the user to the important experience gained and lessons learned, and the impact they had on the design of CDFR and EFR.

1.2. How have we gone about producing it?

The starting point was Chapter 5 of the July 1996 "Outline Review of PFR" by DCG Smith. A list of components and reactor experiments was drawn up (see below) based on the sections of Chapter 5. Section 5.2 (Fuel) was excluded because it is the subject of the work on Technical Area 1.

For each item in this list supporting documentation was sought. Smith indicates where some are to be found (his Appendix B provides suggestions to "the reader who wishes to dig deeper"). The leading sources seem to be the PFR Design Safety Report and a "comprehensive bibliography covering fast reactor work within the United Kingdom" AEA-TS-0044. These have not yet been found.

Beaumont's 1993 outages report FR/E/004495A may provide references to valuable documentation but has yet to be utilised. A report by Cruickshank and Judd in IAEA-TECDOC-1180 (2000) has some useful references.

Within the time and funding available it was not possible to cover all the PFR plant items. Those for which PFR experience was most significant were selected.

1.3. Outcome: what has been achieved

A guide has been prepared for a number of the main plant areas but not all and not all components within each main plant area. The paper by DCG Smith served as the skeleton of the operational experience as originally intended and the Wood archive has been used almost, but not entirely, for all the supporting information and data. The Wood archive includes documents generated by Wood and their predecessor companies (AMEC Foster Wheeler, AMEC, NNC, NPC and TNPG) and by AEAT, UKAEA and by all the participating organisation in the fast reactor project including, BNFL/NNL, CEGB Nuclear Electric and various contractor companies. Wood, through its predecessor companies, participated in the design, construction and operational support for PFR and in the committees through which the commercial reactor development was pursued from CDFR (Commercial Demonstration Fast Reactor) to EFR (European Fast Reactor). They must be congratulated on keeping such a good archive and NNL for funding the digitisation without which this work would not have been possible on this timescale and with the funding available.

Of the documents recording the work of the participating organisations (mainly UKAEA and AEAT) the archive contains only those that were copied to Wood's predecessors. For most of the duration of the Fast Reactor project they appear to have received a comprehensive account, but during the final years, after the formation of AEAT, it seems to have been less complete. It is noticeable that there seem to be fewer documents from the period 1988-93 than from earlier years. This may be because the UKAEA/AEAT work was not documented so carefully or because the documents were not archived properly or copied so widely. To complete this index, therefore, it will be necessary to search other archives and collections.

The Wood archive was made available and searches were possible using the title, author, committee and/or date. A keyword search of the content of the documents was not possible. Best endeavours have been used to provide the report listing relevant to each topic but this cannot be considered complete as some documents will inevitably have been missed and even some included which are not directly relevant.

The areas covered in this guide are as follows:

Primary Circuit:

- Core
- Core support structure
- Strongback
- Above-core structure
- Intermediate heat exchangers
- Decay heat exchangers
- Primary and secondary pumps
- Primary vessel
- Roof cooling system

Control System:

- Absorber rod mechanisms
- Failed-fuel detection system
- Automatic protection system

Fuel Handling Route:

- Not covered

Auxiliary Equipment:

- Decay heat rejection loops

Secondary Circuits:

- Secondary pumps (included with primary pumps because of their similarity)
- Evaporators
- Superheaters and reheaters
- Secondary cold trap

The limited extent of the present work did not allow some items to be covered:

Not Covered:

- Primary cold trap loop
- Fuel handling route
- Under-sodium viewing
- Secondary circuit leak detection and effluent systems

2. Primary Circuit

PFR is a Pool type reactor with all the major components housed in a single vessel, as opposed to a Loop type reactor where the Intermediate Heat Exchangers (IHX) and Pumps are housed in separate vessels connected by pipes. The reactor elevation is shown in Figure 1with cross sections taken through the IHX and Pumps and Figure 2 sectioned through the fuel handling route.

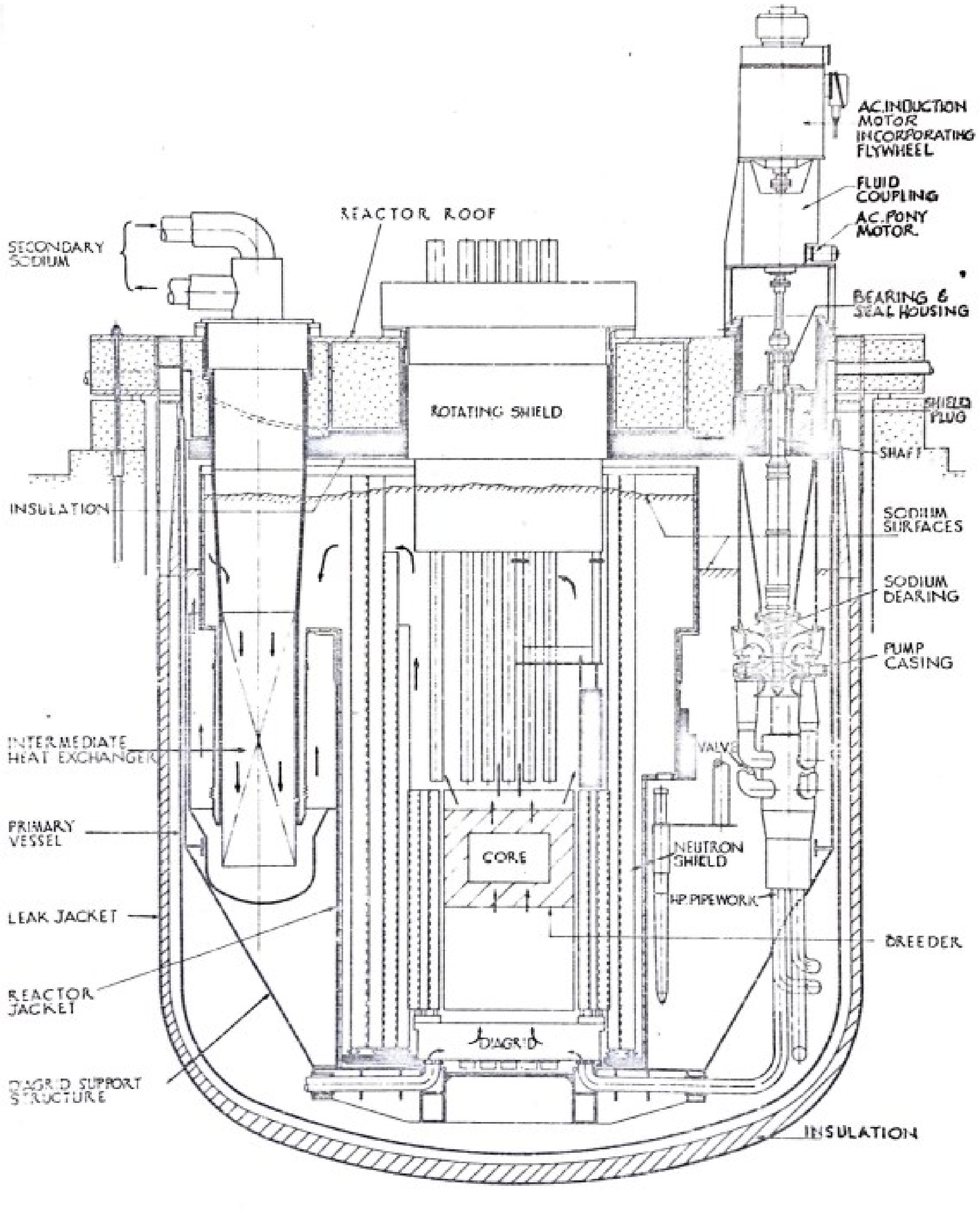

FIG Nº1. ELEVATION THROUGH PFR PRIMARY CIRCUIT

Figure 1: PFR Primary Circuit Elevation Pumps and IHX

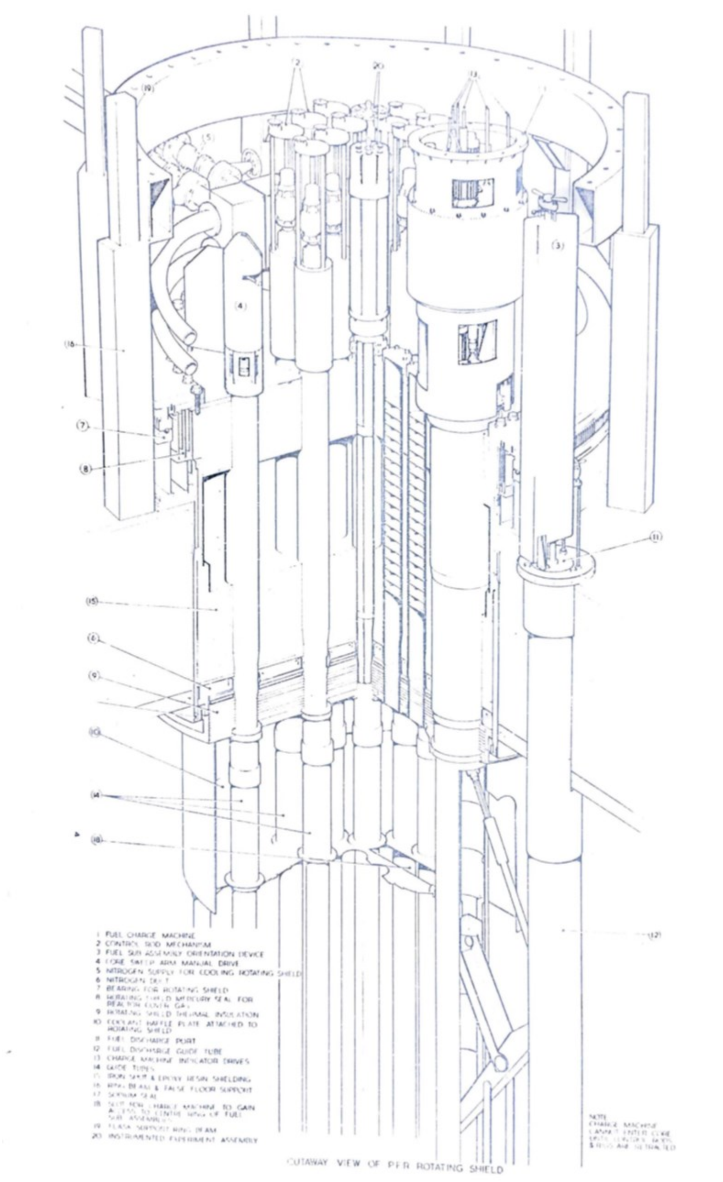

Figure 2: PFR Primary Circuit Elevation: Rotating Shield

2.1. Core and Blanket

(Numbers in brackets refer to sections of Table 1)

The core plan is shown on Figure 3 and consisted of 2 enrichment zones, I and II, which was surrounded by the radial breeder zone III and breeder reflector zone IV. The Core zones I and II are concentric but the radial breeder was offset to give 3 rows of breeder on one side, intended to represent the increased breeding capability which was anticipated to be required in future commercial reactors.

The reactor was first made critical in 1974 in a straightforward and uneventful manner. Subassemblies containing solid steel pins were replaced in a pre-planned sequence by subassemblies containing fuel in the form of mixed uranium-plutonium oxide (MOX) pellets clad in sealed stainless steel pins. The objective of the first loading was to check the theoretical reactivity prediction. 54 subassemblies were installed and the reactor was found to be within 0.6% of critical with the absorbers at maximum height and therefore with minimum effect.

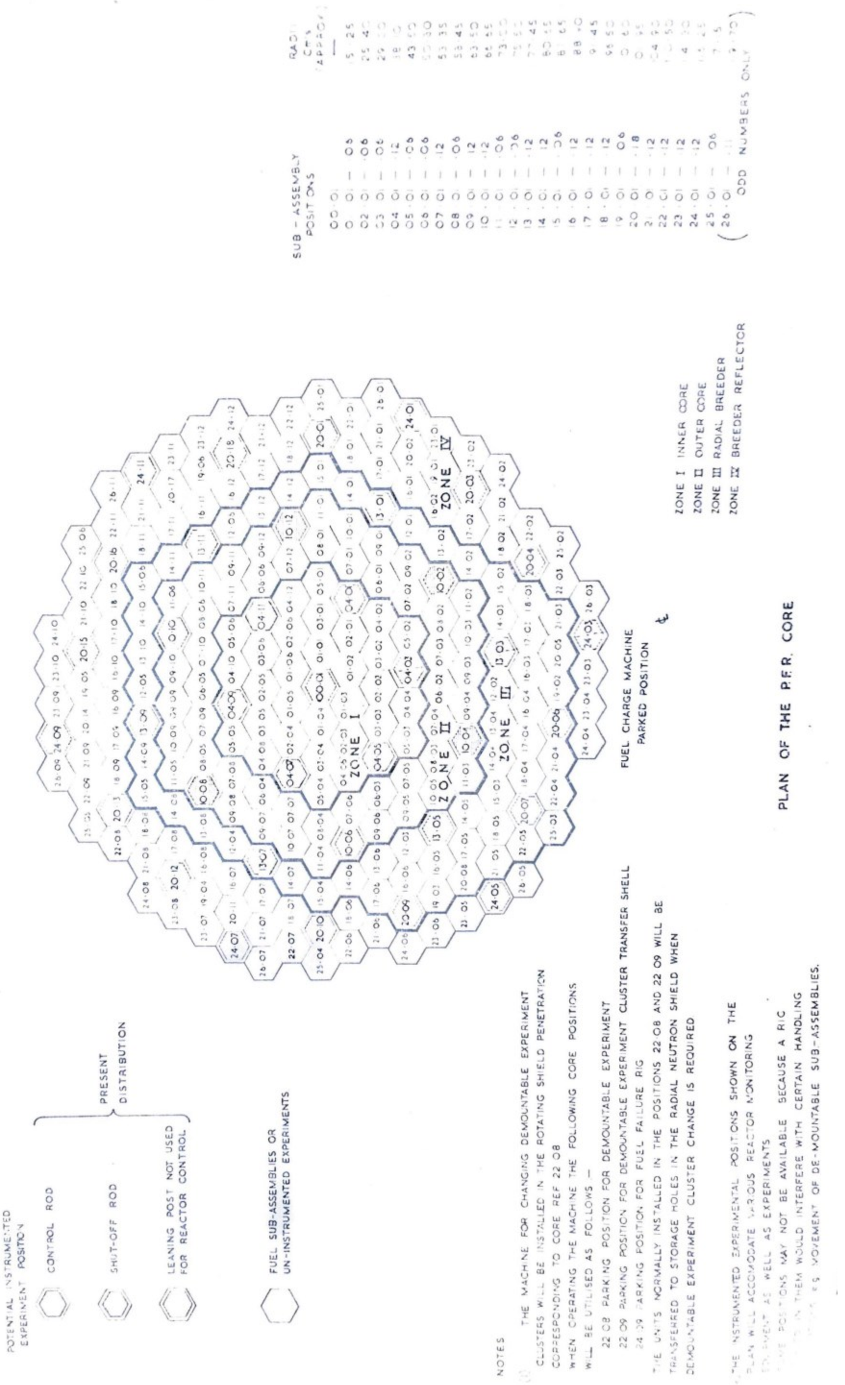

Figure 3: PFR Core Plan

The difference between observed and predicted reactivity was 1 %. The first operating core, with 65 subassemblies, was also 1% more reactive than had been predicted. Subsequent analysis explained the difference and revealed the need to use actual data rather than nominal data; for example the measured positions of the subassembly clusters in the diagrid and the actual isotopic compositions and densities of the individual fuel subassemblies that differed marginally but significantly from the nominal values. A series of low power reactor physics experiments was conducted using fission and ionisation chambers and activation foils (1). The results confirmed the pre-calculated neutron flux and power density distributions, the reactivity invested in the absorber rod system, the principal temperature coefficients of reactivity and the central reactivity worth of plutonium.

Early operation showed the reactor to be very stable and easily manoeuvrable. Unexpected reactivity effects arising from subassembly bowing in thermal gradients had been discovered in earlier reactors so PFR had been designed to eliminate such effects. The subassemblies were grouped in clusters of six round hexagonal “leaning posts”, which were intended to constrain any tendency to bowing. Experience indicated that this arrangement was effective in that no positive reactivity feedback was observed. Thermal and irradiation creep were not observed to reduce the efficacy of the leaning posts to any significant extent. Measurements of reactivity and power noise, 0.1 %, at the bottom end of the anticipated range, indicated that the leaning posts minimised vibration of the subassemblies (2). Throughout the life of the reactor the rate of loss of reactivity with burnup was measured (3), and as a matter of routine the reactivity worth of the control rods was determined (4). Various other nuclear reaction rate measurements were made (5).

There were more than 30 refuelling campaigns or “reloads”. Each was pre-planned to provide additional reactivity sufficient to compensate for burn-up during the forthcoming operating run and to accommodate any changes in the experimental fuel loading. The reactivity target lay within a small window: too much reactivity would have meant that on start-up the control rods would have been inserted so far that, in the event of a trip, complete insertion would have resulted in an unacceptably small reduction in reactivity; too little would have compromised the length of the next operating run. A further and much tighter criterion was imposed by the operating directives, which required that the predicted post re-load reactivity should be within about 40 cents of the observed value. This limit was equivalent to about 13 mm in the control rod curtain height and was designed to ensure that a significant misloading could not pass unnoticed.

Originally a critical approach was made after each reload, but later improved reactor physics calculation methods were developed allowing the sub-critical margin to be monitored as the reload proceeded (6). The calculations allowed for the perturbation of reactivity, neutron source magnitude and distribution, and flux distortion at each fuel element interchange. Typically a set of fifteen 2-dimensional calculations, representing fifteen interchanges, sufficed to provide the operators with adequate predictions of the count rates on each of the low power counting channels during the complete reload operation. The accuracy with which the sub-critical reactivity, which varied during a reload, could be measured was typically ± 10-15 cents and was mainly determined by the counting statistics. An important reason for monitoring the sub-critical reactivity was to guard against the accidental loading of an outer-core fuel subassembly, with its higher enrichment, into an inner-core position where, if it had gone undetected, its power rating would have been damagingly high.

During the reload, the shutdown margin was controlled administratively to ensure that the reactor would remain sub-critical in the event of the worst two misloadings. In practice this amounted to 12$ and corresponded to the erroneous removal of two absorber rods. An independent safeguard against an unplanned criticality was provided by the "Charge Machine Freeze", to be activated by any unexpectedly high count rate on the low power fission chambers. It was designed to de-energise the charge machine hoist and could have been recovered only after administrative intervention. It was deployed but never triggered.

The stability of the reactor depended on the reliable negative power and temperature coefficients of reactivity, which were measured at various steady operating conditions of power and coolant flow-rate (7). In addition the fast-acting component of the power coefficient, which is important in protecting the reactor in the event of an accidental addition of reactivity, was determined experimentally by imposing a rapid negative change in reactivity of -15 cents by dropping a control rod (8). This technique was possible because of the low level of power noise. The original intention had been that the time-dependence of reactivity feedback would be investigated by means of a reactivity oscillator, which had been provided but was never needed (9).

The results of these tests were compared with theoretical models of reactivity feedback. The principal component of the power coefficient was associated with the Doppler Effect, but with a significant contribution from the expansion of the fuel; the latter being governed at low power by the expansion of the fuel pellets and at higher power by the expansion of the cladding which the fuel had expanded to contact. For changes in coolant outlet temperature due to changes in power or flow-rate, the expansion of the control rod insertion mechanism was important. For changes in coolant inlet temperature the radial expansion of the core support system imposed a large reactivity change. Each of these feedback components had a different time-constant. All were negative and thereby dynamically stabilising. The good agreement with theoretical models gave confidence that no significant feedback factors had been overlooked.

Overall the experience gained established that the available reactor physics methods and data were able to provide information with sufficient detail and accuracy for the safe operation of the reactor and accurate management of the core and breeder, thus fulfilling an important objective of PFR as a prototype (10).

2.2. Core support structure

The PFR core support structure comprised a number of elements, namely the Fuel Element Carriers and Guide Tubes, Diagrid, Strongback and the Strongback support skirt carrying the load back through straps to the reactor roof. This section of the report deals with the Fuel Element Carriers Figure 4; (Ref. UKAEA (EG) 14915A), Guide Tubes (Ref. UKAEA (RG) 14919 and 14919B) and Diagrid (Ref. UKAEA (EG) 47049A).

The diagrid consisted of a large plenum with the top and bottom plates separated by tubes into which the Fuel Element Carriers were located. There was a second low pressure plenum forming an annulus off the top of the main plenum. The low pressure plenum supported the inner 3 rows of neutron shield rods which required a cooling flow. High pressure coolant was taken from the main plenum to the low pressure plenum through 3 gags located in the bottom of three special shield rods. The coolant supply from the main plenum was taken in a number of pipes distributed around the plenum but just below the plenum top plate. The purpose of this was to remove any gas that might have built up and avoid its passage through the core. The absence of any indication of unaccountable reactivity changes confirms the success of these measures in preventing gas accumulation and release.

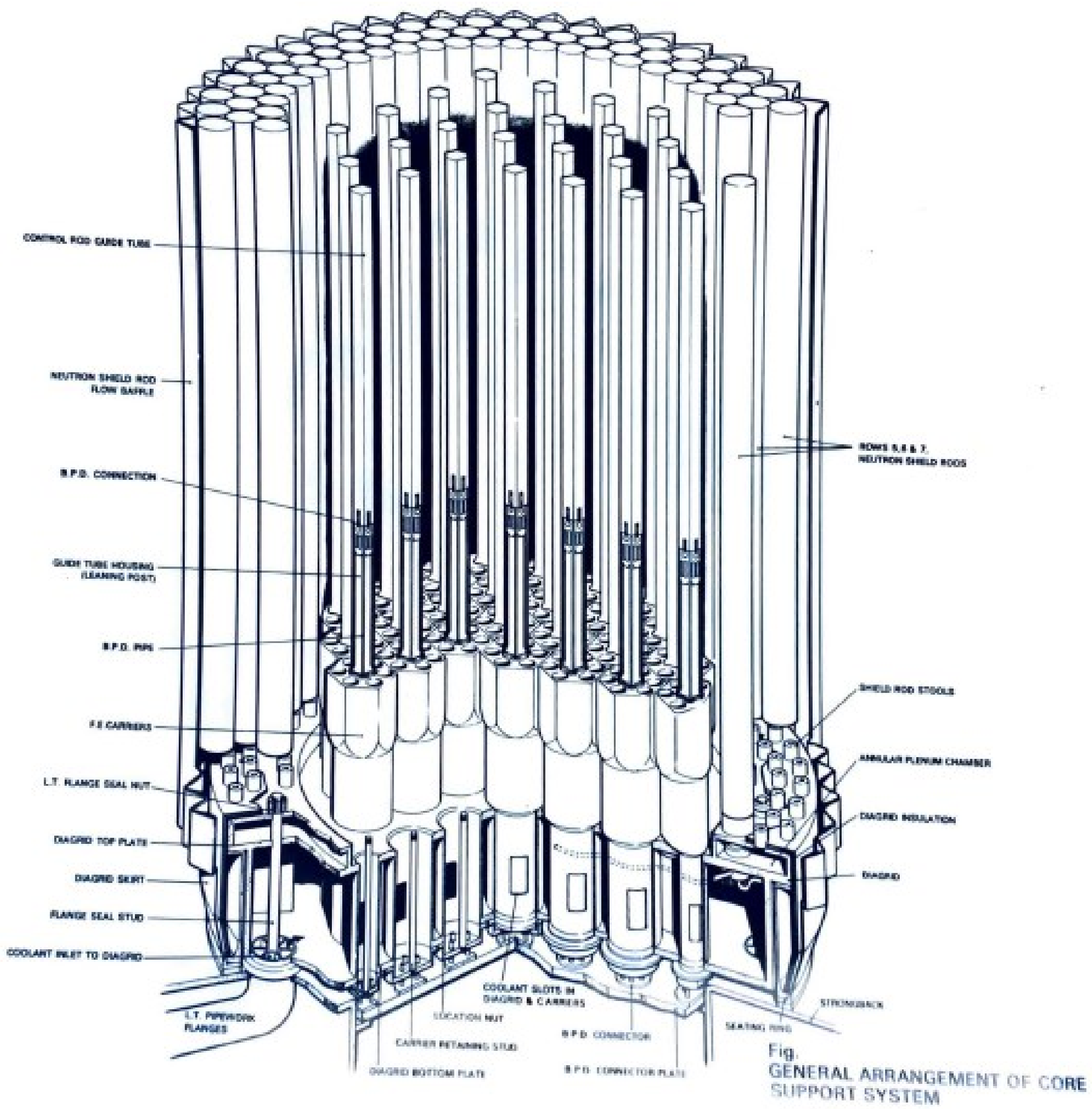

Figure 4: PFR Core Support Structure

The diagrid received the coolant from the primary pumps via 12 ducts which entered from the bottom of the diagrid. The ducts were keyed to the Strongback in such a way that they located the diagrid whilst allowing for the differential expansion between the diagrid and Strongback. The diagrid transmitted the core weight through a thin skirt to the Strongback which also accommodated the difference in thermal expansion to the Strongback. A description of the core support structure, the functional requirements and safety role is presented in relevant sections of the PFR design safety report (Ref. 25165.pdf).

Each of the 37 Fuel Element Carriers (FEC) supported 6 sub-assemblies (SAs) in a ring surrounding a Leaning Post (except for 6 corner carriers which supported 4 SAs). The SAs had a spike which entered the FEC with the lower location being offset to force the SA into contact with pads on the Leaning Post. Coolant entered the SAs at the spike inside the FEC. The FEC had a vent pipe taking any leakage from the lower spike to provide a hydraulic hold down. This leakage flow was taken back through the FEC to the cold pool whereas leakage from the upper location went to the hot pool and led to a small degradation of the core outlet temperature. The Guide Tubes were originally integrated with the Leaning Posts.

As well as the hold down pipes passing through the very high velocity coolant in the FECs the BPD samples were taken back through the FECs. Flow-induced vibration was a concern and careful attention given to the support of the pipes. Failures of the pipes would have been revealed by lifting of the SAs due to loss of hold down or by the excessive flow to the location sampling at the BPD. No failures were revealed during the life of PFR. (In designs since PFR the preferred route for the BPD samples was up via the ACS rather than down through the core support structure avoiding the potential vibration in the FEC, but this change was also influenced by the shorter transit times for delayed neutron samples which are important in large commercial reactors.)

Another potential issue for which no adverse feedback has been revealed is the contact bushes in the FECs for the SA spike. To minimise leakage and cavitation the SA spike clearance had to be small and because of the poor wear properties of stainless steel a wear-resistant contact surface was provided. The contact wearing surface of choice would have been stellite but because of the lower thermal expansion and the danger of production of cobalt-60 a local weld deposit strip at the contact point was chosen. While there may have been some scoring of the SA spike there was no evidence of high contact loads or difficulty in extracting the SAs.

The FECs were made removable with a single pre-tensioned central stud with a special nut to aid release, had removal been necessary. No tools or equipment were provided for the removal. In the same way the diagrid was also a removable component but at another level of complexity. None of these components was removed during the operation of the plant.

During construction the pressure loss of an as built FEC was measured by RNL and found to be very high. An important contribution to the high pressure loss was the mounting point for the retention stud forming a throat through which the hold down and BPD pipes also passed. The central FEC also had a splitter plate segregating the inlet flow to 2 sets of 3 SAs to minimise the risk of faults affecting all 6 central SAs. This further increased the pressure loss of the central FEC. The decision was taken to re-design the central FEC with a lower pressure drop. This was achieved by removing the constriction provided by the stud by extending the stud. A second measure was the elimination of the hold down pipes and allowing the hold down leakage flow to enter the hot pool rather than piping it down through the FEC. These steps reduced the overall pressure loss for the whole core because the gagging requirement of all other SAs was reduced accordingly. The manufacture of the new central FEC took advantage of the slippage in the construction programme due to other causes.

A dummy core was loaded and remained in place during sodium fill and commissioning (Ref.29799.pdf) and was intended to be in place for a water test of the primary circuit hydraulic performance, but the test was cancelled. Operation of the charge machine and the fuel route required the dummy core to be in place when the core was loaded. The dummy core remained in place until after sodium fill allowing measurements and checks before the actual core was loaded (Ref. 12306.pdf).

An unforeseeable problem arose when, in 1966, experimental evidence from DFR revealed for the first time that fast neutrons had induced substantial swelling in stainless steels. The implications of this Neutron Induced Void (NIV) swelling were profound. Modifications had to be made to components which would be exposed to a high flux of fast neutrons. Two lines of attack were followed. The first was a search for an acceptable alloy which was resistant to swelling, for example a high strength steel or nickel based alloy such as PE16; and secondly, a solution based on a component design change. The components most strongly affected were the fuel pin cladding, the subassembly wrappers and the guide tubes in which the control and shutdown rods moved.

The existence of a threshold in the swelling/dose rate relationship added to the difficulty. Another complexity arose from the fact that much of the data on swelling came from irradiated fuel pins. Inevitably this information related to steel under stress whereas the sub-assembly wrappers and guide tubes were relatively unstressed. The neutron spectrum differences between DFR and PFR were also relevant. Some guidance was becoming available from the Variable Energy Cyclotron at AERE and comfort was drawn from the consistency between data from this source and from the reactor experiments.

So far as the guide tubes are concerned, the fabrication of the FECs had not started. Although the FECs with integral Guide Tubes were designed to be removable as units it was intended in case of emergency repair and not a routine operational procedure. A challenging design change appeared necessary to terminate the Leaning Post below the core and out of the range of NIV and designing a Guide Tube that was readily removeable.

In October 1969, a paper to the Fast Reactor Development Committee discussed the options being considered to circumvent the effect of voidage in core components. The currently available information, extrapolated to PFR lifetime, meant, that the core design had to be radically modified. The least disruptive solution was to rotate the sub-assemblies part way through their core life.

The operational implications of the proposed core design changes were received by PFR Operators with strong misgivings. It signalled the end of the planned “Weekend Shutdowns”. In May 1969 the operators requested that design or material changes be sought which would limit the need for sub-assembly rotation to less than once per core lifetime and Guide Tube and Control Rod rotation should be not be required more than once per year. These contstraints were aimed at making full use of the scheduled long maintenance shutdown and had, as an objective, a return to the seven week refuelling cycle. In the event the short refuelling cycle and weekend shutdown were effectively abandoned from this point.

The challenge was to have a removable Guide Tube design that could be handled by the charge Machine and normal fuel handling route and a retention latch accommodated in the very limited space. By early 1970 the design and specification of the Guide Tubes and the Special Handling Tools (Ref. UKAEA (RG) 14919) was complete and ready for manufacture. Then during 1971 the guide tube were revised to use PE16 wrapper in place of the stainless steel to reduce the swelling (Ref. UKAEA (RG) 14919B).

The implications of NIV swelling could have been a major set back to the Project. Delays in the construction programme alleviated the position to some degree by allowing time for further experimental and theoretical development in the understanding of the effects of NIV swelling and a better knowledge of the properties of some of the contending fuel pin and fuel element materials (Ref. 03491.pdf, 13160.pdf. 11735.pdf, 14297.pdf, 09952.pdf, 09953.pdf, 09954.pdf, 09955.pdf, 09956.pdf)

The Guide Tubes and the components they housed (Control Rod, Shutdown Rods. Flowmeters and 4" Breeder reflectors) went through a series of changes from the original to Mk. I, II, II, IV and finally Mk. 6 (Ref.23069.pdf). Some Guide tube changes were required because of changes to the component they housed but there were issues with the Guided Tubes which required modification. These design changes (e.g. Ref.04672.pdf) include brittle latches (Ref.01847pdf, 28582.pdf) in the original Guide Tubes, concern about the BPD pipes becoming disconnecting due to differential growth between the pipe and the wrapper, and moving the lifting slots to below the core so that axial growth was not important for handling. None of the concerns led to failure but design changes were considered prudent. This evolution can be followed in a number of references (Ref. 00226.pdf, 01179.pdf, 01798.pdf, 01800.pdf, 01807.pdf, 01808.pdf, 01818.pdf, and 08419.pdf)

The factors determining the Guide Tube life were axial growth limited by the capability of handling tool or the above core Sweep Arm, bow limited by the need for unhindered movement of the component inside the Guide Tube, and handling by the charge machine and removal tool (Ref. 13027.pdf). The limit arising from the guide tube removal tool was examined (Ref.27975.pdf), which also made recommendations to be included in a new version of the tool.

Ways had to be found to improve the procedure for the replacement of the centre Guide Tube to reduce the anticipated 20 days needed to create access to the centre positon by the charge machine due to the need for active component handling (Ref 01820.pdf).

The Leaning Post bolts were identified as a possible limiting condition and the stressing assessed due to the dilation of adjacent subassemblies (Ref.25218.pdf and Ref. 25362.pdf). The safety case for the core support had to be developed in 1990 as the responsibility for licensing had been passed to the NII. Further work was necessary with respect to the critical defect size and monitoring and updating the safety case documentation (Ref.2243.pdf). Further information is provided under the Strongback section.

Other issues related to structural integrity and monitoring of the core support are covered in the following section on the Strongback.

An extensive list of references is given in Table 2.

2.3. Strongback and Internal Structures

This section of the report deals with the Strongback and Strongback Support skirt and includes the Reactor jacket which is also supported by the Strongback Figure 1 and Figure 2.

The Strongback was a box structure on which the Diagrid sat along with the 4 rows of outer neutron shield rods. The Diagrid was supported by a thin cylindrical shell to allow differential thermal expansion to the Strongback and sat on a levelling ring to provide the fine positional adjustment as a construction aid. The reactor jacket also was supported by the Strongback.

The hot pool (or inner pool as it was known in PFR) boundary was formed by the Reactor Jacket, Strongback, Diagrid Support Skirt and the Diagrid itself. The Reactor Jacket, Strongback and Diagrid Support Skirt were protected from the high hot pool temperatures by multi-layer gas filled stainless steel quilted insulation. The reactor Jacket had integral pods which located the Intermediate Heat Exchangers, forming a seal with them to complete the hot pool boundary. Because the pods and the Strongback were in the cold pool the insulation played an important role in limiting the temperature differences within the structures.

The low temperature pipes between the Primary Pumps and the Diagrid passed through the Strongback. The pipes were attached to the Strongback below the Diagrid and were keyed to the Strongback to locate the Diagrid but at the same time allow radial expansion of the Diagrid.

The intention was to ensure all the core support structures were at the relatively low temperatures of the cold pool so below the creep regime. A consequence of this was that monitoring throughout the life was very difficult and was not given a high priority during the design. An alternative to support the Strongback off the Primary Vessel would have contravened the design philosophy of a simple low-stress primary vessel without attachments, nozzles or penetrations, which is possible with a Pool type reactor. Some commercial reactor designs since PFR with much greater emphasis on in-service monitoring have opted for the alternative of supporting the Strongback off the primary vessel while preserving the benefits of the Pool type reactor with a simple primary vessel.

A temperature gradient similar to that which caused concern for the primary vessel top strake was also affecting the Strongback support skirt. Analysis (Ref. 18098.pdf) concluded that the conditions were not as damaging as those for the primary vessel, and that the solutions would be effective for both. Consequently all the attention was focused on the Primary Vessel for this phenomenon.

However assuring the structural integrity of the core support became an increasing requirement during the life of PFR. A Structural Integrity Advisory Group, with both AEA and independent membership, was setup to advise the Operators and assist with the preparation of safety cases.

In 1990, after the NII took regulatory control for PFR, they requested that the AEA provide a statement of the current safety case concerning failure of the core support structure. The case presented in a series of papers (Ref. 17339.pdf, 19837.pdf, 19838.pdf and 19947.pdf, 22493.pdf, 25252.pdf) was for the incredibility of failure based on three elements:

1. The original high standard of design, materials specification, fabrication and pre-service inspection (including reviewing the radiographic records)
2. A lack of any significant defect initiation and growth mechanisms
3. The large critical defect size including a justification for ignoring weld residual stresses

Claims were also made for the relevance of monitoring the primary vessel for leaks based on any degradation of properties in the core support being the same as the primary vessel. There was a review of possibilities for inspection and monitoring of the core support structure concluding effectively that all that was reasonable was being done (which was not much). There was also consideration of secondary support options should the primary support fail (Ref. 22494.pdf).

The NII approved the restart of the reactor at that time but required various follow-up actions including justifications of the claims using fracture mechanics (Ref. 19947.pdf), assessment of deflections and movements as a result of the defects and the sensitivity of any methods of detection. These included the results of routine absorber rod exercising and using the charge machine to check the core height (Ref. 27974.pdf) and further development of methods of monitoring movement of the top of some of the outer shield rods.

A large programme was undertaken to support the claims made for core support structure as is reflected in the large number of reports during 1991/92 covering structural integrity, secondary support options and methods of monitoring (see Table 3).

Ultrasonic under sodium viewing equipment was developed in parallel to PFR operation and was used to scan the top of the core. It was used to measure the length growth of Guide Tubes (and sub-assemblies) and could serve as an indication of the continuing soundness of the core support. Although it was not a routine method of monitoring it could potentially provide accurate information on the position and any tilt of the core support structure (Ref.03491.pdf, 08322.pdf).

A list of references relevant to the Strongback is given in Table 3.

Reactor Jacket

The Reactor Jacket formed the hydraulic boundary between the inner plenum (hot pool) and cold pool which effectively was subjected to the full core temperature difference (Figure 1 and Figure 2). The IHX pods and trays were an integral part of the reactor jacket and were located in the cold pool so there was a potentially large temperature gradient within the structure. This was alleviated by insulating the Reactor Jacket with gas-filled stainless steel panels on the hot side so that it operated close to the cold pool temperature.

Although no issues were apparent during PFR operation, other than those covered in the IHX section, it was a design feature that is unlikely to be repeated because of the difficulty of assuring the integrity of the 12000 panels through the reactor life.

In 1968 it was decided to use the as-built reactor in a water test during final commissioning to check some of the major features of the primary circuit, rather than to build a large scale water model. The water test would simulate the vibration and flow distribution, the acoustic noise and cavitation behaviour. Past experience had shown that access to the primary circuit would probably be necessary during commissioning and this would be easier if sodium had not been admitted. The need for the test was further reinforced during construction as confirmation of the acceptability of core sub-assembly vibration and primary pump cavitation.

In November 1971 doubts about the water test first surfaced. Theoretical calculations had revealed that there were hole sizes where water ingress was possible into the gas panels that would prevent the egress as steam fast enough to avoid distortion by internal pressurisation during drying. So it was a major decision to abandon the water test and the main reason was the concern for the reactor jacket gas filled quilted panels.

A consequence was to provide additional sodium instrumentation. In the event, the reactor proved to be extremely free from acoustic noise and vibration induced noise, vindicating the decision to omit the test

A list of references relevant to the Reactor Jacket is given in Table 5.

2.4. Above Core Structure ACS

The role of the ACS was to support and guide the control and shutdown rod mechanisms and protect them from the dynamic buffeting in the core outlet, locate core instrumentation and provide a secondary hold-down of core sub-assemblies. Located immediately above the core it was subjected to the most severe thermal environments of all the reactor structures. At the time it was designed there was limited knowledge and understanding of how to design for such a demanding thermal environment and the phenomena of thermal striping and helium embrittlement were not known.

The ACS had to facilitate the fuel handling mechanism which in the case of PFR was a pantograph charge machine that required a “slot” in the structure to allow the machine to reach the full radius of the core, the circumferential coverage being achieved through rotation of the shield from which the ACS was suspended.

Early during the PFR construction phase a further role of the ACS started to became apparent from the results of tests using the RNL water model of the hot pool. The ACS had to be modified to suppress turbulence and entrainment of cover gas. With the knowledge from PFR and improving knowledge of thermal hydraulics this further role of the ACS was recognised as the control of the primary sodium flow distribution into the hot pool to achieve the required thermal-hydraulic conditions of the free surface to avoid cover gas entrainment.

During the life of PFR there was no evidence of the cover gas being entrained. The greatest concern from entrainment was for safety because of the reactivity change and reduced cooling that would have resulted if entrained gas had passed through the core, and worse if a large bubble of gas had been allowed to build up; for example in the diagrid, and then released to pass through the core. It was obvious that the considerable turbulence at the free surface and the build-up of a large “hump” of coolant above the core were not acceptable. The model showed the success of the addition of a flow baffle about 300 mm below the stagnant free surface level (the level rose as the core flow was increased). Although it may have been obvious that the conditions were not acceptable without the additional baffle the interpretation of the free surface observations in the water models was not a precise science. Judgements had to be made about the possibility that vortices observed as dimples on the surface of the water model would entrain gas, and the fact that gas entrainment was not an issue in the reactor indicates they were made correctly and shows the success of water modelling of the hot pool.

The success of the water models in reaching an understanding of the thermal hydraulics behaviour was increasingly aided by the development of computational methods during the life of PFR (Ref.26684.pdf, 21483.pdf, 21619.pdf).

Early operation of PFR was characterised by an unexpected number of trips, each of which caused a downward temperature ramp in the hot pool and particularly the ACS. In 1977, a detailed reappraisal of the integrity of the ACS drew attention to a number of features and the safety case was made of continued operation (Ref. 05343.pdf which references the supporting documents).

The Anti Vibration Grid (AVG) support system originally comprised ten 150 mm dia. columns each attached at its upper end to the rotating shield baffle plate by a draw bolt which was welded in position. The transient temperature and hence stress distribution within the weld was difficult to assess. An attempt was made to treat the feature in a formal manner, e.g. using the ASME code of practice but the complexity and therefore the required conservatism indicated a short life for the attachment. In view of the importance of the location of the AVG additional support was provided. In 1980 four tie rods were fitted each terminating at its lower end in a foot which hooked under the AVG structure. Each of these rods was tensioned to carry its share of the weight, the tension being measurable and adjustable while the reactor was shut down (Ref.21626.pdf and Design Safety Report A6.2.1.d was not found).

The AVG was constructed in the form of a welded honeycomb grid, bolted and welded to a star shaped outer section. The reappraisal referred to above focused on these joints and again, using ASME code case 1592 a limited life was estimated.

The AVG was also perceived to be at risk from Thermal Striping caused by the temperature difference between the jets from adjacent subassemblies which was greatest at the boundary between core and breeder. The two streams interact causing hot and relatively cool eddies to impact on surfaces. The resulting thermal cycling could have occurred with a periodicity of the order of a second, fast enough to initiate high cycle fatigue cracking within the reactor lifetime if the temperature cycle amplitude was sufficiently high. The appraisal referred to above concluded that the AVG was possibly being damaged by this process. This topic became the subject of very extensive investigation on the thermal hydraulics and material response to the high frequency temperature fluctuations (this is reflected in the large number of reference documents Table 4).

Experimental rigs were used to determine the failure mode and rate of damage accumulation as functions of the amplitude and frequency of the applied strain. Reactor models were also used to obtain detailed information about the transient temperature fields and therefore the strains which the features were expected to experience. Wherever possible measurements in PFR were used to check and sometimes normalise the rig results.

For the support columns a programme of experimental work on thermal shocks to seventeen replicas showed that, extrapolating from the worst case, the number of trips from full power required to initiate a crack was 104 and for the crack to grow to half thickness required a further 152. The next worst case required 102 for initiation and a further 333 for half thickness penetration. The results of the worst case are interpreted as giving a minimum life of 35 years at the then current rate of damage accumulation. A damage account based on measured reactor temperature transients was kept and used to assess fatigue damage and also crack growth by means of a fracture mechanics assessment. Both these routes were believed to be pessimistic and the safety case rested on experimental results backed by the redundancy offered by the additional (monitored) tie-rods.

The susceptibility of the AVG welded structure to thermal shock was assessed theoretically; a fracture mechanics crack propagation analysis was used as the basis for damage accounting (Ref.21637.pdf, 25138.pdf). The residual life was dependent on the assumptions made concerning the extent of allowable cracking. The use of Linear Elastic Fracture Mechanics (LEFM) was justified theoretically and experimentally and demonstrated an allowable design life of 44 cycles which was improved by a factor of more than 3 compared to ASME Code Case N47.

The thermal striping of the AVG was originally assessed on the basis of the limited amount of information from in-reactor instrumentation and temperature measurements made on existing out-of-pile rigs. Interpretation was hampered because the rigs were insufficiently detailed; thermal inertia of the reactor temperature sensors was uncertain but known to be high enough to modify the response. The spatial distribution of the fluctuating temperature field was not well known so that extrapolation from reactor measurements at a few points to a more general survey introduced further uncertainty. In addition the alleviating effect of the stagnant boundary layer of coolant adjacent to a surface was not quantifiable. The resultant pessimism which was built into the thermal striping assessments of the AVG might have proved to be operationally limiting on maximum power, but at the time the maximum reactor power level was restricted by the steam plant.

Improved general understanding of the thermal striping phenomenon emerged as results from purpose built rigs became available. Operational constraints were applied to the positioning of subassemblies in the core and breeder to avoid large differences in the exit coolant temperature of adjacent sub-assemblies.

Eventually the constraint became an imposed limit of 65 °C between the temperature of the exit coolant from individual subassembly and the (flow weighted) mixed mean temperature of its neighbours. This limit was set to prevent surface cracking. It was pessimistic in that some cracking was tolerable, and that failure could occur only if a crack initiation was followed by significant propagation; it was optimistic in that some surface defects might always have been present since manufacture.

In March 1982 a revised safety case was made for the continued operation of PFR based on a review of all the work that had been performed in support of the ACS (ref 18101.pdf). The document referred to an extensive set of supporting documents some of which are available and listed in Table 4. It provides the basis for the 65 °C striping limit and references the supporting work that lead to this recommendation. It also reviews the thermal shock and thermal striping loads on the ACS, the methods of assessment and their validity, the damage levels in the various components of the ACS and experimental evidence of failure. On the subject of embrittlement due to helium generation it provides a reasoned case why this would not be an issue during the reactor life and if it were to occur in the highest irradiated parts at the core centreline it showed that the structure was highly redundant and that the loading would be transferred to other parts. The report makes specific recommendations for continuing the assessment and investigative work.

In the 1984 PFR annual safety review (Ref.18103.pdf) it was possible to conclude in the section concerning the ACS as follows:

"As explained in last year's report, thermal shock aspects of the Above Core Structure are satisfactory but questions remain regarding thermal striping. For a number of years, operations have been conducted to preclude thermal striping for the anti-vibration grid AVG. However data from the relatively recent (1:5 scale) model at RNL indicates that the earlier rules have been optimistic, although since the new proposed rules are not impinged by reduced power operation on only 1 or 2 circuits there has been no issue during the period covered by this report."

This annual report said that an R&D review of the whole subject of thermal striping was in hand and that acceptance by the Safety Working Party was still awaited.

A last comment: There has been a large resource expended in establishing the case for continued operation of the ACS. It is situated in the core outlet sodium with a mixed mean outlet temperature of 560 °C. The ASME code case 1592 (later N47) was intended to extend the application of the code to high temperatures (above 427 °C) but the conservatism in the code was limiting. So there was a massive effort to understand the thermal environment and structural response of the PFR ACS. As an example using the elastic route of code case 1592 it was only possible justify a striping temperature difference between adjacent streams of a little over 30 °C but by understanding the thermal hydraulics and attenuation of the downstream temperature difference and the structural response a temperature limit of 65 °C was established to avoid initiation of cracks. It would be of great value (but expensive) to examine the PFR ACS and assess whether early operation had caused damage by thermal striping, temperature gradients, thermal shock and irradiation and whether the subsequent steps to limit damage had been successful.

A list of references relevant to the ACS is given in Table 4.

2.5. Intermediate Heat Exchangers IHX and Pods

The Intermediate Heat Exchangers (IHXs) were of the contraflow shell and tube type. They transferred heat from the primary to secondary sodium and were shielded from the core by the neutron shield rods to prevent activation of the secondary sodium. The IHXs formed part of the primary containment boundary and protected the reactor from the effects of sodium water accidents at the Steam Generators.

There were a total of 6 IHXs connected in pairs to the 3 secondary circuits. They were installed in pods attached to the reactor jacket with hot coolant from the upper plenum (hot pool) arriving at a pair of IHXs along three trays which were also part of the reactor jacket. Adjacent IHXs were connected to different secondary circuits which led to some issues when operating with less than the full complement of secondary circuits. The primary coolant flow through the IHXs was driven by the difference in levels between the upper plenum and the outer (cold) pool.

Primary coolant entered at the top of the IHX passing through the tubes whilst secondary sodium was taken via a central duct to the bottom of the tube bundle and passed on the shell side over the outside of the tubes. The decision to have the primary sodium inside the tubes was largely influenced by the pressure loss which was predictable and constrained by the sodium pool level difference. With secondary sodium on the shell side, there is greater freedom to manage the flow distribution at the expense of an increased pressure loss. A sleeve valve was mounted at the entry to the IHX to isolate the primary flow in the event that operation was required with the associated secondary circuit shut down. Another feature of the IHX was that it had a decay heat removal heat exchanger in the form of a coil located immediately above the tube bundle in the most advantageous position for inducing natural circulation.

The IHXs were designed to be removeable and were cylindrical so that they could be removed through penetrations in the reactor roof. For convenience and compactness the secondary sodium entered the IHXs through concentric ducts.

Although there was little experience of such large stainless steel heat exchangers they proved reliable components throughout the life of PFR. However a number issues arose giving rise to concern about the integrity of the IHXs leading to extensive investigation including specific thermal hydraulic modelling and analyis. These issues can be summarised as follows:

1. Reactor trip causing drop in primary sodium temperature affecting the top tubplate
2. Secondary trip causing a rise in secondary sodium temperature being returned to the IHX affecting the bottom tubeplate and pod in the reactor jacket
3. Thermal striping caused by leakage bypass of the isolation valve affecting the bottom of the IHX and pod
4. DHR pipe cold leg as it entered the hot sodium
5. Corrosion as a result of high oxygen content in the secondary sodium

The tube to tubeplate welds which were the source of such problems in the Steam Generators remained sound. Although sensitivity to leaks is not the same for the IHXs even a small leak would have been revealed over such long periods of time by coolant inventory change. The tubes of the heat exchanger were essentially straight with a sinusoidal bend located towards the low temperature end to accommodate tube to tube and tube to shell temperature differences.

For a period of 19 days during 1981, sodium in the secodary circuit had a high impurity burden. The oxygen was eventually cold-trapped to less than 10 ppm, but the high level, ca 85 ppm, might have been present in the circuit while still hot, 530 °C. Estimates were made of the likely levels of corrosion in the tubes and possible implications. It was concluded that in the event of a violent sodium/water reaction following a large leak, any degradation in the tubing would have led to only a limited leakage, generating insuffient pressure to damage the reactor containment. It was also concluded that the tubes would not have suffered significant corrosion during the 1981 incident. The accumulated corrsion damaged was assessed and reported after each reactor run (Ref. 13398.pdf, 16048.pdf, 19942.pdf and 19945.pdf)

Attention was paid to the possibility of significant thermal shock to the top and botttom tube plates from multiple trip action and thermal fatigue resulting from asymmetry in coolant flow paths when one or two of the secondary circuits were not operational. A damage accountancy procedure was implemented

A trip from the steam side resulted in seccondary sodium being returned to the IHX at a high temperature thereby applying a rising temperature ramp to the bottom tubeplate and also to the weld connection between the IHX pod and the reactor jacket. A trip from the reactor side led to a fall in temperature of the primary coolant at the IHX inlet, and hence a falling temperature ramp to the top tubeplate. A water model of the hot pool was used to establish the temperature ramp at the IHX (Ref. 07887.pdf).

To promote mixing of the secondary coolant each of the three secondary circuits was fed by a pair of IHX from differing pods.

Extended operation with one or two of the secondary circuits shut down could lead to flow patterns with fluctuating temperatures, a phenomenon known as thermal striping. The isolation valve at the primary inlet to each IHX associated with the shut-down secondary circuit would be closed. In practice there was a leakage bypass of the IHX isolation valves which was investigated with various measurements and tests on the reactor (Ref. 01377.pdf) and estimated to be in the range 0-10%. It is important to appreciate that the leakage was not a failing of the valve but came from the blanking plates between the support straps of the IHX above the valve. Each IHX isolation valve was a sleeve valve with piston ring seals, and above the sleeve the openings between the support straps were blanked off with bolted plates. The hot leakage flow traveled through the un-cooled tubes of an IHX associated with a non-operating secondary circuit and emerged to mix with cold sodium from the adjacent operating IHX, with the potential for causing fatigue to the tube to tubeplate welds.

A second potential source of thermal fatigue conditions arose from the way that the IHXs were paired in their pods. When one secondary circuit was shut down, each of the two pods which housed the relevant IHX operated with one out of its two IHXs nominally valved off. In these circumstances cool sodium from the pod with two operational IHXs could flow through a pod inter-connecting pipe and mix with coolant in a pod containing a nominally valved off IHX, the latter being at a higher temperature. This condition was more severe when, for a few months during early commissioning, PFR operated with only one secondary circuit and one of the pods had both IHXs valved off. The pod to reactor jacket weld was potentially at risk.

Extensive experimental work, mostly on rigs using water (Ref. 01146.pdf) but some with sodium (Ref. 04009.pdf), and theoretical studies (Ref. 01550.pdf, 02184.pdf, 02187.pdf) were undertaken and operational limits for example on the core temperature rise was set and adjusted as confidence rose. Eventually it was shown that the potential shock and thermal striping amplitudes were not realised in practice. Convective and conductive heat transfer between hot and cold regions in the coolant and the attenuation through the liquid boundary layer were among the important relieving mechanisms. A damage account was maintained to monitor the accumulated shock damage to the ligaments in the bottom tube plate. It was not a life-limiting feature. The structural analyis of the pod/reactor jacket welds were presented in reports (Ref. 25177.pdf, 21641.pdf, 05350.pdf).

In 1983, while the outcome of the investigations were uncertain, the design of replacement IHX units was initiated (Ref. 4012.pdf). The design was to have minimum change but encorprate improvements to enhance margins for endurance judged against structural codes. The thermal and hydraulic performance of the replacements had to be similar to the orginal units. The improvements that were made to the design were a demonstration of the knowledge acquired from the PFR experience. These improvements which are presented in the System Design Specification for the spare IHXs (Ref. 04977.pdf) include the connection between shell and tubeplates. They also include measures to overcome the isolation valve bypass leakage and recognise the more greater emphasis on monitoring and ISI (Ref 01784.pdf)

When the order for the replacement was placed in 1985 it was decided to limit this to one unit. The IHX was never installed, which for the PFR operators was welcome, but regrettable that the extensively instrumented tube bundle could not yield valuable information for future design, understanding and validation of thermal hydraulic codes.

In the event of a large sodium/water reaction, a consequential failure of the IHX shell could prejudice reactor safety. In 1991 the IHX became the focus for the attention of the Structural Integrity Advisory Group because, in principle, the cracking mechanism identified in the SGU vessels could also occur in the hot regions of the IHX. Because this component was not available for inspection it was necessary to make a leak-before-break case, supplemented by a probabilistic approach, to verify that the risk of IHX failure following a sodium/water reaction was acceptably small (Ref. not found).

The concern for the DHR pipe cold leg as it entered the hot sodium arises from the startup of the DHR loop when cool NaK (150 °C) entered the heat exchanger coil when the primary sodium could be at a high temperature (up to 550 °C). This gave rise to a large through-wall temperature gradient at its highest where the pipe passed into the primary sodium, and was exacerbated by sodium level changes. The concern resulted in a recommendation not to use the DHR system for routine operations (e.g. cooling the IHX valve leakage flow) so was not life limiting (Ref. 21488.pdf). It was a feature that was improved in the spare IHX design by adding a thermal sleeve over a section of the pipe (Ref. 04012.pdf, 04976.pdf, 04977.pdf, 24974.pdf, 25202.pdf). The thermal sleeve was included on both the inlet and outlet pipes because the DHR flow occasionally reversed.

There is evidence that radioactivity built up in the lower, cooler, part of the IHX particularly when fuel leaks occurred. It was believed that absorption of active fission products species occurred in the cooler part of the IHX tubes whenever fuel sustained a gas leak. The activity decayed during long shutdown periods.

Footnote:

The IHX pods were an intergral part of the Reactor Jacket which formed the hydraulic boundary between the inner plenum (hot pool) and cold pool. The IHX units were sealed to the pods using piston ring seals which is an arrangement that gives the most compact primary circuit. There has been interest in using the piston seals concept in future designs because it is a compact solution although probably not with the Reactor Jacket as designed for PFR.

An extensive list of references relevant to the IHX, IHX pods and Reactor Jacket is given in Table 5.

2.6. Decay heat exchangers

The Decay Heat Removal Loops are covered in section 4.1 and the sodium/NaK heat exchanger coils are integrated in the IHX units and covered in section 2.5 above.

2.7. Primary pumps (including Secondary Pumps)

The 3 primary pumps were of the single-stage centrifugal-flow type suspended vertically from the roof and driven by squirrel-cage induction motors through fluid couplings. The shaft and impellor had two hydrostatic bearings. The lower bearing was immersed in sodium and fed by pressurised sodium and the upper journal bearing was lubricated with oil Figure 5 and Figure 6.

There were also 3 secondary pumps each one serving one secondary circuit.

The primary pumps had pony motors driven from guaranteed electrical supplies to ensure flow at all times. They were coupled to the pump shaft by a clutch which engaged when the pump speed fell below 10% of full speed. There was also a flywheel mounted on the shaft to provided added inertia to prolong the rundown of the pumps following a reactor trip caused by the of loss of electrical supplies to the pumps. A description of the system extracted from the PFR Station Manual is given in a report (Ref.23203.pdf) which is part of a submission to the NII in 1992.

The pump lubrication oil system was originally designed so that, in the event of bearing seal failing in service, the quantity of oil which could leak from the bearing to the pump drain tank would not exceed the capacity of this tank.

As well as the design measures to prevent oil ingress into the primary sodium, as a precaution against an undetected leak, an in-line carbon meter was developed at AERE which involved the continuous measurement of the rate of diffusion of carbon through an iron membrane. This provided the operator with direct guidance about the activity of carbon in the primary circuit (Ref.28020.pdf).

Each of the pumps had a filter installed as a precaution to collect any debris during commissioning with the intention that it was to be removed after commissioning.

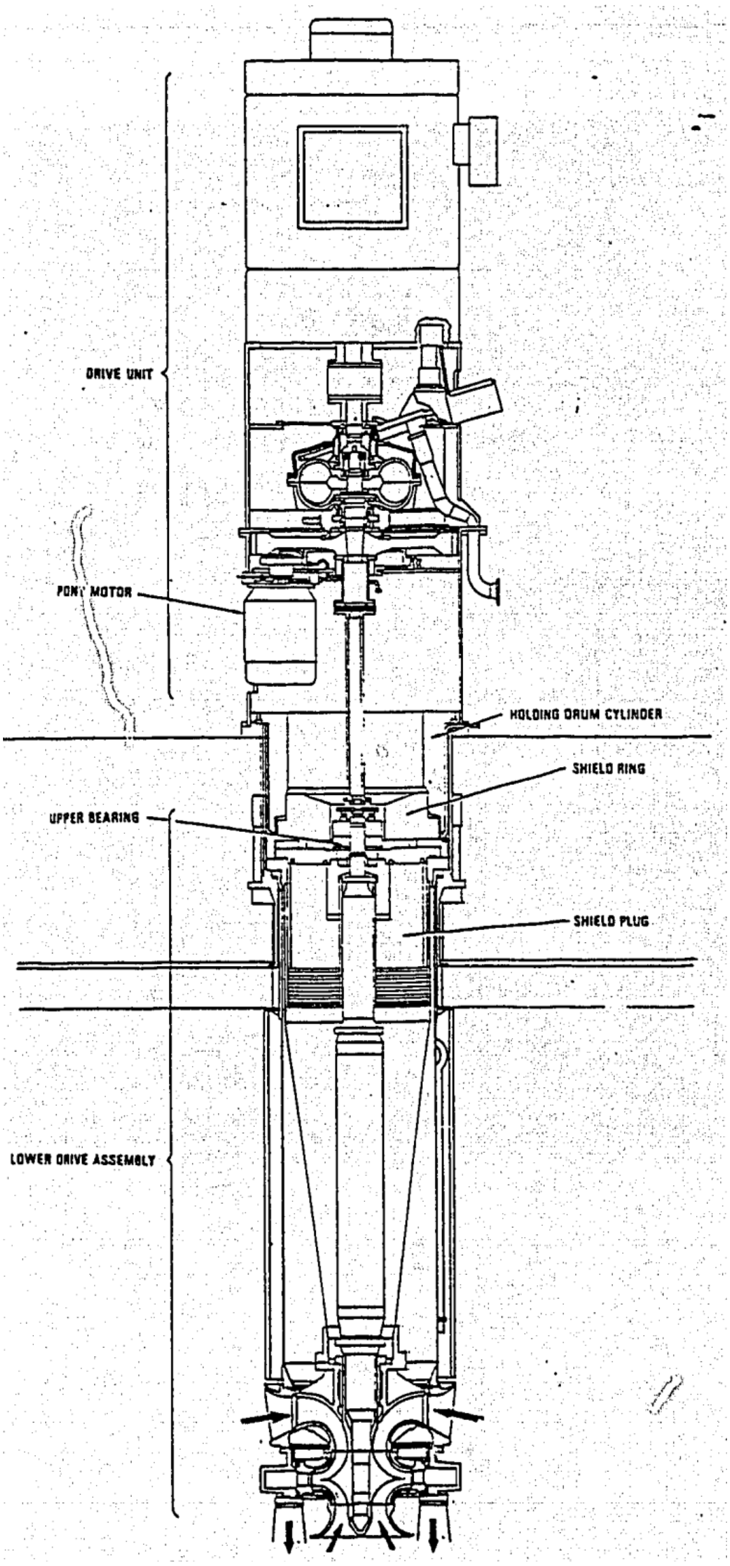

Figure 5: Primary Pump

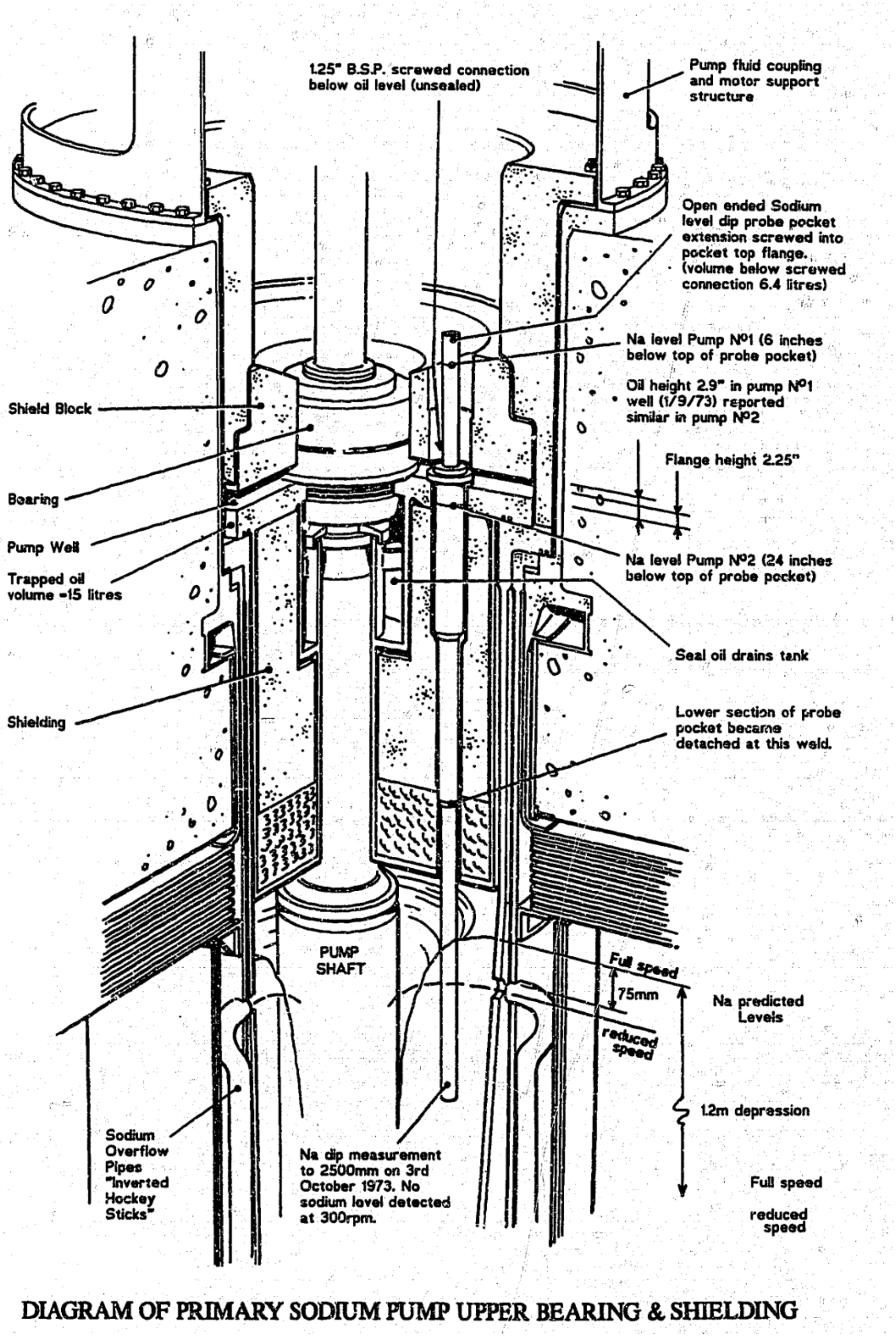

Figure 6: Primary Pump Bearing and Seal Arrangement

The pumps proved to be reliable components with the exception of 2 issues one of which occurred during early commissioning in 1973 and the other causing oil ingress into the primary system on 3 occasions.

During commissioning the shaft of one of the primary pumps seized. This was caused by the shaft rubbing on the bore of the drain tank in the pump support due to a combination of inadequate clearances and thermal bowing of the shaft. The pumps were located in the cold sodium pool, but immediately above the pool surface one side of the pump was adjacent to the reactor jacket which was at a higher temperature giving rise to a temperature gradient across the shaft. This caused bowing only while the shaft was stationary, and this occurred for a short period after shutdown from high power. The design intention was to have a greater clearance around the shaft but an adverse combination of the tolerance limits gave rise to contact between the shaft and housing. The bow of the shaft was increased due the rubbing contact causing a hot spot further increasing the loading on the bearing and eventual seizure.

The spare pump was installed and all the pumps were removed from the reactor sequentially and modified. Bowing was also avoided by maintaining a slow rotation of the pump shaft for the period after shutdown. There were no further difficulties during PFR operation.

Another problem arose during commissioning when an inspection revealed that the pawls in each of the primary pump pony motor clutches were worn and one had failed, probably as a result of being used in an unusual orientation. The pawl profile was modified and subsequently routine inspections showed the fault was cleared.

The three primary pumps operated successfully from 1974 until 1991 when the oil lubricated top bearing failed in one pump allowing oil to enter the primary sodium circuit. There were a number of concerns:

1. Reactivity addition as the hydrogenous material passed through the core.
2. Carburisation of the primary circuit.
3. Blockage of narrow sections of the coolant path particularly through the fuel elements.

In 1974, during early commissioning, an oil leak was caused by failure of the sodium dip level probe tube pocket. It occurred whilst the pump tank argon cover gas pressure was low and there was oil above the upper flange of a screwed connection and subsequent leakage of the bearing running seal (Ref. 23047 pdf). The flange was modified to prevent future leakage on all the pumps. The oil may have entered the primary sodium but the amount is unknown but an upper estimate was 60 litres. At the time the reactor was loaded with the dummy core so no fuel was present.

A further problem did occur during 1974 when a sodium level probe fractured due to flow induced vibration. Although the level probe was located in a thimble in the pump it was not a specific issue for the pump.

In 1990 blockages in the argon blanket hampered the removal of excess oil from the oil drains tank and although oil probably did not enter the primary circuit, modifications were made to the argon and oil removal systems and to plant and operating procedures, and collectively these prevented further oil incidents of this type.

The primary cause of the 1991 contamination was a blockage in the filters of the main argon cover gas circuit. Subsequent manoeuvring of this gas pressure had an unexpected reaction on the gas pressure local to the pump and as a result oil contaminated the sodium. The amount was estimated to be 35 litres. This incident had a greater impact than previous leakages because it was immediately observed that the profile of the core subassembly exit temperatures was skewed. This caused concern, in the case particulate matter had affected the flow split between individual subassemblies, in which case a subsequent redistribution of the debris might, conceivably, have led to a damaging subassembly blockage.

A review of the design and operation of the lubrication system was undertaken (Ref 21220.pdf) and served as the basis for the safety case for continued operation. It concluded that that the system was designed to work within the Design Safety Report but did not allow the plant operators enough operational margin. Steps taken by the operators to make the system operational, in the worst case, meant that the volume of oil in the system exceeded what could be retained by the bearing oil drain tanks in the event of a bearing seal leak. Design and operating procedures changes were proposed (Ref.26768.pdf). A complete review of the system including a reliability assessment was performed (Ref 21210.pdf).

An extensive programme of investigation was undertaken. Monitoring of the primary pump motor current indicated blockage of the filters (Ref.19669). Studies of the effect of injecting oil of the correct type into sodium upstream of a PFR subassembly filter were carried out by a team of Russian scientists at Obninsk. In addition, an extensive series of experiments into the build-up of particle blockages in a subassembly, and the resulting consequences, was undertaken by NNC. An assessment was made of the likely extent of carburisation in the primary circuit and the effect on the structural integrity of the thin-walled IHX tubing (Ref.23166.pdf). Tests were undertaken at Harwell (Ref.22248pdf) and Dounreay (Ref. 22247.pdf) to establish the effect of irradiation at temperature in the presence of sodium on the oil. The pins from 2 subassemblies which had shown high outlet temperatures were examined and found to have surface deposits resulting from the oil ingress (Ref.23895.pdf, 25428.pdf). These assessment provided confidence that no hazard from carburisation was present.

The primary pump filters located in the primary pump valves were not removed after commissioning, presumably because the pressure drop across the filters was small. The filters were not designed to have a large pressure drop so incurred some damage when deposits resulting from the oil ingress were trapped. The filters were removed and inspected to assess the extent of damage and blockage (Ref. 22246.pdf, 23527.pdf, 25429.pdf). Despite being originally intended only for commissioning the meshes were replaced with filters designed (Ref.23941.pdf) and tested (Ref. 23894.pdf) for a higher pressure drop and the renovated units were reinstated. This was a major engineering task for which there was no previous experience. It was completed without mishap. The removal of the filter/valve assemblies changed the flow conditions in the primary circuit which required submission to the Safety Working Party (Ref 21267.pdf) for a change to the PFR Operating directive. Subsequent inspection of the filters showed that they had not constituted a safety problem.

The PFR Primary sodium pumps oil management system, history and strategy for the future (Ref.23203.pdf) and peer review (Ref.23739.pdf) was prepared for the NII in 1992. The reactor was cautiously returned to power with no detectable detrimental effect from the oil ingress. The incident is believed to have imposed a fifteen month delay on the operational programme.

The three secondary pumps also proved reliable components with only 2 incidents during their operating life which required the removal for repair. In 1973, during commissioning, and in 1984, spray-fused stellite coatings on a pump shaft sleeve became detached causing seizure.

In total the primary pumps and motors accumulated more 430,000 hours of operation and the secondary pumps 300,000 hours.

A list of references relevant to the Pumps is given in Table 6.

2.8. Primary vessel

The PFR Primary Vessel was a thin shell that was relatively lightly loaded containing the primary sodium and low-pressure argon cover gas blanket Figure 1. It had an important safety function as part of the primary containment. There were no connections or attachments to the vessel below the sodium level and only the connection with the core support skirt in the cover gas space at low temperature immediately below the Reactor Roof. The claimed high structural integrity was based on this simplicity combined with a demonstration that it satisfies the requirements of ASME Section III design code for nuclear vessels operating at temperatures below the creep range.

In the autumn of 1975, it became apparent that the design features were not producing the temperature gradients that had been predicted in the region of the sodium free surface. This became the infamous Primary Vessel Top Strake problem where the temperature gradient immediately above the sodium surface moved axially as the sodium level changes causing cyclic thermal loadings. As a consequence further movements of the sodium level were prohibited from March 1976 until a detailed assessment was made.

The main features of the Primary Vessel top strake region, core support skirt, tuning fork and roof diaphragm connection are shown in the (Figure 7) (extracted from Ref. 18098.pdf). The top of the primary vessel was at roof temperature, about 50 °C. The vessel temperature at and below the sodium level followed closely the sodium outer pool temperature, about 400 °C. The shape of the temperature profile between these steady values was determined by the combined effect of the tapered external Rocksil insulation and the forced cooling from the argon roof cooling supply.

The argon roof cooling gas was admitted to the various roof penetrations, from where it flowed down the penetration annulus, radially outwards across the upper surface of the diaphragm, up over the inner leg of the tuning fork, down over the outer leg and into the narrow annular passage bounding the outer face of the Rocksil insulation (Figure 8). The outlet from the passage was at the bottom of the Rocksil pack from whence the argon flowed up the annulus inboard of the leak jacket and exited through the 10 radial outlet ducts.

The temperature profile was monitored at six equispaced circumferential positions by a vertical row of 6 thermocouples on a 53 cm. pitch. During commissioning the temperature profile was found to have an unacceptably steep gradient in the region just above the sodium level. This resulted in local thermal cyclic stressing as the sodium level changed in response to normal operational manoeuvres and to trips.

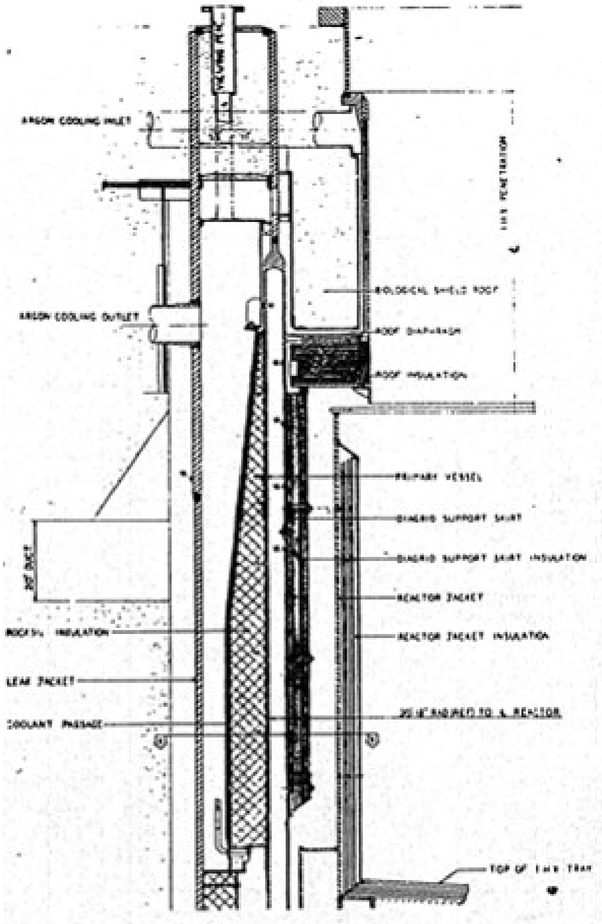

Figure 7: Primary Vessel Top Strake

The severity of the temperature profile could be ameliorated by reducing the roof cooling flowrate (Ref. 27588.pdf) and the consequences for the roof diaphragm deflection were acceptable. So from July 1976 the top strake operated with reduced cooling flow and a better temperature profile.

To improve conditions further, pipework and valves were installed to divert the roof cooling flow from the small plenum just above and outboard of the tuning fork. The diverted flow was led upwards through the roof in newly installed penetrations and piped to join the radial outlet ducts external to the reactor vault (Ref. 27565.pdf). This system was commissioned in May 1977.

Axi-symmetric analysis of the original condition predicted ratcheting of the primary vessel in an inwards radial direction. But visual examination did not reveal any perceptable change. Linear displacement trasducers known as “Pogo Sticks” with remote indication of their tip position were installed at five stations round the the vessel and at thirteen vertical positions. Pogo Sticks measured the radial distance between the reactor vessel and leak jacket. Interpretation of the sensor readings has not been completely straightforward but they have always indicated a negligible permanent set when read in the off-load position. By 1985, all surveys had revealed neglible displacements (less than 5 mm) and the monitoring ceased.

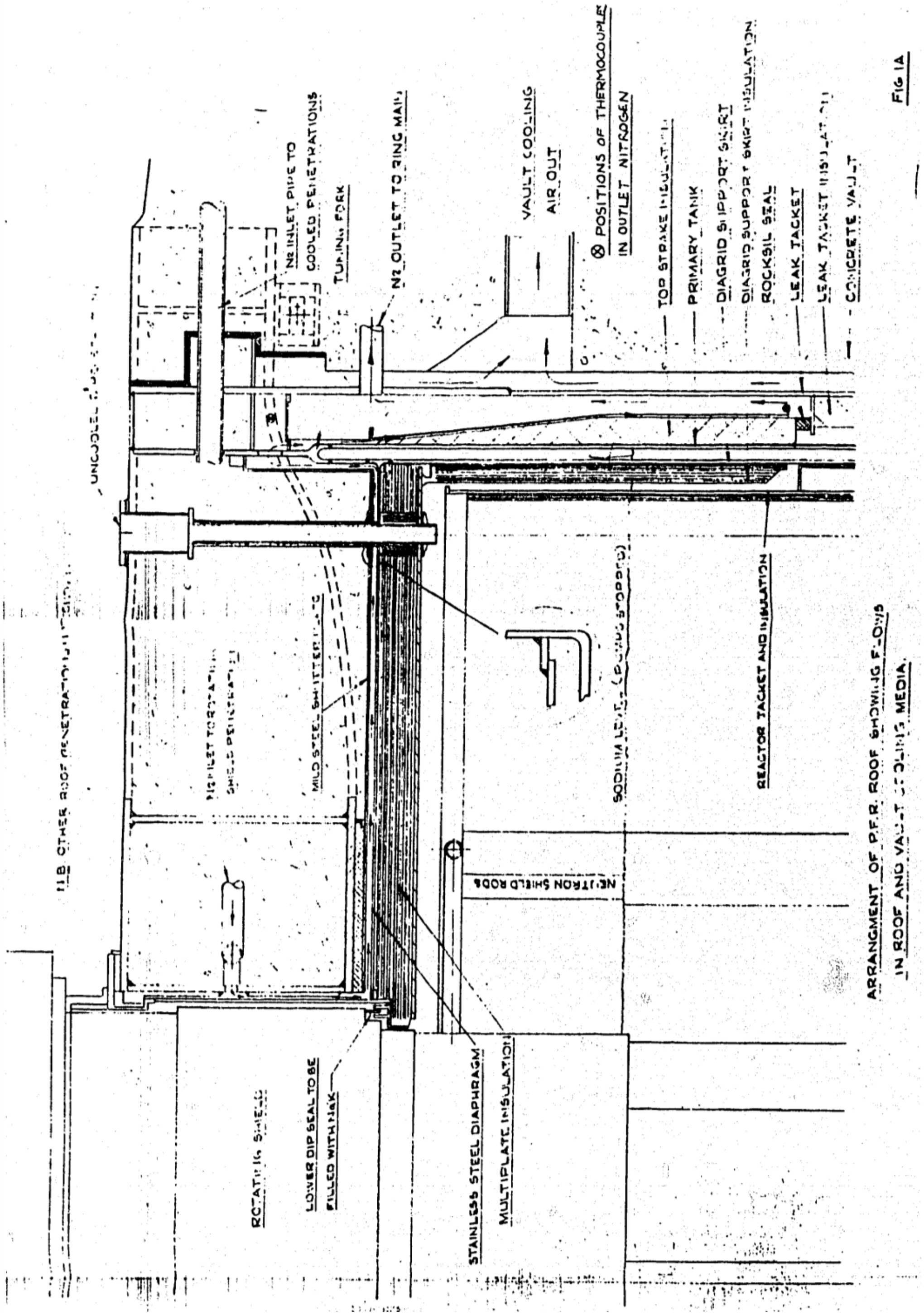

Figure 8: Reactor Roof Arrangement

The modified roof cooling, coupled with the operating directives designed to minimise sodium level changes, substantially reduced the accretion rate of damage to the primary vessel. Based on fatigue analysis, the damage accumulated prior to the cooling change was 23.5% of the design limit (Ref. 25935.pdf). After April 1976 the increment was 3.2%, and the increment recorded in the PFR Annual Safety Review 1989 was negligible for 1988/89 (Ref.22418.pdf)

In 1977 a reappraisal of the implications of the added roof cooling capability resulted in a recommendation for the provision of a system for protecting the primary vessel from excessive pressure differences. A lute filled with di-methyl silicane was installed in 1979 (Ref. 15912.pdf). The pressure difference between the gas blanket and roof cooling argon was limited to 0.1 bar, gas from whichever side was in excess of this would bubble through the lute while preserving the gas seal.

A list of references relevant to the Primary Vessel is given in Table 7.

2.9. Roof and Roof Cooling System

Reactor Roof

The Reactor Vault Roof formed part of the primary containment and supported the reactor components and Rotating Shield (Figure 8). It operated at low temperature (50 °C) so was protected from the high sodium pool temperatures (inner pool 560 °C) by stainless steel multi-plate insulation on the underside in the cover gas blanket and cooled by gas to maintain acceptable temperatures. The reactor roof was a structural steel fabrication with concrete shielding. Personnel access was required on top of the roof.

Apart from the considerable difficulties that were encountered during the fabrication of the roof the main feedback of interest concerned the cooling system.

The roof structure comprised an inner and outer ring beam with radial spokes between them. There were cylindrical penetrations for the components and a diaphragm separating the roof cooling gas from the reactor cover gas blanket. The structure was filled with concrete for shielding.

During fabrication cracking occurred after welding due to lamellar tears which was attributed at least partly to the choice of material. The material used i.e. BS1501/221/32A was proposed by the contractor as an alternative to that specified by UKAEA. This was because, in the fabricator's experience of the specified material, BS1501/224/32A, it could be dirty and give trouble in welding, although this view was not shared by all fabricators. The fabricator's specified material was accepted by the UKAEA with assurances that it had the same mechanical properties. Overcoming the welding problems involved various measures to alleviate the lamellar tearing including buttering of joint faces and inspection at various stages of the welding. Questions were raised about the structural integrity of finished structure which was addressed in the PFR safety report extract (Ref. 25178.pdf).

One conclusion from this experience is that any further fabrication of this type would be made from vacuum degassed steel to the original specification which has proved to be free from lamellar tearing problems in experimental constructions (Ref. 27648.pdf).

The reactor roof fabrication problems were the main cause of the delay in the PFR construction programme. In 1968, less than 18 months after the main contract was placed, the primary circuit was 12 weeks late. This delay continued to extend and was 20 weeks by

May 1968 and in February 1970 the partly fabricated structures were removed from the contractor's works and taken to the PFR site for completion. By the time it was lowered in place in August 1971 it was almost 3 years late. It has to be noted that the delay gave time to resolve a number of issues not least amongst these were the SGU weld problems and understanding the neutron induced voiding in material close to the core and the design changes that were necessary as a consequence.

Roof cooling system

The requirement is for the cooling system to maintain the concrete temperature and temperature gradients within the structure within acceptable limits.

The roof cooling system performance was assessed prior to reaching full sodium temperature (late 1973) which resulted in recommendations for the control of the cooling system and included limiting the reactor temperature to 300 °C. The critical issue was recognised as the roof diaphragm and the expansion at the connection to the small penetrations at the periphery (Ref.27648.pdf). An increase in the heat removal capacity was proposed and flow and temperature control which would aim at limiting the cyclic damage to the roof diaphragm (Ref. 27648). This was based on nitrogen as the cooling gas.

Initially nitrogen was selected as the coolant gas but this was changed to argon because of the concern about corrosion due to the radiolytic formation of nitric acid in the presence of low levels of moisture (Ref. 21319.pdf). Subsequently the results of the initial corrosion tests were showing that this decision could have been unnecessarily conservative.

A detailed analysis was performed of the reactor roof diaphragm semi-rigid connections between the diaphragm and some of the smaller penetrations. As a consequence fatigue tests were performed on the semi-rigid connections. A detailed review of the tests (Ref. 21319.pdf) showed that failure would occur in excess of 2000 cycles compared to an estimated 650 cycles during the lifetime of the reactor. The argon flowrate was controlled to maintain the roof diaphragm expansion at an approximately constant value as measured by three movement transducers installed in special penetrations.

Subsequently changes were made to the roof cooling system to alleviate the primary vessel top strake temperature gradients and these are described under the primary vessel section above. The roof cooling gas exits the roof over the top strake and the changes that were made reduced this cooling (Ref. 27588.pdf). Design changes were also made to the way the coolant gas was routed from the roof requiring additional pipes, penetrations and diverter valves (Ref. 27565.pdf).

A list of references relevant to the Roof is given in Table 8.

3. Control System

3.1. Absorber Rod Mechanisms

PFR had two types of absorber rod; one was the control rods and the other the shutdown rods Figure 9 and Figure 10. There were 5 control rods (worth collectively about 13$) located in ring 4 inside the core region which controlled the reactor power by vertical positioning. There were also 5 shutdown rods (also worth about 13$) and located in ring 10 on the core/radial breeder boundary. Both sets of rods were released on a reactor trip and allowed to drop into the core under gravity. This section of the report covers operational experience of the actuation mechanisms for the rods. Although the experience with the absorber rods is not covered in this section the design of the rods did evolve during the operating life of PFR along with their guide tubes (see core support section of this report) and had an impact on the actuation. The last design update was for Mark IV which had much greater clearances to allow for bow.

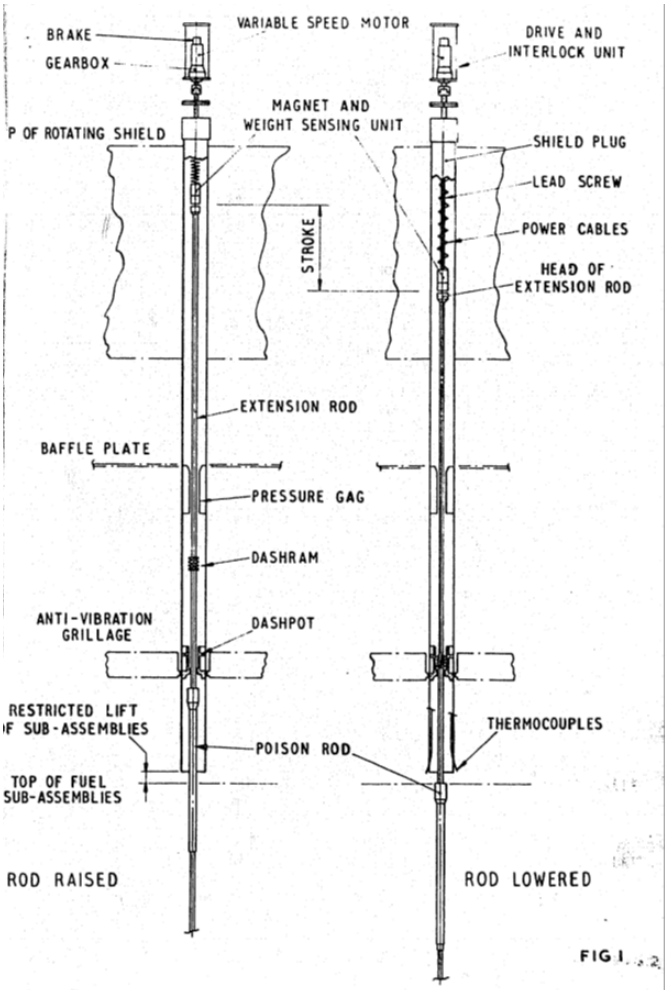

Figure 9: Absorber Rod Mechanism

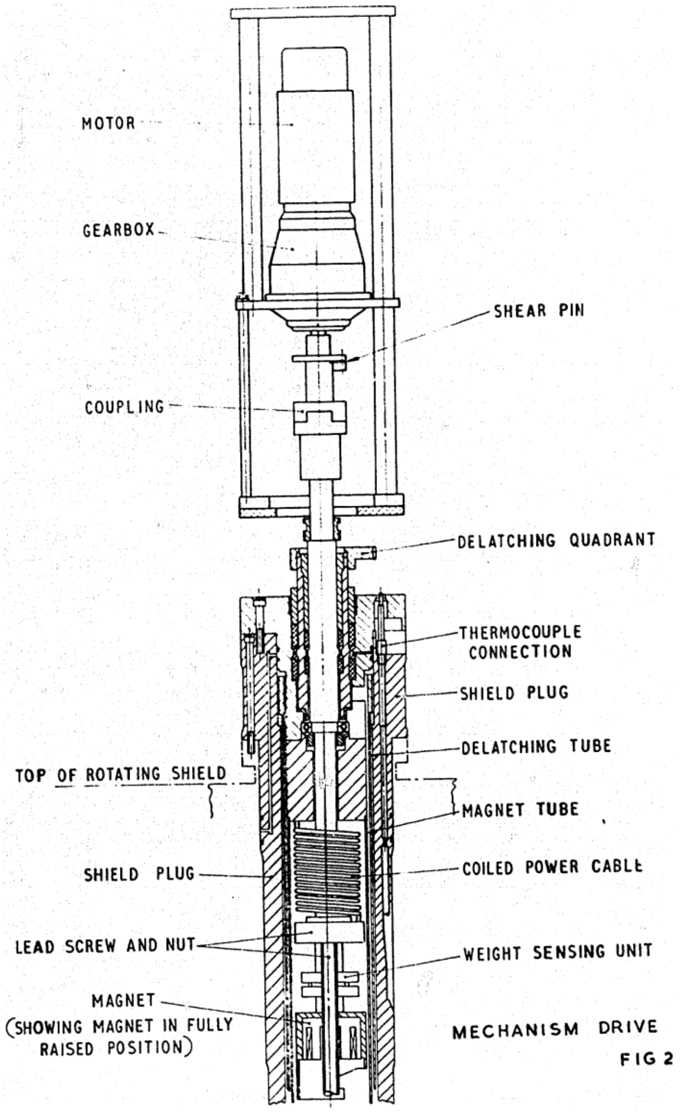

Figure 10: PFR Absorber Rod Mechanism

Each of the absorber rods, with its fairly flexible extension rod, moved within two vertical guide tubes. One suspended from the rotating shield, the other latched on to the core support structure. The pairs of guide tubes were designed to be vertically aligned at full power. During commissioning a deliberate misalignment of up to 25 mm was temporally applied and shown to give no detectable hindrance to a falling rod. Subsequently this aspect received a lot of attention as neutron induced voidage caused bowing of the lower guide tubes.

The absorber rod actuation mechanisms consisted of a drive unit, lead screw, magnet and weight sensing unit, extension rod and delatching tube (Ref. 27781.pdf). The absorber rod was driven up and down by a motor/gear box which sat on top of the rotating shield, and drove a lead screw. This rotated at constant height causing a captive nut to rise and fall carrying with it an electro-magnet. The magnet when energised retained an extension rod on the lower end of which was the absorber rod. Further information can be found on the magnet and ball-screw design (Ref.26719.pdf, 25495.pdf & 27780.pdf), the manufacturing specifications (Ref. 24290.pdf, 25490.pdf, 25491.pdf, 25492.pdf & 25495.pdf) and information on the absorber rods is contained in a design report (Ref. 26719.pdf).

The performance of the control and shutdown system was assessed at the design stage (Ref.02756.pdf). The reactivity worth of each rod as a function of its vertical position with respect to the core was measured during commissioning and agreement with prediction was confirmed. The dimensions of the PFR core are only a few neutron mean free path lengths so that it is tightly coupled neutronically and the shape of the flux distribution was little modified by control rod movement; this ensured that the reactivity controlled by the rods summed almost linearly. Interaction effects between rods were relatively small and easily measured, especially with the reactor close to critical; the appropriate adjustment to the calculated reactivity was adequately predictable.

The effect of differential thermal expansion on the reactivity coefficients arising from the core support structures being close to the core inlet temperature and the actuation from above the core at core outlet temperature was assessed (Ref.03639.pdf) and investigated through a series of measurements on the reactor (Ref.03849.pdf, 03851.pdf).

The concerns and issues that arose during the operational life were mainly related to two phenomena:

1. Neutron induced bowing of the lower guide tube and the effect this had on the alignment with the actuation mechanisms, and;
2. Sodium aerosols which deposited on the magnet faces, resulting in rods falling off their magnets causing reactor trips (a total of 20 during the operating life).

The consequences of neutron induced voidage were first appreciated during the reactor construction and this led to a series of absorber rod and guide tube modifications to accommodate the swelling and consequential bowing of the wrappers. Rotation of the lower guide tubes and their associated absorber rods was identified as a requirement and a first assessment of the frequency of rotation was made (Ref. 13160.pdf). It also emphasised the importance of the absorber rod weight sensor, which was incorporated as part of the actuation mechanism, to check that the rods movement was not being impeded.

Two steps were taken to provide confidence in the continued ability of the rods to perform their safety function; one to move as quickly as possible to have control and shutdown rods of varying ages in use and the second to exercise the rods periodically each in turn over a selected range to demonstrate their free movement. Reactivity compensation was provided by movement of some of the other rods with little impact on the flux shape as mentioned above. The rod weight was recorded during the exercise to check that there was no significant friction between the rod and guide tube and therefore that the rods were free to drop. The time taken for each rod to become fully inserted was always recorded for each reactor trip confirming the free movement under gravity of each rod. Only small and unimportant changes to the drop time were recorded, associated with deposition on the electromagnet from which each rod was suspended.

During exercising increased friction was observed as a difference between the true weight and apparent weight of an absorber rod. In the range between the operating position and full insertion, the friction force was always a small fraction (<10%) of the rod weight and so a successful drop was always achieved. There were many studies of the bowing limits for free entry of the absorber rods (e.g. Ref. 27965.pdf and 27970.pdf). A water model was built in REML to investigate the effect of misalignment between the guide tubes and the friction loads for various offsets (Ref.12328.pdf). Friction measurements were interpreted with the assistance of a computer code PEBBLE (Ref. 06715.pdf and 08988.pdf) which used algorithms to relate neutron induced swelling and bowing to neutron dose. It calculated normal loads at contact points and converted them to friction forces using a surface friction coefficient. The exercising programme for each operational run was prepared in the light of results from the previous run and had to be approved by the Safety Working Party. (Ref. 18774.pdf for run 22 is a typical "exercise schedule" with others listed in the table below.)

Design improvements were made sequentially and the Mark IV rod and guide tube design had much greater clearances to allow for bow. A spring was incorporated as a stabiliser because of the increased risk of vibration due to the greater clearances and flow induced vibration was investigated in a water model (Ref.13616.pdf). PEBBLE indicated that the life of an absorber might be limited by pellet swelling rather than bowing (Ref. 13594.pdf).

Aerosols arise because the reactor roof is cool and immediately above the hot sodium pool (550 °C). The build-up of sodium aerosol deposits on the absorber rod actuation mechanisms and particularly the magnet faces was prevented by an argon purge. The amount required for prevention was investigated experimentally (Ref.04339.pdf) and recommendations made for a steady state flow with an increase in flow when the magnets released the rods (i.e. electrical holding current was cut) to prevent a partial vacuum being filled with aerosols as the rods drop (Ref. 03639). The purge flow to each absorber mechanism was not monitored so confirmation that the desired purge flow was being achieved was not possible and aerosol deposition continued to give some problems.

Sodium aerosol deposition has been a problem on most if not all sodium cooled fast reactors with diverse solutions proposed (Ref. 23830.pdf). A hot argon purge flow may have been more effective but was not compatible with the resin bonded shot in the PFR rotating plug shielding.

For the first seven operating runs the reactor power was relatively low. Friction measurements were usually consistent with prediction. A general trend to higher friction forces then became apparent. Replacement of the absorber rod or guide tube did not systematically reduce the high friction. It has since been established that the main source of friction derived from the deposition of sodium aerosol on the extension rod in the gas blanket region, especially at geometric discontinuities (e.g. keyways). The increased friction became apparent at about reload 7 when the reactor power level was raised, increasing the concentration of aerosols in the argon cover gas.

The aerosols also penetrated as far as the electro-magnets that hold the absorber rods allowing small deposits to develop between the magnet faces. This made the rods prone to dropping off adventitiously requiring an increased current to avoid drop off. The inconvenience was effectively countered by cleaning the magnet faces about once a year. This was another procedure that required endorsement by the Safety Working Party. (Ref. 19940.pdf is a typical Maintenance Inspection and Test Schedule). Another effect was that the magnet release times could increase. Although the increase was only a fraction of second it was another indication that the magnet should be cleaned. Release times were therefore studied whenever a trip occurred.

Neutron induced voidage swelling and sodium aerosol deposition together account for the high friction and delayed magnet release of the absorber rods. Other anomalous behaviour has been observed on shutdown rod 3, shutdown rod 4 (Ref. 23067.pdf), and during the early stages of reactor operation, an intermittent minor fault was observed on several rods. On two occasions the drop time for shutdown rod 3 was abnormally long but in each case analysis of the power transient suggested that the first part of the fall, where most of the reactivity insertion occurs, was close to normal. This fault was believed specific to the rod and guide tube in shutdown rod 3 position at that time.

Shutdown rod 4 experienced high friction forces and it became evident that the interference was below the sodium level, ruling out sodium aerosols as the primary cause. The unusually high and increasing friction occurred from October 1987 until the absorber and associated guide tube were replaced in May 1988. PIE revealed scores and abrasions on the wrapper, consistent with contact between wrapper and guide tube. Irradiation induced distortion of these particular components was not expected to differ from other wrapper/guide tube combinations. The difference, unique to this rod and guide tube, was that the adjacent subassembly wrapper was made of En58b which was subsequently known to exhibit breakaway swelling at dose rates above 50 displacements per atom, under the PFR conditions then obtaining. This caused substantial bow to develop more quickly than anticipated, sufficient to deflect the guide tube associated with that shutdown rod. Use of this wrapper material was discontinued and temporary operational procedures, earlier rotation and removal, were introduced to counter this bowing problem. Thus the effects of irradiation induced bowing have generally been found to be predictable and manageable.

Several absorbers displayed anomalous weights on a few occasions during the early life of PFR. In each case, brief reversal of the rod movement followed by renewed forward motion caused the anomaly to disappear. It is presumed to have been caused by “ledging”, interference between rod and guide tube. As mentioned above the continued evolution of the absorber rod/guide tube designs gave increased clearances and improved profiles which have eliminated the problem and it did not occur after introduction of the Mark III and IV absorber rods.

There was also an incident at low power where the operator called for the control rods to raise and when he stopped the operation they continued to rise. Design improvements to the control system were required to prevent the problem (Ref.21490.pdf)

As well as the operational issues which are covered above there were a number of issues related to safety and the evolving safety requirements during PFR operating life. Shortly after the PFR freeze date the basic approach to reactor safety became more numerically quantified. The Farmer Criterion was applied when power was first being raised, and Probability Risk Analysis was used from about 1990. The PFR Project tried, with considerable success to apply these new methodologies to its safety assessment.

Whilst the operating experience endorsed the reliability of the absorber system the requirement to demonstrate that it meets the demanded reliability of 10^{-6} per demand was onerous. The most potentially hazardous situation that can occur during operation is considered to be the result of a control rod runaway and safeguards were incorporated into the control and automatic protective systems to cope with the event. The relatively limited diversity between the control and shutdown rods design was questioned with the concern being common mode failure and various solutions investigated. This was examined by a working group which concluded that the target could be achieved but depended on the rod exercising programme and the age distribution of the rods which has already been mentioned above.

The provision of a completely separate Alternative Shutdown Device (ASD) was investigated under the ALARA objective but no design was found which would have made a significant contribution to overall safety and which was also free from considerable operational disadvantage. The possibility of an Ultimate Shutdown Device, which could be invoked if the normal shutdown system had failed, perhaps as a consequence of an incident, was also considered. Again no scheme was produced which, when examined in detail, offered significant safety improvement and all would have been operationally costly. These are important considerations in future reactors at the design stage.

A secondary shutdown rod was originally provided, to be inserted in the core centre position, where bowing is a minimum. It was to replace a temporarily inserted thimble which was used for the first critical approach and the reactor physics commissioning measurements. However, the singularity of the rod and its lack of diversity, in comparison and style, minimised its value in a probabilistic risk assessment and it was removed at reload 4 and was not replaced. (There are a number of references related to the centre shutdown rod 21491.pdf, 21492.pdf)

In summary during the twenty years of operation every rod has dropped on every occasion that the reactor was tripped and only one rod exhibited a significant anomalous drop. On this occasion it was probable that most of the reactivity was inserted quickly. The anomalies referred to have proved operationally distracting but did not compromise safety. This good performance has to be judged against a requirement that to shut down the reactor only three of the rods need to be inserted from the ten available.

A list of references relevant to the Absorber Rod Mechanisms is given in Table 8.

3.2. Failed-fuel detection system

Plant description

Fission products in particular the precursors of delayed neutron emitters, recoil from exposed fuel into the coolant. This effect has been employed in the Burst Pin Detection BPD system to detect the presence of exposed fuel and forms part of the reactor protection system. An earlier indicator of a pin failure is the release of fission product gases so it was decided in the early stages of operation of PFR to provide the facility to monitor the release by the installation of Beta precipitators in the argon gas blanket and the BPD systems, and a gamma spectrometer into the gas blanket system. The principles of failed fuel detection and methods of monitoring are presented by Diggle and Cartwright (Ref. 03204.pdf, 23662.pdf).

Two BPD sampling systems were installed in separate thimbles mounted in the reactor roof; one took coolant samples from the 6 IHX inlet trays (Figure 11) and the other directly from the core subassembly outlets (Figure 12). The system taking samples from the IHXs was known as the Bulk system and that taking samples from the subassemblies the Bulk and Location the later taking its name because it had the facility to locate the subassembly with the failed pin. The thimbles housed electro-magnetic EM pumps in a gas atmosphere (argon) and were positioned below the sodium level so that they were automatically primed.

The samples to the Bulk system remained in separate pipes through a multi-channel 6000 amp EM pump with a flowmeter in each pipe to provide confirmation that all IHXs were being sampled. The samples came together in the monitoring chamber above the reactor roof.

The Bulk and Location system consisted of 186 pipes mostly taking samples from the outlet of the subassemblies. The pipes were routed down the absorber rod guide tubes through the Fuel Element Carriers to the cold pool below the diagrid. From there they were routed to the BPD thimble. On arrival at the thimble they entered the selector valve where all except one sample was combined and ducted to a 10,000 Amp EM pump.

There was a separate location loop in a rotating shaft which cycled round the selector valve taking each of the 186 samples in turn. It was designed so that there was a continuous flow in all pipes and no degradation in the transit time. The pump in the location loop was above the sodium level and required the Bulk pump to be operating to prime. The location sample also passed the Bulk sample monitoring chamber so ensuring that all 186 (not 186 less one) samples were monitored continuously.

Some of the 186 location channels were available for additional sampling of experimental subassemblies, held in DMSAs.

The decision by the operators to include monitoring of the gas leaker phase required the thimble to be removed from the reactor using the bagging technique for sodium contaminated but not active components. There were subsequently further evolutions of the Location loop which are described below as part of the operation feedback.

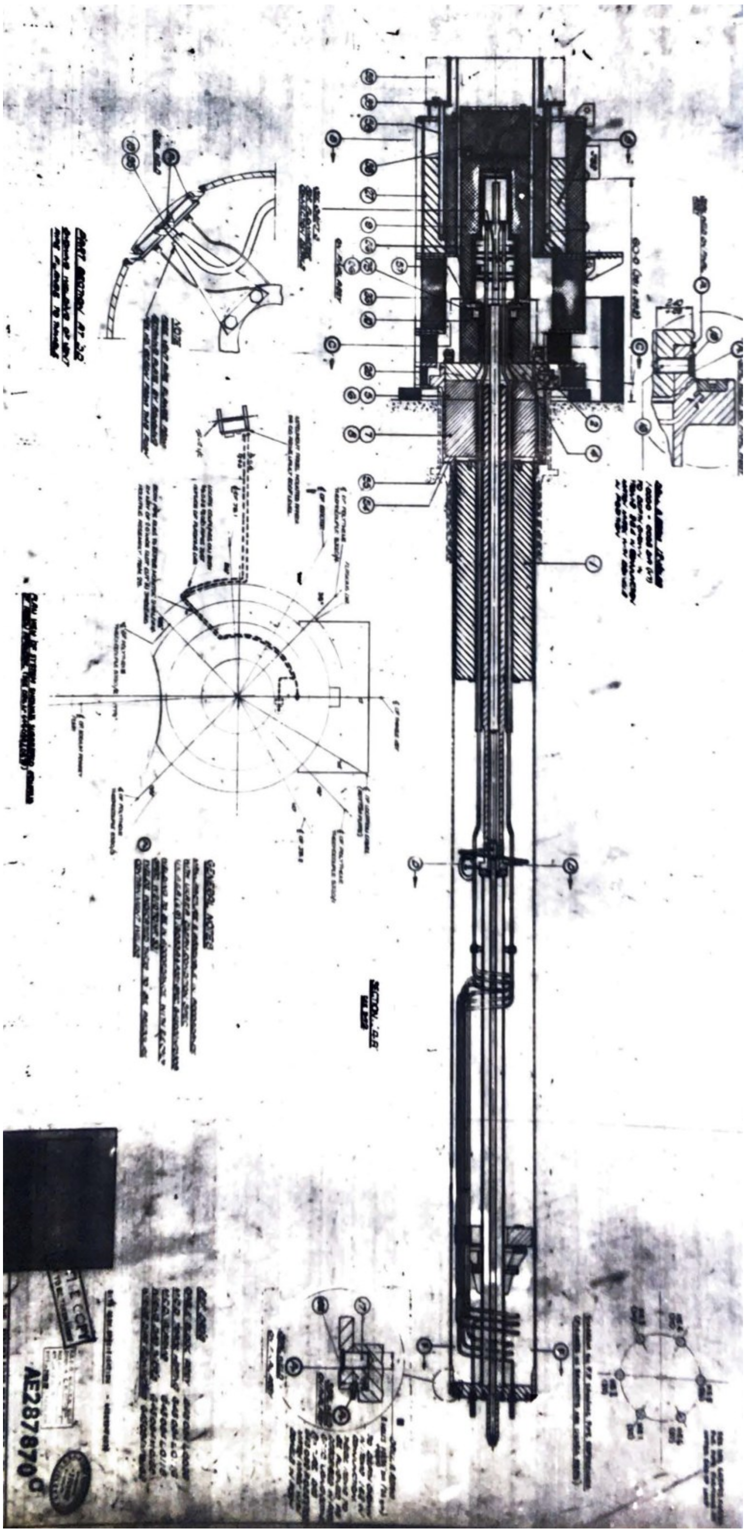

Figure 11: Failed Fuel Detections: BPD "Bulk" Sampling

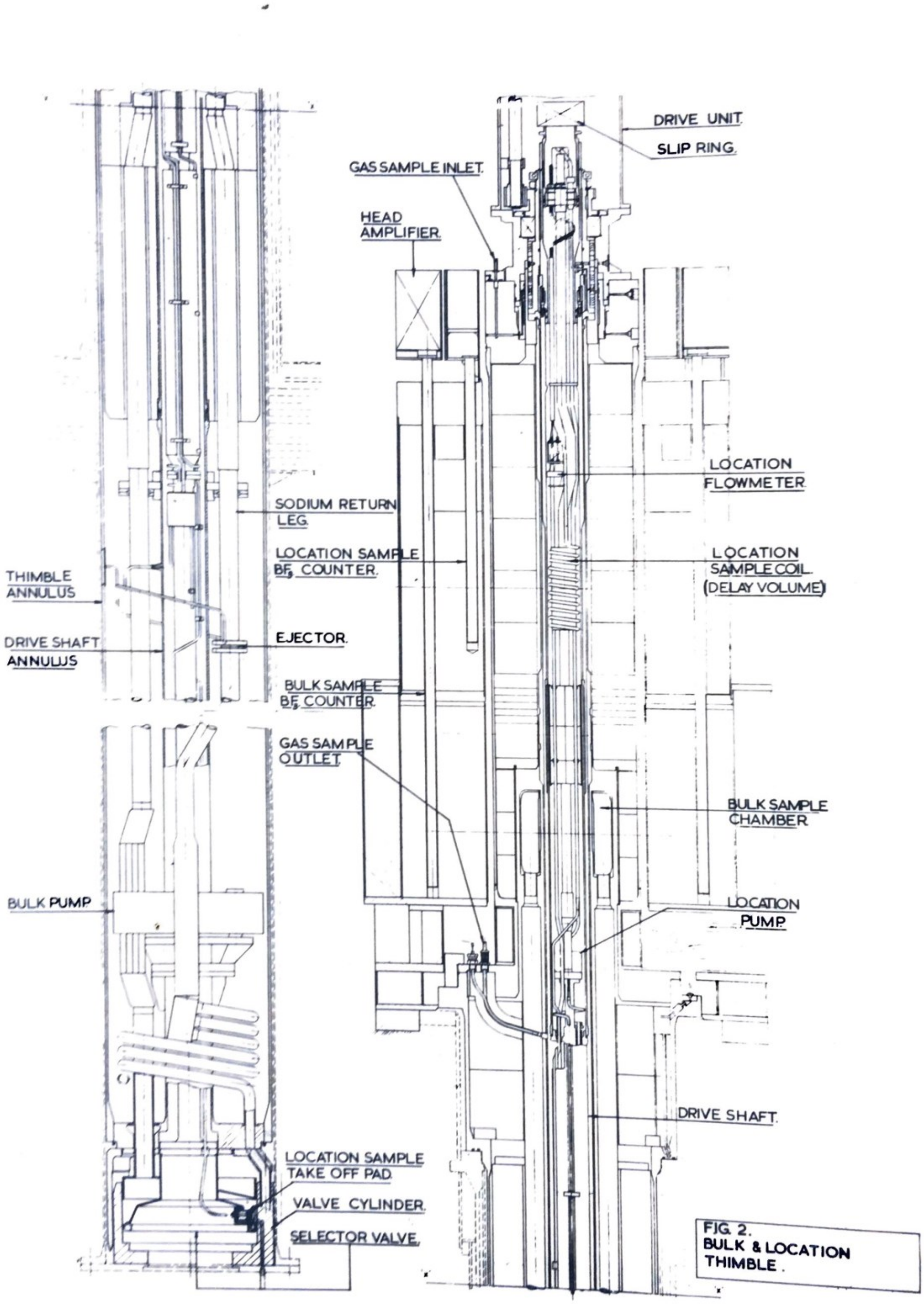

Figure 12: Failed Fuel Detection: Bulk and Location Sampling

Operation feedback

The reactor delayed neutron monitor (DNM) trip was set at a level which was equivalent to the exposure of a small fraction of the fuel in one fuel pin. It provided operational protection against contamination of the primary circuit by fissile material; it was more stringent than would have been needed for safety protection against the break-up of fuel pins. An alarm signal warned the Operator when the DNM signal reached an action level: this was at a lower level than the trip.

A major advantage of DNM is that the precursors have a half-life of seconds and, provided no fuel is released into the coolant, there is no build-up of background. Having removed a subassembly containing exposed fuel the primary coolant delayed neutron level returns to a very low background determined by surface contamination of cladding with traces of uranium from the fuel fabrication plant and any temporary inclusion of uranium foils for calibrating the detection system.

As has already been mentioned during the early stages of PFR operation, the operators, while accepting that fine cracks are innocuous, nevertheless wished to be able to monitor the development of a failed fuel pin while it remained in the Gas Leaker phase. This required removal of the Bulk and Location unit from the reactor for modification to the Location loop for gas stripping and monitoring by a precipitator. When reinstalled in the reactor the Bulk loop failed to start and the cause of the failure was not resolved. As a consequence the Location loop could not be primed because this required the Bulk loop to be operating. A temporary Location Loop was then installed using a gas lift pump which served also as the “stripper” for the gas sample to the Beta precipitator. The temporary loop was used to analyse the first fuel failures (Ref.1).

With hindsight the interaction between the Bulk and Location loops was an unnecessary complication and it would have been better had the location loop operation been independent of the Bulk loop.

Additional instrumentation was also provided in the argon gas blanket for bulk monitoring by Beta precipitators and a gamma spectrometer. Two Beta-precipitators were installed; the operational directives required that one was on-line; the other was on stand-by. It was anticipated that fuel failures would first be detected on the Beta precipitator, intergranular cracking permitting gaseous diffusion of the noble gases, whereas the delayed neutron detection system responded to a signal only when the crack developed to the point where exposed fuel was in direct contact with the coolant.

The original intention was that both Bulk delayed neutron detection systems would be part of the reactor automatic protection system but the IHX system worked consistently well such that it was possible to rely on that system. The burst pin detection equipment was calibrated using foils (Ref.03620.pdf). An RNL thermal hydraulic scale model of the hot pool using water as a simulant fluid was also used to demonstrate the adequacy of the IHX sampling (Ref.12517.pdf, 29868.pdf, 29869.pdf). This was from the point of view of the mixing behaviour so that all subassembly outlets were represented giving protection to the whole of the core and breeder and transit times were short enough for the delayed neutron sample.

All the failures identified have provided large signals in the gas blanket monitor ranging from 0.15 GBq/m^2 to 400 GBq/m^2 (normal background was 7 MBq/m^2). Absorber rod exercising, small pre-planned movement of control rods to check freedom from stiction excited an enhanced fluctuating signal if a failed fuel pin was present in the core or breeder.

The measured delayed neutron signals were related to the area of the physically observed failure using a parameter "Equivalent Recoil Area". ERA, an internationally agreed term defined as that area of smooth surface of fissile material from which fission products are emitted by recoil to procure the observed signal. Using a calibration source the reactor signals can be normalised and transformed into ERA. PIE measurements of exposed area are commonly 50-100 times less than the ERA; mechanisms other than recoil must predominate but the ERA parameter has proved a useful gauge.

The signals from the cover gas, location loop and bulk DN monitor for the first two fuel failures were analysed in 1979 (Ref. 03783, 03223pdf, and Ref.1). Although there were some reservations in the interpretation of the signals for various reasons, including uncertainties in the transit time, which would be resolved by further water model test, the failures followed the anticipated pattern of development from gas leaker to DN emitter. The Gas Leaker stage persisted for a long time (days) and then developed further to expose fuel, and at this point delayed neutron signals were detected. It was also found that the location selector valve setting was four ports out of position. Post irradiation examination of failed fuel pins confirmed the normal failure mode as being a longitudinal slit.

Throughout the life of PFR, apart from two cases where prompt action was taken for operational reasons, each subassembly with a failed pin was removed at the next scheduled shutdown with delayed neutron signals usually much less than half the operator action level. In two of the cases where no exposed fuel was detected the "leaker" was left in the reactor for 80 and 145 equivalent full power days respectively.

The reactor background level reached equilibrium for each of the detection systems. The level was perturbed by the inclusion of detector calibration foils and by the very small amount (less than 1 g) of fuel lost from all failures. The benign mode of failure gave confidence that the operators could monitor suspect situations satisfactorily and that the reactor protection system was sound.

A consequence of the policy of operating the reactor with failed fuel present was that some fission products were released into the reactor instead of into the reprocessing plant. Penetrations in the reactor roof were sealed with a double O-ring and the interspace maintained at a small excess argon gas pressure. The consequential small in-leakage built up cover gas pressure and a controlled vent was made to atmosphere via a delay tank and discharge stack. In the case of some penetrations, leakage passed the seal admitted some cover gas directly into the reactor hall from where it was ultimately discharged via the ventilation system into the discharge stack. The total discharge to the atmosphere was small in radiological terms: the activity was also well below authorised discharge limits even when the reactor was operating for substantial periods with failed fuel. Caesium, long lived sodium 22, and corrosion products were retained in the coolant.

Active isotopes plated out in the IHX were believed to be from caesium and were assessed using thermos-luminescence dosimetry. The deposition may have been associated with carbonaceous deposits resulting from the small oil spillage into the reactor which occurred early in the reactor life.

The PFR experience contributed to the understanding of the magnitude and mechanism of DN emission from fuel endurance failures (Ref.26175.pdf)

Lastly, it is interesting to note that all failures leading to exposed fuel were in experimental designs. The design of the standard fuel pin which was used in most of the core, is extremely robust even up to 10% burn-up

NNL Corporate memory references:

Ref.1 "An Assessment of BPD Signals from the First Fuel Failures in PFR" by D.K. Cartwright FRDC/FEWP/P(79)38 held in NNL corporate memory as 150428.pdf).

A list of references relevant to the Failed Fuel Detection is given in Table 10.

4. Auxiliary Equipment

4.1. Decay heat rejection loops

Decay Heat Removal

The normal route for the removal of decay heat was via the secondary sodium circuits and the steam plant. If this was not available decay heat could be disposed of by one or more of three thermal syphon loops filled with eutectic sodium-potassium alloy (NaK, melting point -12.6℃). Each loop consisted of a heat exchanger coil immersed in the primary sodium adjacent to the inlet to an intermediate heat exchanger, and an air heat exchanger (AHX) on the roof of the reactor containment building. The connecting pipework had no valves or pumps so that the NaK circulated by natural convection. In the event of complete loss of electric supplies each loop was capable of removing 1.5 MW of decay heat from the reactor. Each AHX was equipped with 2 fans connected to emergency diesel-driven guaranteed-interruptible power supplies, which could enhance the decay heat removal to over 4 MW per loop. When the reactor was operating normally heat removal was limited by dampers which restricted the airflow to the AHXs. The probability that the entire decay heat rejection system would fail was evaluated and shown to be acceptably small (1). Fluctuating thermal stresses at the point where the cold NaK downcomer entered the hot primary sodium pool were found to be acceptable.

Each AHX consisted of forty serpentine parallel tubes welded to pulled tees in two headers. The tubes were finned along the horizontal straight lengths and plain at the bends where they were clamped together and supported. Further rigidity was provided by cleats welded to the tops of the fins on adjacent tubes. The intention was that the flow of NaK should be from the top down but in practice it was found that it would sometimes flow round the loop in the reverse direction. This made little difference to the performance of the loop.

In March 1975 a leak occurred in a one of the welded pulled-tee connections between a cooling tubes and a header in Loop A. The affected tee was cut out and examined (2). The failure was attributed to a cold tear which developed during manufacture, and a repair was effected by fitting a replacement tee. Similar leaks occurred in 1981 and 1982 (2) and a common-mode failure was suspected. As an interim measure operational constraints were imposed to minimise the risk of further leaks.

Meanwhile action was taken to find the cause of the failures. Strain gauges and thermocouples fitted to one of the loops revealed considerable temperature differences between the tubes and large strains (3). Many of the cleats joining adjacent tubes were found to have broken (4). Flow on the air side of the AHXs was measured and significant irregularities were found (5). The overall performance of the loops was measured (6). These measurements and laboratory simulation (7) showed that the most important cause of the temperature differences was irregular NaK flow caused by gas locking of some of the tubes. When an AHX was filled, gas locks would occur in the horizontal tubes. A gas-locked tube remained cold and oxide impurities were precipitated causing permanent blockage. Because of the temperature difference 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 subject to low-frequency high-strain fatigue as a consequence of normal plant manoeuvring so that it cracked and eventually leaked. Vacuum filling of the heat exchangers and batch cold-trapping of the coolant improved but did not eliminate the problem (8).

In 1985 replacement AHXs 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 improvements were made:

1. A two-degree 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 AHXs were fitted in 1986 and the third in 1987. Operation was trouble free until 1996, when the loops were finally emptied for decommissioning. The improvements are described by Melhuish and Sandison (1992) and summarised by Cruickshank and Judd (2000).

Open Literature References:

Melhuish, D. B. and A. Sandison: "Engineering Improvements to PFR": *Nuclear Energy,* 1992, 31.193-205

Cruickshank, A. and A. M. Judd, “Problems Experienced During the Operation of the Prototype Fast Reactor, Dounreay, 1974-1994 – Cracks in the PFR Air Heat Exchangers”, in *IAEA-TECDOC-1180*, 2000.

5. Secondary Circuits

Steam generating units

Each of the three secondary sodium circuits had three steam-generating units (SGUs), an evaporator, a superheater and a reheater. Each SGU was a tube-in-shell heat exchanger consisting of a vessel containing sodium in which was immersed U-shaped steam tubes suspended from a thick tubeplate. During operation an argon-filled gas space kept the tube-to-tubeplate welds clear of liquid sodium. The evaporators were made of ferritic steel while the original superheaters and reheaters, which operated at higher temperatures, were austenitic. The working steam pressure was 17.5 MPa while the sodium pressure was around 0.2 MPa.

Very small leaks of steam or water into the sodium were detected by monitoring the hydrogen concentration in the argon cover-gas by means of katharometers. Larger leaks were detected by monitoring the pressure in the sodium expansion tanks.

The steam plant was filled with water and commissioned in 1973 and the secondary sodium circuits were filled in mid-1974. Within a few weeks leaks of steam or water into the sodium of superheater 3, evaporator 2 and superheater 2 were detected, and found to be due to cracks in tube-to-tubeplate welds. They were quite unexpected because all the SGUs had been helium leak-tested after manufacture. It was hoped that the leaks were the result of fabrication defects which had eluded detection. The heat transfer surface was sufficiently oversized by design to allow the affected tubes to be plugged and removed from service, and operation continued. However these leaks turned out to be the first of a series that continued until the problem was finally resolved in 1983 for the evaporators and 1987 for the superheaters and reheaters, and which severely impeded the operation of the reactor.

Open Literature References for Steam Generating Units:

IAEA “PFR Operating Experience”, pp 30-59 in *IAEA-TECDOC-1083*, Vienna 1999

IAEA “Prototype Fast Reactor”, pp 29-56 in *IAEA-TECDOC-1569*, Vienna, 2007

Cruickshank, A. and A. M. Judd, “Problems Experienced During the Operation of the Prototype Fast Reactor, Dounreay, 1974-1994 – The Under-Sodium Leak in PFR Superheater 2”, in *IAEA-TECDOC-1180*, 2000.

Judd, A. M., R. Currie, G. A. B. Linekar and J. D. C. Henderson “The under-sodium leak in the PFR superheater 2, February 1987” *Nuclear Energy* **31**, 221-230, 1992

Smith, D. C. G. *An Outline Review of PFR Development and Operation* UKAEA, 1996

5.1. Evaporators

(Numbers in brackets refer to sections of Table 12.)

The shells and tubeplates of the evaporators were made of unestablished $2^1/4$Cr 1Mo ferritic steel while niobium-stabilised $2^1/4$Cr 1Mo was used for the tubes. Each evaporator had 498 U-tubes. The tubes were attached at each end to the tubeplate by butt-fillet welds, 996 welds in all. The geometry of the welds was such that they were manufactured to a high standard automatically without crevices, but the design effectively precluded post-weld annealing so that they were hard and brittle and had high residual stresses. Nevertheless the designers believed that cracking on the water side would be prevented by the layer of magnetite that would be formed by high-temperature oxidation of the steel surface, and by operational control of the water chemistry to exclude aggressive species. On the sodium side protection was assumed to be provided by the very low probability of formation of the caustic conditions needed to initiate stress corrosion in a sodium rich environment.

Because on occasions the evaporator tube bundles were moved they were usually referred to by their manufacturing works unit (WU) number rather than the number of the secondary circuit. Initially WU1 was located in circuit 1, WU2 in circuit 2 and WU3 in circuit 3, but early in life WU2 and WU3 were interchanged so that WU2 was in circuit 3 and WU3 in circuit 2. WU1 was located in circuit 1 for the whole of the life of the plant. Having suffered a large number of leaks WU2 was removed in 1983, re-tubed with tubes made of 9Cr 1Mo steel, renamed WU2A, but never deployed. WU4 was manufactured as a spare and inserted in circuit 3 to replace WU2.

The first leak in evaporator WU3 in circuit 2 in 1974 was assumed to be due to a manufacturing defect but when further leaks appeared in 1976 generic weakness was suspected. In 1977 samples of the failed welds were removed for examination (1). This revealed cracking on the sodium side, and while the significance was not initially understood, methods to detect partially-penetrating sodium-side cracks by using non-destructive techniques from the water side were developed. Efficient techniques were developed for identifying a leaking weld among the 996 revealed when an evaporator was opened after a leak had been detected (2). Large numbers of leaks appeared again in the late 1970s, but it was not initially certain whether they grew from defects on the water or the sodium side.

Because it still seemed possible that cracks were initiated from the water side attention was paid to the integrity and general state of the magnetite layer. Visual and non-destructive inspection techniques were used to determine its thickness, the integrity within the tubes well below the tubeplate, and later a complete tube removed from WU2 was examined to determine the state of its magnetite and to check for signs of mechanical damage (3). Irregularities were caused by differences in the heat flux due to irregularities in the flow of sodium (4). Ultrasonic methods were used to measure the thickness of the tube walls to check for damage caused by fretting or galling where the tubes passed through support grids (5).

Shot peening, a technique for bombarding the metal surface with shot leaving the metal surface in a state of compressive stress to a depth of 0.5 mm, strongly reducing the propensity to cracking, was developed and after extensive trials applied to all the welds in all the evaporators (nearly 3000 in all) but on the water side only, access to the sodium side being too difficult. It was probably successful in reducing the risk of water side cracking.

However a series of leaks in 1980 and 1981, after the welds had been shot-peened, suggested that cracks were being initiated from the sodium side. It was thought that these might have been due to stress-corrosion caused by the caustic residue from previous leaks of water into the sodium. In an attempt to eliminate this possibility a regime was instituted whereby, after a leak had occurred and been repaired, the sodium level was raised to allow the underside of the tubeplate to be washed with hot sodium to remove the caustic material. The holes left by the welds cut out for metallurgical examination allowed observation of the underside of the tubeplates to determine the effectiveness of this procedure (6). A test rig was used to replicate after-leak conditions on the sodium side of the welds (7). It was eventually concluded that tubeplate washing was ineffective in removing caustic material from the tips of cracks.

Initially leaks had been repaired by inserting explosively-welded plugs into both ends of the affected tube, thus removing it from service. Later repairs were effected by bridging the leaking weld with a cylindrical sleeve of 9Cr 1Mo steel inserted into the hole in the tubeplate and extending downwards into the top 8 cm. of the tube. The lower end of the sleeve was brazed to the inside of the tube and the upper end was then explosively welded to the tube plate (8). Following extensive laboratory trials sleeves were first fitted in 1980 and several small campaigns followed as the incidence of leaks increased from 1980 - 1983. There were no leaks from sleeved tubes.

By 1981 new ultrasonic non-destructive testing techniques that had become available were used to examine all the remaining welds in all the evaporators (9). Many were found to have extensive cracking on the sodium side and in 1983 the decision was made to bridge every tube-to-tubeplate weld with a sleeve (except for those tubes already plugged. A comprehensive review was written in 1982 that gives the background to this decision (10). The background to this decision is described in detail in the ten years after the evaporators were completely fitted with sleeves, no further leaks were detected.

In 1975 a spare evaporator tube bundle was ordered, essentially identical to the original units. By the time it was required for service the sleeving process had been developed and, to avoid doubts it was sleeved prior to being installed in 1983 as a replacement for WU2. The latter had experienced a large number of leakages and on two occasions sodium had entered the water side; after the second of these occasions a number of tubes were found to be unserviceable and it was decided to re-tube the unit, using a spare tube plate and 9% Cr 1% Mo tubing. Several other modifications were made, among them the use of explosive welds. These proved only marginally acceptable following inspection and the unit, renamed WU2A, was not installed and therefore not tested.

Invariably a leak from a tube-to-tubeplate weld started very small but escalated over a period of days as the crack enlarged by erosion and corrosion. Rig tests were done to understand this process. The hydrogen detection system proved reliable in detecting leaks at this early stage (11). Only rarely did the leak become large enough to trigger the expansion tank pressure trip. The ultimate protection against over-pressurisation of the IHXs was provided by bursting discs which allowed sodium-water reaction products to be vented to atmosphere through an effluent system, at the same time triggering the closure of steam and sodium isolation valves and the evacuation of sodium to dump tanks and steam to atmosphere through the main safety valves. However the minimisation of damage to the IHXs required that this extreme level of protection would be invoked only rarely, and thus depended on a leak-before-break argument, that leaks would always start small and escalate only slowly, allowing time for detection and orderly depressurisation. Analysis of the non-destructive testing data on incipient cracks did not support the leak-before-break argument (12).

A method for local in-situ heat-treatment of welds to soften them was developed and tested on WU2 after it had been removed from service (13), but it was not used on operating evaporators which had all been sleeved.

5.2. Superheaters and reheaters

(Numbers in brackets refer to sections of Table 12)

Because the superheater and reheaters were not interchanged (unlike the evaporators) it is unambiguous to identify a tube bundle by the number of the secondary circuit in which it was located.

In October 1974, during early steam commissioning, a leak was detected in superheater 3, and three months later a leak occurred in superheater 2. In both cases the fault was in a tube-to-tubeplate weld and a repair was effected by explosively plugging the faulty tube.

In the case of superheater 3 inspection revealed cracks in the tube plate which were sufficiently accessible for removal by grinding and shallow enough that sufficient thickness remained in the plate for further service use. In one place a crack had passed from the tube into the tubeplate, so to ensure that cracks developing in any adjacent tubes would be harmless to the surrounding ring of tubes, they were plugged.

Inspection of superheater 2 also found cracking but in a region more difficult to access. A safety case for continued operation was made, based on passivation by sodium washing, a leak-before-break safety argument and periodic ultrasonic inspection. No significant crack growth was observed during the rest of the unit's life.

Early in 1976 reheater 3 failed, and inspection disclosed that a significant amount of sodium had entered the tubes. The tubeplate was found to be damaged beyond repair so the tube bundle was removed and replaced by a sodium-side flow restrictor which enabled circuit 3 to be used, although there was a small reduction in the turbine power.

These original tube bundles were fabricated in austenitic stainless steel to meet high temperature steam conditions. This steel is prone to chloride and caustic stress-corrosion cracking. This was compounded by the fact that, like the evaporators, the tubeplate was not stress relieved after welding, the designers having accepted the argument that the martensitic state of the weld posed less of a threat than the potentially distorting effect of heat treatment on the tube plate. However when a leak occurred there was a possibility of secondary cracking of the tubeplate, and moreover the cracks could propagate rapidly.

This unfavourable experience early in the operating life of the plant led to a decision to order a complete set of spare superheater and reheater tube bundles made of 9Cr 1Mo ferritic steel. To make them fit the original vessels had to be extended by spool pieces. The spares, known as replacement tube bundles or RTBs, were delivered to site in 1984. The failed re-heater was replaced and the remaining RTBs were stored.

The three original superheaters and two of the three reheaters continued in service without leaks from 1975 until 1986 when a very small leak occurred in superheater 3. The failed unit was replaced by one of the RTBs, partly to gain operational experience of the change-over.

Four months later in early 1987, while the plant was operating at full power, a leak occurred in superheater 2. Unlike all the previous leaks it was located below the sodium level and not at the tube-to-tubeplate weld. It escalated rapidly and within a few seconds a bursting disc was ruptured. The plant protection system operated as intended. The event is summarised by Cruickshank and Judd (2000) and details are given by Judd *et al.* (1992).

Post-incident examination showed that the initial failure was the result of a steam tube vibrating against the peripheral baffle causing fretting wear and, ultimately, a leak. The local chemical reaction between sodium and water caused the adjacent tubes to be overheated, thinned and corroded adjacent until they failed

Further examination revealed more fretting damage elsewhere in the same tube bundle and also in the tube bundle previously removed from superheater 3, and the possibility of a generic weakness had to be considered. It was decided to exchange all the remaining austenitic superheater and reheater bundles by the RTBs, the design of which, unlike that of the original units, had been supported by full-scale rig experiments to demonstrate the absence of damaging vibration.

The post-incident investigation revealed that more tubes had failed than had hitherto been thought possible. The unexpected severity of the failure led to a detailed re-appraisal of the safety case, because near-simultaneous failure of several tubes could possibly cause a pressure surge that would damage the intermediate heat exchangers. A theoretical model was set up which indicated that the probability of large numbers of tubes failing on the timescale of a second or so was very low (14). The model was subsequently improved by tests carried out in the Super Noah test rig. Reassessment of the strength of the secondary circuit components and pipework indicated much better ability to withstand over-pressurisation than had been thought at the design stage. These two factors led to the conclusion that there was no possibility of rupturing the intermediate heat exchangers thereby compromising the reactor safety. Given the high probability that the initiating leak grows over a period of seconds, the under-sodium hydrogen detection system offers the possibility of isolating the affected unit and dumping the steam before extensive escalation takes place.

5.3. Secondary cold trap

The purpose of the cold traps was to control concentrations of dissolved oxygen and hydrogen in the sodium. There were two cold traps, one for the primary circuit and one for the three secondary circuits. In each, the sodium was cooled to a temperature at which the oxide and/or hydride was precipitated and trapped in a wire mesh basket.

The Secondary Cold Trap (SCT) was required primarily to remove the hydrogen produced in the steam generators by corrosion on the water side of the evaporator tubes, which diffused through the wall of the steel tubes into the secondary sodium. The hydrogen concentration had to be controlled in order to maintain the sensitivity of the hydrogen detection system which was the primary means of detecting small leaks in the secondary heat exchangers. The SCT was first used in 1973 to clean up the sodium from the three secondary circuits after initial filling. When PFR was at full power the SCT was in almost continuous use, being switched from one secondary circuit to another.

The SCT formed part of an auxiliary pumped sodium loop that could be made, by means of permanent pipework and valves, to communicate with each of the secondary circuits in turn. During commissioning, prior to filling with sodium, oil and tarry residues were found in the small pipes and valves of the loop. These had come from oily residue left on the wire mesh after manufacture. The mesh was in the form six of toroidal “doughnuts” which had to be cleaned with acetone prior to installation.

After commissioning when the SCT was in use a gradual reduction in the sodium flow as the mesh became blocked by impurities indicated that the maximum loading had been reached. The cold trap basket was then removed, cleaned, fitted with new set of doughnut meshes and reused. It was expected that loadings in excess of 100 kg of mixed sodium hydride and oxide would be achieved but in practice they were much smaller. The sixth basket was installed in July 1980 and was 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 behaviour it was decided to remove it for examination to ascertain the reasons for early blockage. Results of the examination are summarised by Cruickshank and Judd (2000).

During the period of repeated evaporator leaks the SCT was heavily loaded and basket changes were frequent, and there was often difficulty in removing the basket from the trap. This was because as well as oxide and hydride the deposits contained hydroxide which tended to weld the basket into the vessel. These deposits could not be melted below about 400 °C, a temperature that could not readily be archived in situ. The solution was to provide a complete spare cold trap vessel and basket, engineered so that the entire SCT could be changed on load quickly enough to avoid impurity levels rising above acceptable limits. The procedure is described in (6).

During the early years of operation the frequency of leaks in the evaporators resulted in the need to remove large quantities of oxygen and hydrogen. When secondary circuit sodium became very badly contaminated "batch" cold trapping was used, partly to avoid overloading the cold trap basket and partly to avoid blockages of the small diameter pipes in the SCT loop. Batch cold trapping was carried out by discharging the secondary circuit sodium at 300 °C to one of the dirty dump tanks where it cooled over some 10-14 days to 150 °C allowing the oxide and hydride to precipitate. The sodium was then transferred to the clean-up tank, reheated to 300 °C and returned to the circuit. After the sodium had been circulated a second treatment was sometimes necessary because the IHX section of the secondary circuit containing about 9 tonnes of sodium did not drain when sodium from the rest of the circuit was dumped.

Detailed results of the examination of basket 6, summaries of experience with baskets 1-8, modifications that improved the performance of the SCT, and descriptions of attempts to calculate the performance of the SCT, are given in the reports listed in Table 13.

Open literature reference for the SCTL:

Cruickshank, A.; and. Judd, A. M.; “Problems Experienced During the Operation of the Prototype Fast Reactor, Dounreay, 1974-1994 – Blockage of the PFR Secondary Cold Trap”, in *IAEA-TECDOC-1180*, 2000.

6. Concluding Remarks

The report is focussed on plant areas which were identified as providing the most valuable feedback for the design and operation of future fast reactors. So the plant areas which were judged to have the most valuable feedback were those where there were problems not anticipated in the design. They also reflect the changing safety and monitoring requirements, particularly the need to provide assurance of the integrity of the plant because in-service inspection and monitoring was not a high priority at the design stage. It has not been possible to cover all the areas that were identified initially so a further prioritisation was necessary. Of course there were many aspects of PFR operation which were success stories, demonstrating the soundness of the design and technology that are not covered in this report.

This report summarises the operating experience for the selected areas and provides reference documents that can be used to obtain a better understanding. As anticipated these are mainly what was identified as secondary level documents which provide a good understanding and may lead to primary sources (measurements, first-hand descriptions, etc.) if they still exist. These reference documents were selected from an extensive list of potentially relevant documents drawn from the Wood archive which have been listed for each plant area in the tables at the end of the report. The selection of the reference documents was not a rigorous process and other valuable documents will not have been identified.

The Wood archive has been relied on extensively in producing the report with only a limited effort in searching documents from other sources. As most of the documents used where from the UKAEA FR committees they will be available from other sources but at this time the Wood archive was very convenient.

The PFR experience can be helpful on several levels. Firstly on the level of the underlying technology with thermal hydraulics and structural integrity being the most obvious. The PFR safety case was updated as the issues arose and the case had to be made for continued operation against a background of changing requirements and practices. The updated sections of the safety report are useful secondary level documents some of which are included in the report but the evolution of the safety case has not been included. Then there is the value to the design of future reactors. All aspects covered in the report were reflected in the parallel development of the UK CDFR and then the European EFR design.

7. Tables of References

Table 1: Woods Report List: Core and Breeder

Reference No.	Title	Location	Authors	Year
1	**Start-up experiments**			
03904.pdf	AN EXPERIMENT TO CHECK THE REACTIVITY SCALE OF THE PROTOTYPE FAST REACTOR	TRG MEMO 7460	LENNOX TA	
04354.pdf	THE INITIAL APPROACH-TO-CRITICAL OF THE PROTOTYPE FAST REACTOR	TRG REPORT 2908	WHEELER RC;CROWE DS;HENDERSON JDC	
2	**Reactivity noise**			
03865.pdf	POWER NOISE MEASUREMENTS AT VARIOUS REACTOR POWERS AND PRIMARY SODIUM PUMP SPEEDS	PFR EXPERIMENTAL RESULTS SHEET NO.51	CROWE DS;SUTHERLAND AJ	
03876.pdf	POWER NOISE MEASUREMENTS AT 500 MW(TH)	PFR EXPERIMENTAL RESULTS SHEET NO.40	CROWE DS	
03890.pdf	MEASUREMENT OF POWER NOISE	PFR EXPERIMENTAL RESULTS SHEET NO.25	ETHERINGTON EW	
3	**Reactivity and burnup**			
03786.pdf	MEASUREMENT OF REACTIVITY BURN UP FOR RUN 2	PFR EXPERIMENTAL RESULTS SHEET NO.93	SUTHERLAND AJ	
03879.pdf	MEASUREMENT OF REACTIVITY/BURN-UP IN THE PERIOD MAY TO JULY 1976	PFR EXPERIMENTAL RESULTS SHEET NO.37	SUTHERLAND AJ	
03892.pdf	ESTIMATE OF THE BURN-UP RATE FOR PERIOD 21 NOVEMBER TO 22 DECEMBER 1975	PFR EXPERIMENTAL RESULTS SHEET NO.23	CROWE DS;SUTHERLAND AJ	
03893.pdf	MEASUREMENTS OF REACTIVITY COEFFICIENTS ASSOCIATED WITH TEMPERATURE, POWER AND BURN-UP DURING OCTOBER 1975	PFR EXPERIMENTAL RESULTS SHEET NO.22	CROWE DS;SUTHERLAND AJ	
15141.pdf	AN EXAMINATION OF THE CHANGE IN CORE REACTIVITY OF PFR DURING RUN 16C	PFR/TC/P(89)343	CROWE DS;DISBURY WH;NEWTON TD	
4	**Calibration of control rods**			

Reference No.	Title	Location	Authors	Year
03807.pdf	PFR CONTROL ROD CALIBRATION IN SOURCE REGIME (< 1KW) TO INVESTIGATE RELOAD REACTIVITY DISCREPANCY - START OF RUN 6	PFR EXPERIMENTAL RESULTS SHEET NO.117;OETD.TECH NOTE NO.503	CROWE DS;LORD DJ;SUTHERLAND AJ	1981
03808.pdf	CALIBRATION OF PFR ABSORBER RODS AT START OF RUN 6	PFR EXPERIMENTAL RESULTS SHEET NO.118;OETD.TECH NOTE NO.517	CROWE DS;SUTHERLAND AJ	1982
03862.pdf	CALIBRATION OF PFR ABSORBER RODS AT START OF RUN ONE	PFR EXPERIMENTAL RESULTS SHEET NO.54	DICKSON AK;LORD DJ;SMITH JC;SUTHERLAND AJ;WEBSTER EB; WILSON C	
03864.pdf	CALIBRATION OF PFR ABSORBER RODS AT END OF RUN TO+	PFR EXPERIMENTAL RESULTS SHEET NO.52	CROWE DS;DICKSON AK;LORD DJ;NEWTON TD;SMITH JC; SUTHERLAND AJ;WEBSTER EB	
03883.pdf	CHECK ON REPRODUCIBILITY OF WORTH OF CONTROL ROD DROPPING FROM 100 MM	PFR EXPERIMENTAL RESULTS SHEET NO.32	CROWE DS;SUTHERLAND AJ	
03885.pdf	MEASUREMENT OF THE RELATIVE WORTH OF EACH SHUT-OFF ROD AT LOW POWER	PFR EXPERIMENTAL RESULTS SHEET NO.30	CROWE DS;SUTHERLAND AJ	
03887.pdf	MEASUREMENT OF THE SHUT OFF RODS SHAPE FUNCTION	PFR EXPERIMENTAL RESULTS SHEET NO.28	CROWE DS;SUTHERLAND AJ	
03889.pdf	PRELIMINARY CONTROL ROD CALIBRATION AT LOW POWER TO MONITOR FOR LOSS OF ABSORBER	PFR EXPERIMENTAL RESULTS SHEET NO.26	SUTHERLAND AJ;CROWE DS	
03897.pdf	MEASUREMENT OF SHUT OFF ROD WORTHS WITH CONTROL RODS POSITIONED TO MINIMISE AND MAXIMISE TILT	PFR EXPERIMENTAL RESULTS SHEET NO.17	CROWE DS;SUTHERLAND AJ	

Reference No.	Title	Location	Authors	Year
15145.pdf	AN EXPERIMENTAL EVALUATION OF THE REACTIVITY ADDITION DUE TO WITHDRAWING TWO ABSORBER RODS FROM PFR WHEN IN ITS FULLY SHUTDOWN CONFIGURATION	PFR/TC/P(89)37 8	NEWTON T	1989
5	**Miscellaneous measurements**			
03813.pdf	PERFORMANCE CURVE CHECKS ON THE PFR LOW POWER CHANNELS	PFR EXPERIMENTAL RESULTS SHEET NO.123;OETD.T ECH NOTE NO.557	SUTHERLAND AJ	1982
03872.pdf	MEASUREMENT OF REACTIVITY HELD UP BY NEPTUNIUM	PFR EXPERIMENTAL RESULTS SHEET NO.44	CROWE DS;DICKSON AK;SUTHERLA ND AJ	
03878.pdf	MEASUREMENTS OF POWER NOISE AND PIN LEVITATION MADE WHEN THE P.S.P. SPEEDS AND REACTOR POWER WERE BEING RAISED (12-30 JULY 1976)	PFR EXPERIMENTAL RESULTS SHEET NO.38	CROWE DS;MCWILLIA M D;SUTHERLAN D AJ;ETHERING TON EW	
03888.pdf	THE MEASUREMENT OF THE BERYLLIUM SOURCE STRENGTH	PFR EXPERIMENTAL RESULTS SHEET NO.27	SOMERVILLE A;CROWE DS	
03891.pdf	MEASUREMENT OF PU 241 DECAY IN THE PERIOD APRIL TO OCTOBER 1975	PFR EXPERIMENTAL RESULTS SHEET NO.24	SUTHERLAND AJ	
03896.pdf	INVESTIGATION OF THE DISCREPANCY IN ESTIMATING REACTIVITY CHANGES USING THE DC14 (FRTG) CHAMBER	PFR EXPERIMENTAL RESULTS SHEET NO.18	CROWE DS;SUTHERLA ND AJ	
02608.pdf	THE PFR REACTIVITY DISCREPANCY AFTER RELOAD 5	PPWP/P(81)329	LORD DJ	
6	**Subcritical reactivity measurement**			
03459.pdf	SUB-CRITICAL MONITORING FOR RELOAD 5 OF PFR	FRMCSG/P(82)2 66	MOFFATT HS	1982
03791.pdf	SUB CRITICAL MONITORING DURING RELOAD 3 IN PFR	PFR EXPERIMENTAL RESULTS SHEET NO.97A;OETD.T ECH NOTE NO.46	CROWE DS	

Reference No.	Title	Location	Authors	Year
03814.pdf	SUB CRITICAL MONITORING DURING RELOAD 5 OF PFR	PFR EXPERIMENTAL RESULTS SHEET NO.124;OETD.TECH NOTE NO.565	CROWE DS	1982
03819.pdf	SUB-CRITICAL MONITORING DURING RELOAD 6 IN PFR	PFR EXPERIMENTAL RESULTS SHEET NO.129;OETD TECH NOTE NO.653	CROWE DS	1982
04424.pdf	SUB CRITICAL MONITORING DURNG RELOADS 7, 8A AND 8B OF PFR	PFR/OC/P(85)237;ND-M-2910	CROWE DS;SUTHERLAND AJ	1985
07737.pdf	SUBCRITICAL MONITORING STUDIES FOR RELOAD 10A OF PFR	PFR/ERS/156;OETD/TN/1290;FRCMWG/P(86)294;PFR/TC/P(86)32	CROAD DS;LORD DJ;TAYLOR JA	
10326.pdf	SUBCRITICAL MONITORING STUDIES FOR RELOAD 13A OF PFR	PFR/ERS165;OETD/TN1468;PFR/TC/P(87)177	CROWE DS;LORD DJ;DISBURY W	1987
10327.pdf	SUBCRITICAL MONITORING STUDIES FOR RELOAD 12 OF PFR	PFR/ERS164;OETD/TN1461;PFR/TC/P(87)175	CROWE DS;LORD DJ;DISBURY W	1987
10328.pdf	SUBCRITICAL MONITORING STUDIES FOR RELOAD 11 OF PFR	PFR/ERS163;OETD/TN1460;PFR/TC/P(87)173	CROWE DS;LORD DJ;DISBURY W	1987
11421.pdf	SUBCRITICAL REACTIVITY MONITORING ON PFR - A STATUS REPORT AND REVIEW (FEB 1988)	PFR/TC/P(88)242	LORD DJ;CROWE DS	1988
17109.pdf	SUB CRITICAL MONITORING FOR RELOAD 20 OF PFR	PFR EXP RESULTS SHEET NO 192;ASMD TECH NOTE 1810	CROWE DS	1990
20544.pdf	SUB CRITICAL MONITORING FOR RELOAD 24 OF PFR	PE1/2407;PFR/ER/202	CROWE DS	1991
7	**Reactivity coefficients**			

Reference No.	Title	Location	Authors	Year
03812.pdf	REACTIVITY EFFECTS OF FLOW CHANGES AT THE END OF RUN 6 OF PFR	PFR EXPERIMENTAL RESULTS SHEET NO.122;OETD.TECH NOTE NO.553	CROWE DS;LORD DJ;SUTHERLAND AJ	1982
03816.pdf	REACTIVITY EFFECTS OF FLOW CHANGES AT START OF RUN 7 OF PFR	PFR EXPERIMENTAL RESULTS SHEET NO.126	LORD DJ;SUTHERLAND AJ;CROWE DS	1982
03825.pdf	REACTIVITY EFFECTS OF FLOW CHANGES AT THE END OF RUN 7 OF PFR	PFR EXPERIMENTAL RESULTS SHEET NO.135;OETD TECH NOTE NO.782	CROWE DS;LORD DJ;SUTHERLAND AJ	1983
03831.pdf	REACTIVITY EFFECTS OF FLOW CHANGES AT START OF RUN 8 OF PFR	PFR EXPERIMENTAL RESULTS SHEET NO.141;OETD TECH NOTE NO.832	CROWE DS;SUTHERLAND AJ;LORD DJ	1984
03843.pdf	REACTIVITY FEEDBACK EXPERIMENTS AT THE END OF RUN 1 (FEBRUARY 1978)	PFR EXPERIMENTAL RESULTS SHEET NO.73	LORD DJ;DICKSON AK	
03852.pdf	MEASUREMENT OF POWER COEFFICIENTS AND REACTIVITY BURN-UP IN PFR FOR AUGUST TO OCTOBER 1977	PFR EXPERIMENTAL RESULTS SHEET NO.64	SUTHERLAND AJ	
03859.pdf	MEASUREMENT OF POWER COEFFICIENTS AND REACTIVITY BURN-UP IN PFR FOR JUNE/JULY 1977	PFR EXPERIMENTAL RESULTS SHEET NO.57	SUTHERLAND AJ	
03868.pdf	MEASUREMENTS OF THE POWER COEFFICIENT BETWEEN AUGUST AND NOVEMBER 1976	PFR EXPERIMENTAL RESULTS SHEET NO.48	CROWE DS;DICKSON AK;SUTHERLAND AJ	
03869.pdf	ON THE DIFFICULTIES OF MEASURING THE INLET TEMPERATURE COEFFICIENT AT POWER	PFR EXPERIMENTAL RESULTS SHEET NO.47	EDGE DM	
03881.pdf	ISOTHERMAL TEMPERATURE COEFFICIENT (18/21 JUNE 1976)	PFR EXPERIMENTAL RESULTS SHEET NO.34	CROWE DS;SUTHERLAND AJ	

Reference No.	Title	Location	Authors	Year
03884.pdf	ISOTHERMAL TEMPERATURE COEFFICIENT (3/4 MAY 1976)	PFR EXPERIMENTAL RESULTS SHEET NO.31	CROWE DS;SUTHERLAND AJ	
03893.pdf	MEASUREMENTS OF REACTIVITY COEFFICIENTS ASSOCIATED WITH TEMPERATURE, POWER AND BURN-UP DURING OCTOBER 1975	PFR EXPERIMENTAL RESULTS SHEET NO.22	CROWE DS;SUTHERLAND AJ	
03895.pdf	REACTOR TEMPERATURE COEFFICIENT	PFR EXPERIMENTAL RESULTS SHEET NO.19	SOMERVILLE AC	
03907.pdf	PFR CORE PERFORMANCE - A COMPARISON BETWEEN PREDICTED AND MEASURED VALUES OF SOME IMPORTANT PARAMETERS	PPWP/P(78)189;CFR/SWP/P(78)4;TC/P(78)1;FRCMWP/P(78)207	SMITH DCG;CROWE DS;LENNOX TA;LORD DJ;SMITH JC;HAMPSHIRE R	
08911.pdf	REVIEW OF PFR REACTIVITY FEEDBACK COEFFICIENT MEASUREMENTS 1974 - 1987	FRDCC/PPWG/P(87)110;nd-m-3827	LORD DJ;CROWE DS;DICKSON AK;SUTHERLAND AJ	1987
10130.pdf	ADDENDUM TO REVIEW OF PFR REACTIVITY FEEBACK MEASUREMENTS 1974 - 1987	FRDCC/PPWG/P(87)110;FRDCC/PPWG/P(87)10	DICKSON AK	
8	**Fast component of feedback**			
22408.pdf	PHYSICS EXPERIMENTS TO INCREASE UNDERSTANDING OF PFR REACTIVITY FEEDBACK MECHANISMS	FRDCC/P(88)285;PFR/SWP/P(88)23;DPC/P(88)12	LORD DJ;WILKES DJ	
18555.pdf	REACTIVITY FAST FEEDBACK MEASUREMENT ON PFR FEBRUARY 1991	206/PEI;PFR EXP RESULTS SHEET 197 ASMD TECH NOTE 1856	CROWE DS	1991
19943.pdf	THE MEASUREMENT OF FAST ACTING REACTIVITY FEEDBACK IN PFR AND A COMPARISON OF MEASURED VALUES WITH THOSE PREDICTED FROM A MODEL USED FOR SAFETY CASE STUDIES OF PFR	PE1/1808;PFR/TC/P(91)482	NEWTON TD	1991
16988.pdf	PFR REACTIVITY FEEDBACK EXPERIENCE AND ITS RELATIONSHIP TO GAS GAP CONDUCTANCE VALUES	CFR/THWG/P(79)173	LORD DJ	
9	**Oscillator**			
03634.pdf	PHYSICS ASPECTS OF THE PFR OSCILLATOR	TC/P(72)15	TAIT D	
21318.pdf	PFR SAFETY REPORT OSCILLATOR MECHANISM	PFR/SWP/P(74)47	TAIT D	

Reference No.	Title	Location	Authors	Year
10	**Reactor physics summaries**			
03898.pdf	REACTOR PHYSICS EXPERIENCE ON THE PROTOTYPE FAST REACTOR	IAEA-SM-244/44	LORD DJ;WEBSTER EB	
03905.pdf	REACTOR PHYSICS INFORMAION FROM THE PFR DURING 1975 AND 1976	PPWP/P(77)137;CFR/SWP/P(77)2;TC/P(77)5	SMITH DC G;WHEELER RC	
03907.pdf	PFR CORE PERFORMANCE - A COMPARISON BETWEEN PREDICTED AND MEASURED VALUES OF SOME IMPORTANT PARAMETERS	PPWP/P(78)189;CFR/SWP/P(78)4;TC/P(78)1;FRCMWP/P(78)207	SMITH DCG;CROWE DS;LENNOX TA;LORD DJ;SMITH JC;HAMPSHIR E R	
03908.pdf	REACTOR PHYSICS INFORMATION FROM THE PFR DURING 1979	PPWP/P(79)269	EDMISTON G;DAWSON C;LORD D;WEBSTER R;SMITH J	1980
03909.pdf	REACTOR PHYSICS INFORMATION FROM THE PFR DURING 1975 AND 1976	TRG MEMO 7522	SMITH DCG;WHEELE R RG	
24491.pdf	THE REACTIVITY HISTORY OF PFR DURING PERIOD 1975 TO 1984 (RUNS 0 TO 8 INCLUSIVE)	ND-M-2913	LORD DJ	1985

Table 2: Wood Report List: Core Support Structure

Reference No.	Title	Location	Authors	Year
00226.pdf	PFR DESIGN REVIEW - ABSORBER RODS, GUIDE TUBES AND 4" BREEDER REFLECTORS	FEWP/P(84)12	BROWNE JJ	
00229.pdf	PFR ABSORBER AND GUIDE TUBE DISCHARGE STRATEGY	FROC/P(84)99; FEWP/P(84)15	DODD CL	
01179.pdf	PEBBLE - A COMPUTER CODE TO STUDY THE ABSORBER ROD INTERACTIONS WITH GUIDE TUBES - DESCRIPTIVE NOTE	FEWP/P(81)47; TN/P(81)188	RIDING DJ	1981
01798.pdf	PFR GUIDE TUBES AND ABSORBERS: ORDERING STRATEGY	FROC/P(84)135	STILLWELL JC	
01798.pdf	PFR GUIDE TUBES AND ABSORBERS: ORDERING STRATEGY	FROC/P(84)135	STILLWELL JC	
01800.pdf	PFR BREEDER REFLECTORS - DESIGN AND SUPPLY	FROC/P(84)137	DODD CL	1984
01807.pdf	PROPOSED MODIFICATIONS TO THE PFR MKIV CONTROL RODS AND GUIDE TUBES	FROC/P(84)124	BROWNE JJ;FORD J	1984
01808.pdf	PROPOSED MODIFICATIONS TO FLOWMETER LOCATION TUBE SPIKES	PFRSWP/ESC/P(84)13;FROC/P(84)123	RIDING DJ	1984
01817.pdf	A FORECAST OF REQUIREMENTS FOR PFR GUIDE TUBES AND CONTENTS	FROC/P(84)113 ;PFR/FS/P(84)1	BROWNE JJ	
01818.pdf	POLICY FOR PFR CORE COMPONENT MEASUREMENT AND ROTATION	FROC/P(84)112	DODD CL	1984
01820.pdf	CHANGING THE CENTRAL GUIDE TUBE PFR, DOUNREAY	FROC/P(84)109	WEBB J	1984
01830.pdf	PFR DESIGN REVIEW - ABSORBER RODS, GUIDE TUBES AND 4" BREEDER REFLECTORS	FEWP/P(84)12; FROC/P(84)102	BROWNE JJ	1984
01832.pdf	PFR ABSORBER AND GUIDE TUBE DISCHARGE STRATEGY	FROC/P(84)99	DODD CL	
01847.pdf	BRITTLE LATCH GUIDE TUBES - A NOTE FOR THE RECORD	FROC/P(82)14	WEBB J	1982
03491.pdf	LENGTH CHANGES IN PFR GUIDE TUBES	DFMC/P(83)22; FEWP/P(83)41	WASHINGTON ABG	1983
04672.pdf	MODIFICATIONS REQUIRED TO ABSORBER RODS AND GUIDE TUBES IN PREPARATION FOR RELOAD 7	FROC/P(83)58; PFR/SWP/ESC/P(83)20	BROWNE JJ	1983
08322.pdf	UNDERSODIUM ULTRASONIC VIEWING IN PFR. AN OUTLINE OF A METHOD FOR ASSESSING DIAGRID SETTING TILT IN - SERVICE		CROAD A	1987
08419.pdf	PFR DESIGN REVIEW - ABSORBER RODS,GUIDE TUBES AND ABSORBER DMSA EXPERIMENTS.BCD FOILS	FRDCC/FEWP/P(87)10	FORD J	1987

Reference No.	Title	Location	Authors	Year
09952.pdf	FURTHER CONSIDERATIONS ON THE BOWING OF SUB-ASSEMBLY WRAPPERS IN PFR AND CFR	FRDC/P(70)22; PFR/TC/P49;FRDC/CPWP/P(70)22	JACKSON GO	
09953.pdf	OPERATIONAL IMPLICATIONS OF STEEL SWELLING ON PFR	FRDC/P(70)32	EVANS AD	
09954.pdf	SUGGESTED REQUIREMENTS FOR A NEW PARAMETRIC OPTIMISATION PROGRAMME	FRDC/P(70)23	JACKSON GO	
09955.pdf	FURTHER CONSIDERATIONS ON THE BOWING OF SUB-ASSEMBLY WRAPPRES IN PFR AND CFR	FRDC/P(70)22; PFR/TC/P.49;FRDC/CPWP/P(70)22	JACKSON GO	
09956.pdf	THE EFFECTS OF STEEL SWELLING IN THE PFR CORE	FRDC/P(70)19; FRDC/CPWP/P(70)17	JACKSON GO	
11735.pdf	RECOMMENDED LIFE LIMITS FOR PFR ABSORBER RODS AND GUIDE TUBES BASED ON THE MATERIAL 54 SWELLING RULE	FRDCC/CFWG/P(88)7;PFR/TC/P988)234;PFR/SWP/ESC/P(88)4	LIGHTOWLERS RJ	1988
12306.pdf	PFR WATER COMMISSIONING TESTS DETERMINATION OF DUMMY CORE PAD REACTION LOADS	PFR/FDWP/P(71)135	HACKNEY S;HOLMES JAG	
12854.pdf	PAD ARRANGEMENT FOR CFR 8 SUB-ASSEMBLIES/LEANING POST CLUSTER	PFR/FEDWP/P(75)363LFEDO/DM 75/57;FRD/P(75)45	SMITH BH;LUNT AR	
12904.pdf	IRRADIATED TESTING OF CFR1 LEANING POST DESIGN FEATURES	PFR/FEDWP/P(76)409	BAGLEY KQ;STANDRING J	
12985.pdf	EVALUATION TEST PROGRAMME ON FAST REACTOR ABSORBER RODS AND GUIDE TUBES	PFR/FEDWP/P(77)563	BENTLEY JW	
13027.pdf	PFR ABSORBER RODS AND GUIDE TUBES STATEMENT FOR PFR/FEDWP	RTD/TECH NOTE(78)6	DODD JA	
13160.pdf	PRELIMINARY EXAMINATION OF THE BOWING OF THE PFR CONTROL AND SHUT OFF RODS AND THEIR ASSOCIATED GUIDE TUBES	PFR/FEDWP/P(77)529	SIMPSON A	
14297.pdf	AN INVESTIGATION INTO POSSIBLE LEANING POST BOLT YIELDING FROM THE HIGH BOW OF SUB-ASSEMBLY JRA	PFR/FEDWP/P(89)1453	NELLIGAN	1989
19947.pdf	THE PRESENT STATE OF THE SAFETY CASE FOR THE PFR CORE SUPPORT STRUCTURE	PE1/1466;PFR/TC/P(91)17	HENDERSON JDC	1991
21655.pdf	RECOMMENDED LIFE LIMITS FOR ABSORBER RODS AND GUIDE TUBES IN PFR RINGS 4 AND 10	PFR/SWP/ESC/P(86)31;PFR/TC/P(86)84	LIGHTOWLERS RJ	1986

Reference No.	Title	Location	Authors	Year
21698.pdf	MEASUREMENTS AND ESTIMATES OF SHOULDER BOW FOR FLOW METER LOCATION TUBES AND ASSOCIATED GUIDE TUBES	PFR/SWP/ESC/P(87)38;DFMC/P(87)81	LILLEY RJ	1987
21947.pdf	INTRIDUCTION OF MK IIIG 11TH AND MK IIIJ GUIDE TUBES	PFR/SWP/ESC/P(86)8	BROWNE JJ	1986
21962.pdf	RECOMMENDED LIFE LIMITS FOR PFR ABSORBER RODS AND GUIDE TUBES BASED ON THE MATERIAL 54 SWELLING RULE	PFR/SWP/ESC/P(88)4;FRDCC/CFWG/P(88)7;PFR/TC/P(88)234	LIGHTOWLERS RJ	1988
22493.pdf	THE SAFETY CASE FOR THE PFR CORE SUPPORT STRUCTURE	PFR/SWP/P(90)42	TOMKINS B	
23069.pdf	DESIGN OF THE PFR MK6 GUIDE TUBE	PFR/SWP/ESC/P(86)11	BROWNE JJ	1986
25165.pdf	EXTRACTS FROM THE PFR DESIGN SAFETY REPORT FOR THE DIAGRID,FUEL ELEMENT CARRIERS AND CORE SUPPORT STRUCTURE	PFR/LLF/P(85)6		
25218.pdf	WRAPPER DILATION AND LEANING POST BOLTS	PFR/LLF/P(85)33	DODD JA	
25236.pdf	DISTORTION OF THE PFR DIAGRID SUPPORT STRUCTURE FOLLOWING POSTULATED FAILURE OF THE SUPPORT STRAPS	SIC/220/P(88)17;PFR/LLF/P(88)79	JUDGE RCB	1989
25252.pdf	AN UPDATE OF THE INCREDIBILITY OF FAILURE ARGUMENT FRO THE PFR CORE SUPPORT STRUCTURE	PFR/SWP/P(90)49	PICKER C	1990
25362.pdf	STRUCTURAL ASSESSMENT OF THE PFR LEANING POST TO SUBASSEMBLY CARRIER BOLTS	SIC/220/P(87)1;TPSD/P(87)1538	MICHIE D	1987
25476.pdf	BOUNDARY LAYER ATTENUATION OF SODIUM TEMPERATURE FLUCTUATION NEAR THE BOTTOM OF THE CONTROL ROD GUIDE TUBES	DCWG/P(82)337;CFR/EST/P291;CFR/THWG/P(75)114	BELL RT	
25863.pdf	POBABILITY OF SHUTDOWN FAILURE DUE TO ABSORBER ROD BOWING IN THE PROTOTYPE FAST REACTOR	NJH 84/12	HOLLOWAY NJ	1984
25871.pdf	PRELIMINARY THOUGHTS ON PFR ROD FAILURES DUE TO CHANGE IN GEOMETRY	NJH 84/5	HOLLOWAY NJ	1984
26000.pdf	ABSROD A COMPUTER CODE FOR CALCULATING WRAPPER TEMPERATURES IN CONTROL RODS AND GUIDE TUBES IN PFR AND CFR DESCRIPTION AND USERS GUIDE	ND-M-1121	WILLIAMS BD;MCAREAVEY G	1980
26002.pdf	PFR CONTROL AND SHUT OFF RODS (MK3) DROP TESTS IN WATER	TRG-M-5087	BARRETT WI	
26539.pdf	PEBBLE A COMPUTER CODE TO STUDY ABSORBER ROD INTERACTION WITH GUIDE TUBES DESCRIPTION NOTE	FRDC/FEWP/P(81)47;RTD/TN(81)188	RIDING DJ	1981

Reference No.	Title	Location	Authors	Year
27347.pdf	PROPOSED DEVELOPMENT WORK ON HARDFACING AT RFL SPRINGFIELDS (FAST REACTOR FUEL ASSEMBLY AND LEANING POST COMPONENTS)	FMDC/P(75)21	ROSS E	
27809.pdf	DETERMINATION OF MAXIMUM STRESS IN NEUTRON SHIELD AREA OF DIAGRID PLATE	E/EDD/TECH NOTE 1075	LACEY DR	1981
27963.pdf	BOW LIMITS FOR PFR MK1A SHUT-OFF RODS AND GUIDE TUBES	RTD/TN(79)40	SIMMERS DA	
27970.pdf	BOW LIMITS FOR PFR MK1A SHUT OFF RODS AND MK3 CONTROL RODS IN A MK 3 GUIDE TUBE	RTD/TN(79)66	SIMMERS DA	
27971.pdf	THE EFFECT OF IRRADIATION CREEP ON PFR MK1A SHUT OFF RODS	rtd/tn(79)70	RIDING DJ	
27973.pdf	PFR ABSORBER RODS GUIDE TUBES AND ASSOCIATED ITEMS APPROACHING LIMITS AT RELAOD 4 DURING RUN 5	RTD/TN(79)73	SIMMERS DA	
27974.pdf	AS ASSESSMENT OF IN CORE BOW MEASUREMENTS USING THE PFR CHARGE MACHINE	RTD/TN(79)91	SIMMERS DA	
27975.pdf	NOTES ON THE EFFECT OF SHORTENING THE GUIDE TUBE REMOVAL TOOL SPIKE	RTD/TN(79)92	DIXON JS	
27978.pdf	PFR ABOVE CORE FLOWMETER LOCATION TUBES HANDLING AND DISTORSION CONSIDERATIONS	RTD/TN(80)102	SIMMERS DA	1980
28053.pdf	REACTIVITY NOISE LEVELS IN PFR DUE TO CONTROL ROD VIBRATIONS	PFR/SWP/ESC/P (79)12	LORD DJ	
28057.pdf	REACIVITY NOISE LEVELS IN PFR DUE TO CONTROL ROD VIBRATIONS	PFR/SWP/ESC/P (79)12	LORD DJ	
28274.pdf	B ASSEMBLIES AND GUIDE TUBES MAY 1982 PART 1 LENGTH CHANGE MEASUREMENTS IN EN58B-CWE MEASUREMENTS IN EN58B-CW	DFMC/P(82)11; FRDC/FEWP/P(8 2)28	ILLEY RJ;WILLIAMS DP;BROOK AJ; ILLEY RJ;WILLIAMS DP;BROOK AJ;	1982
28301.pdf	PREPARATIONS FOR RELOAD 5 INITIAL GUIDE TUBE AND GUIDE TUBE REMOVAL TOOL TESTING	PFR/OPS/N637; FRTF/P(81)76	ALLCOCK CC	
28538.pdf	BRITTLE LATCHES MODIFICATION OF MK2 IN CORE GUIDE TUBES FOR RE-USE IN PFR	FRTF/P(81)89	WEBB J;ROUMPH E	1981
28582.pdf	GUIDE TUBES WITH BRITTLE LATCHES	FRTF/P(81)92;R TD/TN(81)193	DODD JA	1981
28587.pdf	A REVIEW OF SUPPORT FOR PFR	FROC/P(82)3	WEBB J	1982
28589.pdf	A REVIEW OF PFR SUPPORT WORK	FROC/P(82)8	WEBB J	1982
31601.pdf	PROPOSED PROGRAMME FOR HIGH TEMPERATURE CRACK GROWTH STUDIES OF RELEVANCE TO PFR CORE SUPPORT STRUCTURE	PFR/SIAG/P(91) 55	CURBISHLEY I	1991

Reference No.	Title	Location	Authors	Year
31603.pdf	PFR CORE SUPPORT STRUCTURE MONITORING OPTIONS FOR CORE SUPPORT MOVEMENTS-AN INITIAL REVIEW	PFR/SIAG/P(91) 59	MELHUISH D	1991
31681.pdf	MECHANICAL TESTS IN SUPPORT OF THE SAFETY CASE FOR PFR CORE SUPPORT STRUCTURE HEAT EXCHANGERS	PFR/SIAG/P(93) 150	PICKER C;ORTNER SR	1993
31682.pdf	MECHANICAL TESTS IN SUPPORT OF THE SAFETY CASE FOR PFR CORE SUPPORT STRUCTURE HEAT EXCHANGERS	PFR/SIAG/P(93) 150	PICKER C;ORTNER SR	1992
31683.pdf	MECHANICAL TESTS IN SUPPORT OF THE SAFETY CASE FOR PFR CORE SUPPORT STRUCTURE + REVIEW OF PFR/SIAG/P(92)150 ISSUE 3 HEAT EXCHANGERS	PFR/SIAG/P(93) 150	PICKER C;ORTNER SR	1993

Table 3: Wood Report List: Strongback

Reference No.	Title	Location	Authors	Year
08322.pdf	UNDERSODIUM ULTRASONIC VIEWING IN PFR. AN OUTLINE OF A METHOD FOR ASSESSING DIAGRID SETTING TILT IN - SERVICE		CROAD A	1987
12267.pdf	THE PROPOSED MODIFICATION TO THE PFR CORE SUPPORT ARRANGEMENT	PFR/FEDWP/P(70)115	HACKNEY S	
16822.pdf	CORE SUPPORT INTEGRITY PROGRAMME	FRDCC/SIWG/DASG/P(79)7	SEED G;MITCHELL C;ASPDEN G	
16840.pdf	EXAMINATION OF DESIGN AND CONSTRUCTION OF VESSEL AND STRONGBACK AND PREPERATION OF SPECIFICATION FOR STRUCTURAL TESTS	SIWG/P(80)33; FRDCC/SIWG/DASG/P(80)33	LEGGATT RH;OGLE MH	
17339.pdf	STATEMENT OF THE PRESENT SAFETY CASE REGARDING FAILURE OF THE PFR CORE SUPPORT STRUCTURE	PFR/SWP/P(90)39	HENDERSON JDC	1990
19837.pdf	STATEMENT ON THE SAFETY CASE REGARDING FAILURE OF THE PFR CORE SUPPORT STRUCTURE (CSS)	PFR/SWP/P(90)44	GREGORY CV	1990
19838.pdf	THE CONSEQUENCES OF CORE SUPPORT STRUCTURE FAILURE	PFR/SWP/P(90)45	GREGORY CV	
19947.pdf	THE PRESENT STATE OF THE SAFETY CASE FOR THE PFR CORE SUPPORT STRUCTURE	PE1/1466;PFR/TC/P(91)17	HENDERSON JDC	1991
22493.pdf	THE SAFETY CASE FOR THE PFR CORE SUPPORT STRUCTURE	PFR/SWP/P(90)42	TOMKINS B	
22494.pdf	COMMENTS ON INSPECTION SUPPLEMENTARY SUPPORT AND MONITORING	PFR/SWP/P(90)43	HENDERSON JDC	1990
25165.pdf	EXTRACTS FROM THE PFR DESIGN SAFETY REPORT FOR THE DIAGRID,FUEL ELEMENT CARRIERS AND CORE SUPPORT STRUCTURE	PFR/LLF/P(85)6		
25252.pdf	AN UPDATE OF THE INCREDIBILITY OF FAILURE ARGUMENT FRO THE PFR CORE SUPPORT STRUCTURE	PFR/SWP/P(90)49	PICKER C	1990
31169.pdf	PFR CORE SUPPORT SSTRUCTURE WORK PROGRAMME AND OTHER WORK ITEMS	PFR/SIAG/P(90)24	LOVE,JB	1990
31171.pdf	COMMENTS ON THE ANTICPATED QUALITY OF A NUMBER OF WELDS IN THE PFR CORE SUPPORT STRUCTURE	PFR/SIAG/P(90)28/FR/ESD/P(90)2165	GORE,AW	1990
31172.pdf	CRITICAL CRACK LENGTH IN THE PFR CORE SUPPORT STRUCTURE	PFR/SIAG/P(91)44	GREEN,D; BATE,SK	1991
31173.pdf	PRIMARY STRESSES IN THE PFR CORE SUPPORT STRUCTURE	PFR/SIAG/P(91)60	CLARKE,PW	1991

Reference No.	Title	Location	Authors	Year
31174.pdf	STRESS ANALYSIS OF THE PFR PRIMARY TANK AND LEAK JACKET WITH CONSIDERATION AND DEFECT TOLEANCE ON THE BASIS OF TEARING INITIATION AND COVER GAS PRESSURE FOR BUCKLING	PFR/SIAG/P(92) 102	GREEN D	1993
31175.pdf	INTEGRITY ASSESSMENT OF THE PFR CORE SUPPORT STRUCTURE TRANSITION WELD	PFR/SIAG/P(93) 176	DANIELS,BD	1993
31176.pdf	STRUCTURAL PERFORMANCE DEPARTMENT AEA TECHNICAL SERVICES RISLEY	PFRDMA/P228; D/108/9.1	DANIELS,BD;B ROADHOUSE, BJ;GREEN,D	1994
31594.pdf	PFR CORE SUPPORT STRUCTURE DEGRADATION	PFR/SIAG/P(90) 42	KNOWLES JA	1991
31599.pdf	FURTHER CONSIDERATION OF MAXIMUM FLAW SIZES IN PFR CORE SUPPORT STRUCTURE (CSS) AT BEGINNING OF LIFE	PFR/SIAG/P(91) 56	PICKER C	1991
31601.pdf	PROPOSED PROGRAMME FOR HIGH TEMPERATURE CRACK GROWTH STUDIES OF RELEVANCE TO PFR CORE SUPPORT STRUCTURE	PFR/SIAG/P(91) 55	CURBISHLEY I	1991
31602.pdf	ASSESSMENT OF RECTANGULAR PATCH PRESSURE TEST AND IMPLICATIONS FOR PATCH PLATE DESIGN	PFR/SIAG/P(91) 57	DANIELS BD	1991
31603.pdf	PFR CORE SUPPORT STRUCTURE MONITORING OPTIONS FOR CORE SUPPORT MOVEMENTS-AN INITIAL REVIEW	PFR/SIAG/P(91) 59	MELHUISH D	1991
31610.pdf	PEER REVIEW OF THE PFR CORE SUPPORT STRUCTURE SAFETY CASE INPFR/SWP/P(90)49 AND SUPPORTING DOUMENTS+RESPONSE TO PEER REVIEW	PFR/SIAG/P(91) 68	PICKER P;GREEN D	1991
31611.pdf	COMBINED PROGRAMME OF HIGH TEMPERATURE CRACK GROWTH STUDIES FOR PFR CORE SUPPORT STRUCTURE	PFR/SIAG/P(91) 69	CURBISHLEY I;HIPPSLEY CA	1991
31612.pdf	REVIEW OF THE INSPECTION POTENTIAL OF THE PFR CORE SUPPORT STRUCTURE (CSS)	PFR/SIAG/P(91) 70	THOMAS G	1991
31614.pdf	RESULTS OF PFR CORE SUPPORT STRUCTURE MONITORING	PFR/SIAG/P(91) 72	HENDERSON JDC	1991
31615.pdf	PFR CORE SUPPORT FRACTURE TOUGHNESS PROGRAMME	PFR/SIAG/P(91) 73	O'DONNELL IJ	1991
31656.pdf	A CODE ASSESSMENT OF 0.14G SEISMIC LOADING ON THE PFR CORE SUPPORT STRUCTURE	PFR/SIAG/P(92) 126	CLARKE PW	1992
31657.pdf	A CODE ASSESSMENT OF 0.14G SEISMIC LOADING ON THE PFR CORE SUPPORT STRUCTURE	PFR/SIAG/P(92) 126	CLARKE PW;O'GARA DM	1993

Reference No.	Title	Location	Authors	Year
31681.pdf	MECHANICAL TESTS IN SUPPORT OF THE SAFETY CASE FOR PFR CORE SUPPORT HEAT EXCHANGERS	PFR/SIAG/P(93) 150	PICKER C;ORTNER SR	1993
31682.pdf	MECHANICAL TESTS IN SUPPORT OF THE SAFETY CASE FOR PFR CORE SUPPORT STRUCTURE HEAT EXCHANGERS	PFR/SIAG/P(93) 150	PICKER C;ORTNER SR	1992
31683.pdf	MECHANICAL TESTS IN SUPPORT OF THE SAFETY CASE FOR PFR CORE SUPPORT STRUCTURE + REVIEW OF PFR/SIAG/P(92)150 ISSUE 3 HEAT EXCHANGERS	PFR/SIAG/P(93) 150	PICKER C;ORTNER SR	1993
31691.pdf	EXAMINATION OF THE CREEP-FATIGUE TESTS ON THE PFR ABOVE CORE SUPPORT REPLICAS USING THE R5 ASSESSMENT PROCEDURE	PFR/SIAG/P(93) 158	MAY KA;CURBISHL EY I	1993
31692.pdf	THE SIGNIFICANCE OF THE SINUSOIDAL STRESS VARIATION ON THE CRITICAL CRACK LENGTH AT WELD M IN THE PFR CORE SUPPORT STRUCTURE	PFR/SIAG/P(93) 159	GREEN D;DANIELS B	1993
31693.pdf	THE POSSIBILITY OF BUCKLING IN THE PFR CORE SUPPORT STRUCTURE	PFR/SIAG/P(93) 160	GREEN D	1993
31694.pdf	THE POSSIBILITY OF BUCKLING IN THE PFR CORE SUPPORT STRUCTURE	PFR/SIAG/P(93) 160	GREEN D	1993
31696.pdf	FRACTURE TOUGHNESS TESTS IN SUPPORT OF THE SAFETY CASE FOR PFR CORE SUPPORT STRUCTURE	PFR/SIAG/P(93) 163	O'DONNELL IJ	1993

Table 4: Wood Report List: Above Core Structure

Reference No.	Title	Location	Authors	Year
01809.pdf	SAMPLES FROM JO5 FOR THERMAL STRIPING DAMAGE ASSESSMENT	FROC/P(84)122;PFRESC/P(84)17	BATES PM	1984
03794.pdf	FURTHER RESULTS FROM THE THERMOCOUPLE PROBES ADJACENT TO THE ANTI VIBRATION GRID BAFFLE ATTACHMENT IN PFR	PFR EXPERIMENTAL RESULTS SHEET NO.103;OETD.TECH NOTE NO.175	CROWE DS;SUTHERLAND AJ	
04224.pdf	THEORETICAL ANALYSIS OF THE FLOW BETWEEN THE TOP OF THE CORE AND THE ACS BAFFLE USING THE PHOENICS COMPUTER CODE	TN/P(85)787;THSG/P(85)122 ADDENDUM	SMITH AG	1985
04301.pdf	CRACK PROPAGATION IN PFR ABOVE CORE STRUCTURE SHROUD TUBES DUE TO THERMAL STRIPING	DCWG/P(82)335;ACSCM/P(77)32	PEARCE JHB	
04304.pdf	THE EFFECT OF TEMPERATURE WAVE SHAPE ON SURFACE STRAIN RANGE UNDER HIGH FREQUENCY THERMAL STRIPING CONDITIONS	DCWG/P(82)334;ACSCM/P(77)27	PEARCE JHB	
05253.pdf	THEORETICAL ANALYSIS OF THE FLOW BETWEEN THE TOP OF THE CORE AND THE ACS BAFFLE USING THE PHOENICS COMPUTER CODE	THSG/P(85)122	SMITH AG	
05343.pdf	A SAFETY CASE FOR THE CONTINUED OPERATION OF THE PFR AFTER TO' IN VIEW OF THE CURRENT POSITION OF THE ASSESSMENT OF THE ABOVE CORE SRUCTURE MARCH, 1977	SWP/P(77)18;ACSCM/P(77)24;TF/P(77)235	BROADLEY D;ROSE RT;DURSTON JG	
05367.pdf	ELEMENTARY STRESS ANALYSIS OF THE PFR CENTRAL SHROUD TUBE	FRD/TN/P(78)289	GREEN D	
06027.pdf	EXTENDED ANALYSIS OF THE FLOW BETWEEN THE TOP OF THE CORE AND THE ACS BAFFLE USING THE PHOENICS COMPUTER CODE.	FRD/TN/P(86)874	SMITH AG	1986
06250.pdf	PROPOSAL FOR A 1/4 SCALE 60 SECTOR WATER MODEL TO VALIDATE VICSEN PREDICTIONS OF ACS INTERNAL FLOWS	FREWG/P(86)217	SMITH MR;HULME G	1986
07614.pdf	PROPOSAL FOR A 1/4 SCALE 60o SECTOR ACS WATER MODEL	FREWG/P(86)305	BOLEY	1986
08871.pdf	PROPOSAL FOR A 1/4 SCALE 60 DEGREES SECTOR WATER MODEL TO VALIDATE VICSEN PREDICTIONS OF ACS INTERNAL FLOWS	RES INT 2910 ISSUE A-; FR/THSG/P(86)181	SMITH R;HULME G	1986
09203.pdf	EXTENDED ANALYSIS OF THE FLOW BETWEEN THE TOP OF THE CORE AND THE ACS BAFFLE USING THE PHOENICS COMPUTER CODE	FR/THSG/P(87)321;FRD/TN(86)874	SMITH AG	1986
13294.pdf	ANALYSIS OF AN ACS BAFFLE PLATE TO SHROUD TUBE CONNECTION		SINCLAIR EG	1989
13859.pdf	SINGLE POINT INELASTIC ANALYSIS OF AN ACS BAFFLE PLATE		SINCLAIR EG	1989

Reference No.	Title	Location	Authors	Year
17124.pdf	THEORETICAL ANALYSIS OF THE FLOW BETWEEN THE TOP OF THE CORE AND THE ACS BAFFLE USING THE PHOENICS COMPUTER CODE	FR/TN(85)787;FR/THSG/P(85)122	SMITH AG	1985
18101.pdf	A REVIEW OF WORK RELEVANT TO THE PFR ABOVE CORE STRUCTURE SINCE MAY 1977 AND THE REVISED SAFETY CASE FOR ITS CONTINUED OPERATION	FRD/TN/P(82)510;PFR/SWP/P(82)18	ROSE RT;GREEN D	1982
18103.pdf	PFR ANNUAL SAFETY SUMMARY (MAY 1984)	PFR/SWP/P(84)25	HENDERSON JDC	
20510.pdf	SOME ASPECTS OF HEAT TRANSFER AFFECTING THER THERMAL STRIPING PROBLEM	ND-M-426	DAWSON CW	
20512.pdf	COMPARISON OF PFR ABOVE CORE TEMPERTURE FLUCTUATIONS WITH RNL 1/9TH SCALE AIR MODEL DATA	ND-M-608;PFR/SWP/P(78)86	BETTS C;ASHTON MW;SPANTON JH	
21473.pdf	THERMAL STRESSES IN THE ABOVE CORE STRUCTURE OF THE PFR	PFR/SWP/P(76)39;PFR/TF/P(76)165	BROADLEY D	
21479.pdf	STRATEGY FOR THE ASSESSMENT OF THE PFR ABOVE CORE STRUCTURE AND THE SAFETY CASE FOR CONTINUED POWER OPERATION	PFR/SWP/P(76)54;PFR/ASCM/P(76)8;PFR/TF/P(76)184	BROADLEY D	
21480.pdf	THERMAL FLUCTUATIONS IN PFR ABOVE CORE STRUCTURE A REVIEW OF FLOW AND HEAT TRANSFER EXPERIMENTS	ACSCM/P(77)13;PFR/SWP/P(77)1	BETTS C	
21481.pdf	PFR THERMAL STRIPING PROBLEM SIGNIFICANCE OF DEFECTS ARISING FROM THERMAL STRESSING AND/OR OTHER STRESS CYCLES	PFR/TF/P(77)206;PFR/SWP/P(77)9;PFR/ACSM/P(77)8	COWAN A;WOOD DS	
21483.pdf	TEMPERATURE FLUCTUATIONS WITHIN PFR CONTROL ROD SHROUD TUBES REML WATER TEST RESULTS	ACSCM/P(77)14;PFR/SWP/P(77)14	FEWSTER J	
21484.pdf	SODIUM MIXING TEE EXPERIMENT (SoMITE) THERMAL CYCLING OF 316 STAINLESS STEEL	ACSCM/P(77)15;PFR/SWP/P(77)15	LANGRIDGE EG;BROMIDGE NAC;SCOTT R	
21485.pdf	AIR MODEL MEASUREMENTS OF PFR ABOVE CORE TEMPERATURE FLUCTUATIONS	PFR/SWP/P(77)16;ACSCM/P(77)16	COCHRANE SJR	
21486.pdf	SODIUM TEMPERATURE VARIATIONS OBSERVED ABOVE THE CORE OF THE PFR BY AN INDUCTIVE PROBE	ACSM/P(77)20;PFR/SWP(77)23	DEAN SA	
21495.pdf	METHODS FOR ASSESSMENT OF DAMAGE TO THE ABOVE CORE STRUCTURE	PFR/SWP/P(77)69;PFR/ACSCM/P(77)37;FRD/TN/P(78)250	DIXON M	
21605.pdf	PFR ABOVE-CORE TEMPERATURE FLUCTUATIONS PRELIMINARY RESULTS FROM THE 1/9TH SCALE AIR MODEL FEBRUARY 1978	PFR/SWP/P(78)16;ACSCM/P(78)45	BETTS C;ASHTON MW	
21606.pdf	A BRIEF DISCUSSION OF SOME FACTORS AND ASSUMPTIONS USED TO CALCUALTE THE FATIGUE DAMAGE TO SHROUD TUBE P09	PFR/SWP/P(78)24;PFR/ACSCM/P(78)41;FRD/DM(78)211	DURSTON JG	

Reference No.	Title	Location	Authors	Year
21607.pdf	COVER NOTE ON CURRENT STRUCTURAL TESTS FOR THE ASSESSMENT OF DAMAGE TO THE PFR ABOVE CORE STRUCTURE	PFR/SWP/P(78)35;PFR/ACSCM/P(78)43		
21608.pdf	MECHANICAL PROPERTIES WORK RELEVANT TO THE PFR ABOVE CORE STRUCTURE STATUS OF WORK AND RELEVANCE TO MAY 1977 SAFETY CASE	PFR/ACSCM/P(78)44;PFR/SWP/P(78)36	WOOD DS;SANDERSON SJ	
21610.pdf	EXAMINATION OF PFR CENTRE PHYSICS THIMBLE AND SSD SHROUD TUBE FOR THERMAL SHOCK AND THERMAL STRIPING DAMAGE	PFR/SWP/P(78)57;FRD/TN/P(78)271;PFR/ACSCM/P(78)47	DURSTON JG	
21613.pdf	A PROGRAMME OF THERMAL STRIPING MEASUREMENTS ON THE RNL 1/9 SCALE AIR MODEL	PFR/SWP/P(78)62;ACSCM/P(78)48	BETTS C	
21615.pdf	PFR AVG AND BAFFLE ATTACHMENT LIFE PREDICTIONS AS A FUNCTION OF REACTOR TRIP TEMPERATURE TRANSIENT SHAPES	PFR/SWP/P(78)68;PFR/ACSCM/P(78)50;FRD/TN/P(78)279	DURSTON JG	
21616.pdf	ABOVE CORE STRUCTURE AIR THERMAL SHOCK EXPERIMENT VALIDATION OF RESULTS WITH REACTOR DATA	PFR/SWP/P(78)69;FRD/TN/P(78)285;ACSCM/P(78)52	DIXON M	
21617.pdf	ASSESSMENT OF THE STEADY-STATE AND TRIP TRANSIENT TEMPERATURES EXPERIENCED BY THE AVG SUPPORT COLUMN BAFFLE ATTACHMENT FOR USE IN THE THERMAL SHOCK EXPERIMENT	PFR/ACSCM/P(78)42;PFR/SWP/P(78)73;PFR/TN/P(78)246	DIXON M	
21618.pdf	FURTHER INVESTIGATION OF TEH PFR ANTI-VIBRATION GRID ENDURANCE	PFR/SWP/P(78)74;ACSCM/P(78)55	PEARCE JHB	
21619.pdf	PFR ABOVE CORE FLOW STUDIES USING A 1/4 SCALE SECTOR WATER MODEL	PFR/SWP/P(78)75;FRTHDC/P(78)8	CONROY PJ;ROBINSON RGJ	
21620.pdf	COMPARISON OF PFR ABOVE-CORE TEMPERATURE FLUCTUATIONS WITH RNL 1/9TH SCALE AIR MODEL DATA	ND-M-608(R);PFR/SWP/P(78)86	BETTS C;ASHTON MW;SPANTON JH	
21621.pdf	ELEMENTARY STRESS ANALYSIS OF THE PFR CENTRAL SHROUD TUBE	FRD/TN/P(78)289;PFR/SWP/P(79)1	GREEN D	
21623.pdf	LINEAR ELASTIC FRACTURE MECHANICS ASSESSMENT OF THE ANTIVIBRATION GRID SUPPORT TUBE ATTACHMENT	FRD/P(79)14;PFR/SWP/P(79)19	GREEN D	
21624.pdf	AIR THERMAL SHOCK EXPERIMENT	PFR/SWP/P(79)29;ACSCM/P(79)61	WARDLE PS	
21625.pdf	THERMAL STRIPING OF AVG SUPPORT FEATURE	PFR/SWP/P(79)36;ACSCM/P(79)62	BOORMAN C	
21626.pdf	SAFETY REPORT FOR THE LOADING OF THE ABOVE CORE STRUCTURE SUPPORT	PFR/SWP/P(79)7;PFR/SWP//P(79)33	STACEY J	
21628.pdf	ADDITIONAL NOTES FOR DRAFT ND-M-789(R) ABOVE CORE STRUCTURE AIR THERMAL SHOCK EXPERIMENT	PFR/SWP/P(79)38;ACSCM/P(79)64	LLOYD GJ;KENNETT E	

Reference No.	Title	Location	Authors	Year
21629.pdf	A REVIEW OF CREEP/FATIGUE DAMAGE ASSESSMENT OF THE PFR BAFFLE ATTACHMENT DETAIL	PFR/SWP/P(79)42;FRD/TN/P(79)344	ROSE RT;DURSTON JG;EICKHOFF KG	
21630.pdf	METHODS FOR ASSESSMENT OF DAMAGE TO THE ABOVE CORE STRUCTURE BAFFLE ATTACHMENT	PFR/SWP/P(79)45;PFR/ACSCM/P(79)65;FRD/TN(79)352	DIXON M	
21633.pdf	TEMPERATURE DISTRIBUTIONS AROUND BAFFLE ATTACHMENT IN PFR	PFR/SWP/P(79)61;PFR/ACSCM/P(79)66;FRD/TN(79)358	DIXON M	
21634.pdf	PFR ABOVE-CORE TEMPERATURE FLUCTUATIONS MEASUREMENT OF THRMAL STRIPING PREDICTION PARAMETERS AND REACTOR/RIG COMPARISON	ND-M-1009(R);PFR/SWP/P(79)66	SPANTON JH	1980
21637.pdf	FRACTURE MECHANICS RE-ASSESSMENT OF THE ANTIVIBRATION GRID SUPPORT TUBE ATTACHMENT	PFR/SWP/P(80)18;FRD/TN(80)383	GREEN D	1980
21642.pdf	NON-LINEAR STRESS ANALYSIS WITH KINEMATIC HARDENING OF THE ATTACHMENT OF THE ABOVE CORE STRUCTURE SUPPORT TUBE TO THE ROTATING SHIELD BAFFLE	PFR/SWP/P(80)47;FRD/TN(80)406	GREEN D	1980
22046.pdf	A SAFETY SUMMARY FOR ACSNI	PFR/SWP/P(85)28	HENDERSON JDC	1985
25137.pdf	METHODS FOR ASSESSMENT OF DAMAGE TO THE ABOVE CORE STRUCTURE BAFFLE ATTACHMENTS	PFR/SWP/P(79)45;PFR/ACSCM/P(79)65;TN/P(79)352	DIXON M	
25138.pdf	DAMAGE ASSESSMENT OF THE PFR ANTIVIBRATION GRID SUPPORT TUBE ATTACHMENT BASED ON CRACK GROWTH CONSIDERATIONS	TN/P(81)437;PFR/SWP/P(81)10	GREEN D	1981
25474.pdf	THERMAL FATIGUE CRACK ARREST IN THE PFR ABOVE CORE STRUCTURE	DCWG/P(82)333;ACSCM/P(76)6	PEARCE JHB	
25476.pdf	BOUNDARY LAYER ATTENUATION OF SODIUM TEMPERATURE FLUCTUATION NEAR THE BOTTOM OF THE CONTROL ROD GUIDE TUBES	DCWG/P(82)337;CFR/EST/P291;CFR/THWG/P(75)114	BELL RT	
25477.pdf	INVESTIGATION OF A PFR SHROUD TUBE UNDER THERMAL SHOCK AND THERMAL STRIPING CONDITIONS	DCWG/P(82)338;ACSCM/P(78)40	PEARCE JHB	
25479.pdf	AMELIORATION OF THERMAL STRIPING DAMAGE IN PFR SHROUD TUBES DUE TO INCOPLETE SPATIAL COHERENCE	ND-M-554;ACSCM/P(78)51	LILLER AG	
25860.pdf	FAILURE OF ANTI VEBRATION GRID SUPPORTS PFR SAFETY IMPLICATIONS	NRD/R(78)62	WEBB J	
26519.pdf	THE EFFECT OF TEMPERATURE ON THE ENDURANCE ASSESSMENT OF PFR ABOVE CORE STRUCTURE	PFR/SWP/P(77)37;ACSCM/P(77)28	PEARCE JHB	
26520.pdf	CREEP FATIGUE DAMAGE ASSESSMENT IN PFR ABOVE CORE STRUCTURE BY THE ELASTIC ROUTE OF CODE CASE 1592	ACSCM/P(77)36	JOBSON DA;PEARCE JHB	

Reference No.	Title	Location	Authors	Year
26684.pdf	PFR ABOVE CORE STRUCTURE FLOW EXPERIMENTS ON A 1/5TH SCALE WATER MODEL PRELIMINARY RESULTS	PFR/SWP/P(77)17;ACSCM/P(77)17	WINN WR;CONROY PJ;TAYLOR AF	
26957.pdf	AN EXPERIMENTAL STUDY OF BUOYANCY EFFECTS AND MIXING IN A 1/17 SCALE WATER MODEL OF THE ABOVE CORE PLENUM OF PFR	ND-M-786	BOOTH DA;POWELL WR	1982
27522.pdf	THERMAL SHOCK IN THE PFR ABOVE CORE STRUCTURE SUPPORT FEATURE	ND-M-1026;PFR/SWP/P(80)6	CLAYTON AM	1980
27523.pdf	ENDURANCE TESTING OF REPLICA ABOVE CORE STRUCTURE SUPPORT FEATURES IN AIR	ND-M-1231	WADLE PS;SARGENT TH;KIRKLAND GR	1980
27568.pdf	ENGINEERING SPECIFICATION FOR A THERMAL SHOCK TEST ON A PFR ACS SUPPORT FEATURE IN THE HIGH TEMPERATURE SODIUM LOOP	RED 112/77	MATHER B	
27577.pdf	A SAFETY CASE FOR THE CONTINUED OPERATION OF THE PFR AFTER TO IN VIEW OF THE CURRENT POSITION OF THE ASSESSMENT OF THE ABOVE CORE STRUCTURE MARCH 1977	PFR/SWP/P(77)18;PFR/TF/P(77)235;ACSCM/P(77)24	BROADLEY D;ROSE RT;DURSTON JG	
27589.pdf	ENDURANCE OF PFR ABOVE CORE STRUCTURE RELEVANCE OF DFR MK2 SUPPORT STOOL	PFR/SWP/P(78)33;TN/P(78)248	ROSE RT	
27594.pdf	PFR ABOVE CORE STRUCTURE SUPPORT TUBE ATTACHMENT PROPOSED THERMAL SHOCK EXPERIMENTS	PFR/SWP/P(77)54;PFR/TF/P(77)270	EICKHOFF KG	
27594.pdf	PFR ABOVE CORE STRUCTURE SUPPORT TUBE ATTACHMENT PROPOSED THERMAL SHOCK EXPERIMENTS	PFR/SWP/P(77)54;PFR/TF/P(77)270	EICKHOFF KG	
27595.pdf	ABOVE CORE STRUCTURE AIR THERMAL SHOCK EXPERIMENT VALIDATION OF RESULTS WITH REACTOR DATA	PFR/SWP/P(78)69;TN/P(78)285;ACSCM/P(78)52	DIXON M	
27611.pdf	ASSESSMENT OF THE STEADY STATE AND TRIP TRANSIENT TEMPERATURES EXPERIENCED BY THE AVG SUPPORT COLUMN BAFFLE ATTACHEMNTS FOR USE IN THE THERMAL SHOCK EXPERIMENT	TN/P(78)246;ACSCM/P(78)42	DIXON M	
27616.pdf	OUTLINE SPECIFICATION FOR STRESS ANALYSISI OF PFR ANTIVIBRATION GRID	TN/P(80)369	ROSE RT	1980
27622.pdf	ABOVE CORE STUCTURE INVESTIGATION SUMMARY OF THE FABRICATION DETAILS OF THE ANTI-VIBRATION GRID STRUCTURE	RRD(77)REPORT 43;ACSCM/P77	STACEY J	
27624.pdf	AN INVESTIGATION OF CRACK DEVELOPMENT UNDER THERMAL FATIGUE CYCLING CONDITIONS WITH ALLOWANCE FOR REDUCING SECTION THICKNESS AND ELLIPTICAL CRACK SHAPE	ACSCM/P(76)11	PEARCE JHB	
27625.pdf	CRACK I ITIATION OF STAINLESS STEEL UNDER THERMAL STRIPING	ACSCM/P(77)19;PFR/TF/P(77)212	WOOD DS	

Reference No.	Title	Location	Authors	Year
27626.pdf	A SUMMARISED ASSESSMENT OF THE CRITICAL STRESS CONDITIONS OCCURING INTHE PFR ABOVE CORE STRUCTURE	ACSCM/P(77)22	PEARCE JHB	
27636.pdf	FURTHER INVESTIGATION OF THE STRESS CONDITIONS OCCURRING IN THE PFR ABOVE CORE STRUCTURE	ACSCM/P(77)35	PEARCE JHB	
28728.pdf	ESTIMATION OF THE ENDURANCE OF PFR ABOVE CORE STRUCTURE ANTI-VIBRATION GRID	ACSCM/P(78)46	PEARCE JHB	
28732.pdf	PFR ABOVE CORE STRUCTURE THERMAL SHOCK PRELIMINARY HYDRAULICS CONSIDERSATIONS ON ALTERNATIVE BAFFLING	ACSCM/P(78)49 ;PFR/SWP/P(78)63;ND-M-508	TAYLOR AF;BETTS C;FRANCE J	
28737.pdf	SUMMARY OF RNL EXPERIMENTAL FLOW MODEL WORK RELATED TO THERMAL SHOCK PROBLEMS IN PFR	ACSCM/P(79)56	TAYLOR AF	
28738.pdf	SUMMARY OF THE THERMAL SHOCK EXPERIMENTAL; PROGRAMME IN RELATION TO THE ACS MOUNTING FIXTURE	ACSCM/P(79)57	BUCKLEY FI	
28739.pdf	NOTE ON THE CURRENT STATUS OF PFR THERMAL STRIPING WORK	ACSCM/P(79)58	BETTS C	
28740.pdf	HEAT TRANSFER ASPECTS RELATED TI THERMAL STRIPING	ACSCM/P(79)59	SHERIFF N	
28741.pdf	ANALYSIS OF THE PFR ANTI-VIBRATION GRID A REVIEW OF CTS WORK TO MAY 1979	ACSCM/P(79)60 ;ND-M-808	PEARCE JHB	
28744.pdf	A PRELIMINARY LOOK AT THE MECHANICS OF A LOW LEVEL BAFFLE FOR PFR ABOVE CORE STRUCTURE	ACSCM/P(79)63	BADDLEY AH	
28746.pdf	RE ASSESSMENT OF THE PFR ANTI VIBRATION GRID ENDURANCE OCTOBER 1979	ACSCM/P(79)68	PEARCE JHB	
28747.pdf	DESIGN CRITERIA FOR THERMAL STRIPING A REVIEW	ACSCM/P(79)67 ;DCWG/P(79)20 1;ND-M-706	PEARCE JHB	
	NEW SUPPORTS FOR ANIT VIBRATION GRID PFR DOUNREAY	ACSCM/P(77)39 ;PFR/TF/P(77)2 86	WEBB J	

Table 5: Wood Report List: Intermediate Heat Exchangers IHX, IHX Pods and Reactor Jackets

Reference No.	Title	Location	Authors	Year
00650.pdf	THE EFFECT OF PRESSURE PULSES ON THE INTEGRITY OF PARTIAL PENETRATION WELDS IN THE PFR IHX OUTER SHELL	TN/P(83)588; PFR/SWP/P(83)10;ADD 1.	GREEN D	1983
00676.pdf	PFR IHX TOP: STRESS ANALYSIS SUMMARY REPORT	TN/P(83)625	CHURCH A;CLARK JS	1983
00678b.pdf	ESTIMATED CRACK GROWTH LIFE AT THE LOWER TUBEPLATE/SHELL JOINT OF THE PFR IHX UNDER THERMAL STRIPING CONDITIONS	TN/P(83)628;PFR/SWP/P(83)39	GREEN D	1983
00681.pdf	PFR IHX REMOVAL - AN INITIAL ASSESSMENT OF PRIMARY CIRCUIT FLOWS AND HYDRAULIC LOADS DURING PLUG INSERTION	TN/P(83)638	SMITH AG	1983
00697.pdf	REASSESSMENT OF THE CRACK GROWTH LIFE UNDER THERMAL STRIPING CONDITIONS OF THE PFR IHX LOWER TUBE PLATE/SHELL WELD	PFR/SWP/P(84)1;TN/P(84)662	GREEN D	1984
00911.pdf	PFR-IHX REMOVAL SCHEME - STATUS AT NOVEMBER 1982	OC/P(82)34	HAYDEN O	1981
00928.pdf	REPLACEMENT IHX FOR THE PFR	OC/P(83)57	HAYDEN O	1983
00933.pdf	A REVIEW OF EQUIPMENT NEEDED FOR REPAIR TO AN IHX - PFR DOUNREAY	OC/P(83)62	WEBB J	1983
00934.pdf	PFR IHX REPLACEMENT	OC/P(83)63	HAYDEN O	1983
00935.pdf	A REVIEW OF EQUIPMENT NEEDED FOR INSPECTION OF AN IHX, PFR DOUNREAY	OC/P(83)64	WEBB J;HUNTER D	1983
00951.pdf	REMOVAL OF AN IHX - FURTHER ASPECTS	OC/P(83)80	WEBB J;HUNTER D	1983
01146.pdf	A SUMMARY OF THE POSITION ON THE ASSESSMENT OF POTENTIAL THERMAL STRIPING TO THE BOTTOM TUBEPLATE OF THE PFR IHX DURING SINGLE CIRCUIT OPERATION	PFR/SWP/P(84)47;TN/P(84)702	BROADLEY D	1984
01188.pdf	IHX CLEAN-UP PROCEDURE FOLLOWING A MAJOR SODIUM/WATER REACTION	OC/P(84)119	WEBB J	1984
01377.pdf	LEAKAGE FLOW PAST CLOSED IHX VALVES	OPS/N.308	HENDERSON JDC;WILLIAMS DP	
01442.pdf	THE NEED FOR DHR COILS ON IHX REPLACEMENT (DUMMY) PLUGS	PFR/SWP/P(83)29	HENDERSON JDC	1983
01550.pdf	PFR SINGLE SECONDARY CIRCUIT OPERATION: ANALYSIS OF FLOW AND TEMPERATURE DISTRIBUTIONS UNDER THE INTERMEDIATE HEAT EXCHANGER (IHX) BOTTOM TUBEPLATE BY MEANS OF THE PHOENICS CODE - PART 1 2-DIMENSIONAL ANALYSIS. PART 2 3-DIMENSIONAL ANALYSIS	FR/THSG/P(84)8	BROWN GA;SANDERSON S;SCRIVEN J	
01784.pdf	PFR REPLACEMENT IHX ISI SUMMARY	ED.751	FENEMORE P;BOWKER LJ	1984
02098.pdf	PFR REPLACEMENT IHX ISI SUMMARY	ED 751	FENEMORE P;BOWKER LJ	1984

Reference No.	Title	Location	Authors	Year
02183.pdf	THE VALIDATION OF THE ANTHEA COMPUTER CODE USING DATA FROM THE PFR-IHX	PTWG/P(83)19	BROWN GA	1983
02184.pdf	AN ANALYSIS OF THE FLOW AND TEMPERATURE DISTRIBUTIONS UNDERNEATH THE PFR-IHX BOTTOM TUBE-PLATE USING THE PHOENUCS CODE	PTWG/P(83)23	BROWN GA;SCRIVEN J	1983
02187.pdf	AN ANALYSIS OF THE FLOW AND TEMPERATURE DISTRIBUTIONS UNDERNEATH THE PFR/IHX BOTTOM TUBEPLATE USING THE PHOENICS CODE WHEN PFR IS OPERATING WITH ONE SECONDARY CIRCUIT	NDM-2409;PTWG/P(83)30	BROWN GA;SANDERSON MRS S;SCRIVEN J	1983
02231.pdf	COMPARISON BETWEEN PFR/CFR IHX DESIGNS	PFR/SWP/P(83)25	ROYDEN R	1983
03664.pdf	PFR IHX BOTTOM TUBEPLATE THERMAL STRIPING DURING SINGLE CIRCUIT OPERATION	TN/P(85)779;FR/THSG/P(85)103	PURSLOW B	
03793.pdf	TEMPERATURE VARIATIONS ON IHX PRIMARY SODIUM INLET AND OUTLET THERMOCOUPLES WHILST OPERATING ONLY ON SECONDARY CIRCUITS 1 AND 3	PFR EXPERIMENTAL RESULTS SHEET NO.102;OETD.TECH NOTE NO.102	CROWE DS	
03797.pdf	SHIELDING MEASUREMENT IN IHX IB GAMMA MONITOR THIMBLE IN PFR	PFR EXPERIMENTAL RESULTS SHEET NO.107;OETD.TECH NOTE NO.305	CROWE DS;LORD DJ	1980
03798.pdf	THERMAL NOISE ON IHX PRIMARY SODIUM INLET AND OUTLET THERMOCOUPLES FOR ONE CIRCUIT OPERATION ON 22.8.80	PFR EXPERIMENTAL RESULTS SHEET NO.108;OETD.TECH NOTE NO.312	CROWE DS;LORD DJ	1980
03823.pdf	TEMPERATURE FLUCTUATIONS ON IHX PRIMARY SODIUM OUTLET THERMOCOUPLES FOR ONE CIRCUIT OPERATION ON 23.3.83	PFR EXPERIMENTAL RESULTS SHEET NO.133;OETD TECH NOTE NO.733	CROWE DS;SUTHERLAND AJ	1983
03867.pdf	SHIELDING MEASUREMENT IN IHX 2B GAMMA MONITOR THIMBLE IN PFR	PFR EXPERIMENTAL RESULTS SHEET NO.49	PACKWOOD A;CROWE DS;SUTHERLAND AJ	
04009.pdf	PROPOSALS TO USE A SODIUM RIG TO INVESTIGATE THE PFR IHX THERMAL STRIPING PROBLEM	OETD/SHG/P(83)27	ANDERSON R	1983
04011.pdf	MATERIAL SELECTION FOR REPLACEMENT PFR IHX'S	MTU.R.3027	BESTWICK RDW	1983
04012.pdf	PFR INTERMEDIATE HEAT EXCHANGER SPARE UNITS	TN/P(83)616	MITCHELL C;ROYDEN R	1983

Reference No.	Title	Location	Authors	Year
04637.pdf	A REVIEW OF EQUIPMENT NEEDED FOR REPAIR TO AN IHX PFR DOUNREAY PFR IHX TOP: STRESS ANALYSIS SUMMARY REPORT	FRD/P(83)625	CHURCH A;CLARK JS	1983
04976.pdf	VIEWGRAPHS USED FOR SPARE PFR IHX PRESENTATION AT DOUNREAY ON 20TH NOVEMBER 1984	FRD/TN/(84)727	MITCHELL CH;GREEN D;DEARDEN GL	1984
04977.pdf	PROTOTYPE FAST REACTOR - DOUNREAY SPARE INTERMEDIATE HEAT EXCHANGER	FRD224/SDS/001	LOMAS S	1984
05137.pdf	RNL VOTE-FUNDED WORK IN SUPPORT OF PRIMARY CIRCUIT AND IHX - OCTOBER 1985	FREWG/P(85)179	EICKHOFF KG	
05350.pdf	PFR REACTOR JACKET/IHX POD DAMAGE ASSESSMENT AS OF JUNE 1981	SWP/P(81)33	BROADLEY D	1981
05366.pdf	AN APPROXIMATE METHOD FOR ASSESSING SECONDARY CIRCUIT START UP CYCLIC DAMAGE TO THE PFR IHX TUBEPLATE	FRD/TN/P(82)502;SWP/P(82)14	GREEN D	1982
06327.pdf	STRATEGY AND PROPOSALS FOR REQUALIFYING A DAMAGED IHX REMOVED FROM PFR	C86/606/ED 803	PREECE GE;DONNELLY LF;CLEMENTS R;FORD JV	1986
07489.pdf	PFR IHX REMOVAL AND INSPECTION AND REPAIR. PARTS 1 TO 15 AND APPENDICES.	PFR/TF/P(80)422		
07492.pdf	A NOTE FOR DISCUSSION ON PFR-IHX REPLACEMENT PROPOSALS	FRD/DM/P(78)221	SEED;BOWKER;BOLTON	
07537.pdf	JOINT PROGRAM FOR TESTING A PROTOTYPE LARGE IHX IN PFR AT DOUNREAY	JPD-1	BILLURIS;SEED	
07549.pdf	PFR REPLACEMENT ROOF PLUG FINITE ELEMENT OF THE DECAY HEAT COIL THERMAL SLEEVE	TR 2035;SR 5199	HIBBS	1980
07583.pdf	PFR-IHX INSPECTION. PROJECT NO. 4.3 - IMPROVED INSTRUMENTATION	DM 108;265/HW	SHAW	
07584.pdf	RATCHETTING DUE TO DIFFFERENCE IN RESPONSE OF TUBEPLATE AND BOTTOM GRID	266/HW;DM 112		
07585.pdf	COMPUTER ANALYSIS OF PFR IHX GRID REFERENCE DESIGN	267/HW;DM 113		
07589.pdf	PFR-IHX REDESIGN. PFR-IHX DEVELOPEMENT.	TR 2013;262/HW		
07887.pdf	PFR PRIMARY CIRCUIT-DETERMINATION OF THE SODIUM TEMPERATURE AT THE INTERMEDIATE HEAT EXCHANGER TOP TUBE PLATES SUBSEQUENT TO A SIMULATED REACTOR TRIP USING A 1/5 SCALE WATER MODEL	ND-M-1083(R)	WINN WR;FRANCE J	1980
08378.pdf	PFR IHX 'MINIMUM CHANGE' DESIGN STUDY	TR 2057 ;SR 5495 R	MAPPLEDECK C;SHAW JB	1983
08381.pdf	PROTOTYPE FAST REACTOR,DOUNREAY INTERMEDIATE HEAT EXCHANGER STRAIN SURVEY ON No 6 TUBE BUNDLE	TN/R&D 40	LYONS GA	

Reference No.	Title	Location	Authors	Year
08386.pdf	A REVIEW OF EQUIPMENT NEEDED FOR REPAIR TO AN IHX PFR DOUNREAY	PFR/0C/P(83)62	WEBB J;HUNTER D	1983
09284.pdf	IHX CORROSION PROBLEMS FOLLOWING SODIUM/WATER REACTIONS	FRDCC/MWG/CSG/P(87)121;FRDCC/MWG/P(87)472	WALTERS J	1987
09408.pdf	NNC REVIEW OF PFR IHX DESIGN FOR JAPCO		LOMAS S	1987
09764.pdf	PFR IHX TOP: STRESS ANALYSIS UNDER TRANSIENT THERMAL LOADING CONDITIONS (C87/803)	PFR/TC/P(87)200;PFR/LLF/P(87)66	GILROY JE;RITSON DJ	1987
10388.pdf	SGU'S AND IHX'S FOR PFR - PROVISION IN VOTE FUNDS	FRJC/P(83)22	SMITH GEI	1983
11183.pdf	NNCs ASSESSMENT ON UKAEA DESIGN PROPOSAL FOR THE OPERATIONAL REACTOR CLOSURE PLUG - PFR IHX REMOVAL	ED 726	BOWKER L;JACKSON P;GAMBLE K;BARLOW D	1983
11211.pdf	PRELIMINARY COMMENTS ON THE UKAEA DUMMY IHX PLUG DESIGN PROPOSAL DRAWING OAE 475809 REFERS	ED 715		
13398.pdf	PFR IHX CORROSION ASSESSMENT FOR RUN SIXTEEN	PFR/TC/P(88)288;OETD/TC 1642	JONES K	
16048.pdf	PFR IHX CORROSION ASSESSMENT RUN 17 AUGUST 1988 TO APRIL 1989	PFR/TC/P(89)380;PIDG TMNO 21	JONES K	1989
17045.pdf	AN ANALYSIS OF THE FLOW AND TEMPRATURE DISTRIBUTIONS UNDERNEATH THE PFR-IHX BOTTOM TUBE-PLATE USING THE PHOENICS CODE	CFR/PTWG/P(83)23	BROWN GA;SCRIVEN J	1983
17048.pdf	THE VALIDATION OF THE ANTHEA COMPUTER CODE USING DATA FROM THE PFR-IHX	CFR/PTWG/P(83)19	BROWN GA	1983
17078.pdf	AN ANALYSIS OF THE FLOW AND TEMPRATURE DISTRIBUTIONS UNDERNEATH THE PFR/IHX BOTTOM TUBEPLATE USING THE PHOENICS CODE WHEN PFR IS OPERATING WITH ONE SECONDARY CIRCUIT	CFR/PTWG/P(83)30	BROWN GA	1983
18094.pdf	AN INVESTIGATION INTO THE EFFECTS OF IHX LEAKAGE FLOW DURING SINGLE CIRCUIT OPERATION OF PFR USING A 1/6TH SCALE WATER PERSPEX MODEL	PFR/SWP/P(83)43;FRD/TN/P(83)578	PURSLOW B	1983
18103.pdf	PFR ANNUAL SAFETY SUMMARY (MAY 1984)	PFR/SWP/P(84)25	HENDERSON JDC	
19942.pdf	PFR/IHX CORROSION ASSESSMENT RUNS 19,20 AND 21 AND AA REASSESSMENT OF PREVIOUS DATA	PE1/1807;PFR/TC/P(91)449;PIDD/TM(91)3	MORGAN DJ	1991
19945.pdf	PFR IHX CORROSION ASSESSMENT RUNS 22 AND 24	PE1/1806;PFR/TC/P(91)476;PIDD/TM(91)12	MORGAN DJ	1991
20384.pdf	PFR IHX'S DESIGN OF INSPECTION EQUIPMENT ADDITIONAL INSTRUMENTATION REPLACEMENT ROOF PLUGS & HANDLING FRAME SUMMARY PAPER	PFR/TF/P(80)438	BOWKER LJ;BARLOW D	1980

Reference No.	Title	Location	Authors	Year
20955.pdf	PFR IHX REMOVAL AND INSPECTION (PAPER NO 2)	PFR/TF/P(77)257	FRD&SRD	
21321.pdf	PFR REACTOR JACKET/IHX POD DAMAGE ASSESSMENT AS OF JUNE 1981	PFR/SWP/P(81)33	BROADLEY D	1981
21488.pdf	PRELIMINARY ANALYSIS OF DECAY HEAT REMOVAL COIL STRESSES	PFR/SWP/P(77)38;PFR/SDG/P(77)18;PFR/LFF/P(86)57	DURSTON JG	
21636.pdf	A REVIEW OF TEMPERATURE LIMITS FOR TWO CIRCUIT OPERATION OF PFR	PFR/SWP/P(80)17;PFR/LFF/P(85)44	EVANS AD;BROADLEY D	1980
21639.pdf	PFR IHX LEAKAGE TESTS APRIL AND MAY 1980	PFR/SWP/P(80)30	GALLIE P	1980
21640.pdf	PFR REACTOR JACKET - IHX LEAKAGE TEST	PFR/SWP/P(80)44;FRD/TN(80)404	PURSLOW B	1980
21641.pdf	THERMAL FATIGUE ENDURANCE ASSESSEMENT OF PFR-IHX POD-JACKET WELD	PFR/SWP/P(80)45	BELL RT;CORNWALL WS;LINNING DL	1980
21756.pdf	FATIGUE LIFE OF THE PFR IHX INLET/OUTLEY DUCT FLANGE	RTS/TAD/P(88)1766;SIC/260/P(88)15;PFR/TC/P(88)277	MICHIE D;	1988
21929.pdf	A SUMMARY OF THE POSITION ON THE ASSESSMENT OF POTENTIAL THERMAL STRIPING TO THE BOTTOM TUBEPLATE OF THE PFR IHX DURING SINGLE CIRCUIT OPERATION	PFR/SWP/P(84)47;FRD/TN(84)702	BROADLEY D	1984
22036.pdf	RATIONALISATION OF OPERATING RULES CONCERNING IHX OUTLET TEMPERATURES	PFR/SWP/P(86)49	HENDERSON JDC; McCRINDLE D	1986
22452.pdf	PFR SAFETY WORKING PARTY STRENGTH OF THE PFR/IHX	PFR/SWP/P(88)39	TOMINS B	1988
22472.pdf	ARK CALCULATIONS TO ESTABLISH THE EFFECTS OF MULTIPLE TUBE RUPTURE INCIDENTS ON THE IHX PRESSURE IN THE PFR CIRCUIT	PFR/SWP/P(90)23;PFR/SGSG/P(90)68	WRIGHT PJ	1989
23048.pdf	LEAK DETECTIONS AND DECONTAMINATION PROCESS OF IHX'S	PE4/5354;SCC/LGVE/91-16	SCHINDLER P	1991
23163.pdf	THE DEFECT TOLERANCE OF PFR IHX	PE1/4906;PFR/SIAG/P(92)99;	PICKER C;GREEN D;	1992
23164.pdf	THE CALCULATION OF CRITICAL CRACK LENGTH OF THE PFR IHX SHELL	PFR/SIAG/P(92)101;PE1/4907;	GREEN D;	1992
23569.pdf	THE SAFETY CASE FOR THE PFR IHX DEFECT TOLERANCE AND THE RESPONSE TO ITS PEER REVIEW	PE1/6085;RS/SWP/P(92)43;PE1/6378	HERRICK AR;	1992
23915.pdf	THE DEFECT TOLERANCE OF THE PFR IHXS	PE1/4900;RS/SWP/P(92)22;	HENDERSON JDC;PICKERC;	1992
24463.pdf	CONSIDERATION OF AXIAL AND CIRCUMFERENTIAL SURFACE DEFECTS IN THE PFR/IHX SHELL	PFR/SIAG/P(93)149	GREEN D;GREEN VR	1993

Reference No.	Title	Location	Authors	Year
24465.pdf	FOLLOW UP ITEMS CONCRNING THE SAFETY CASES FOR THE PFR IHX'S AND SECONDARY SODIUM CIRCUITS	PE1/7568;DNSC /P(93)166	HENDERSON JDC	1993
24734.pdf	STRESS ANALYSIS OF A SECTION OF THE PFR REACTOR JACKET INCLUDING AN IHX POD PHASE 2 REPORT ISSUE A	AAD/R27.10.80	ATKINE RESEARCH & DEVELOPMEN T	1980
24737.pdf	STRESS ANALYSIS OF A SECTION OF THE PFR REACTOR JACKET INCLUDING AN IHX POD PHASE 3 REPORT	AAD/R10.12.81	ATKINE RESEARCH & DEVELOPMEN T	1981
24754.pdf	STRESS ANALYSIS OF A SECTION OF THE PFR REACTOR JACKET INC;LUDING AN IHX POD	AAD/R9.11.77	ATKINE RESEARCH & DEVELOPMEN T	
24758.pdf	DRAFT THE MANAGEMENT FOR ANALYSIS OF COMPLEX REACTOR COMPONENTS		ATKINE RESEARCH & DEVELOPMEN T;PLATT R	
24891.pdf	A REVIEW OF THE REQUIREMENTS OF THE PFR REPLACEMENT IHX STRUCTURAL INTEGRITY ACTIVITIES	TN/P(84)672	GREEN D	1984
24974.pdf	DESIGN SUBSTANTIATION REPORT PROPTOTYPE FAST BREEDER DOUNREAY SPARE INTERMIDIATE HEAT EXCHANGER	FRD224/DSR/00 1	LOMAS S;MITCHELL CH;ROYDEN R;DEARDE G;GREEN D	1985
25153.pdf	THE PRESSURE STRENGTH OF THE SECONDARY CIRCUIT BELLOWS UNITS IN THE PFR IHX ASSEMBLIES	PFR/SWP/P(83)2 0	FOLEY J	
25177.pdf	RATE OF DAMAGE ACCUMULATION IN PFR IHX PODS DURING 2 AND 1 CIRCUIT OPERATION	PFR/LLF/P(85)38 ;PFR/SWP/P(81) 33	ROSE RT	1981
25181.pdf	THE PRESSURE STRENGTH OF THE SECONDARY CIRCUIT BELLOWS UNITS IN THE PFR IHX ASSEMBLIES	PFR/LLF/P(85)23 ;PFR/SWP/P(83) 20	FOLEY J	
25182.pdf	THE EFFECT OF PRESSURE PULSES ON THE INTEGRITY OF PARTIAL PENETRATION WELDS IN THE PFR IHX OUTER SHELL	PFR/LLF/P(85)22 ;PFR/SWP/P(83) 10;TN/P(83)588 ;	BUCHTHORPE DE	1983
25189.pdf	AN APPROXIMATE METHOD FOR ASSESSING SECONDARY CIRCUIT START UP CYCLIC DAMAGE TO THE PFR IHX TUBEPLATE	PFR/LLF/P(85)19 ;PFR/SWP/P(82) 14;TN/P(82)502 ;	GREEN D	1982
25197.pdf	DESIGN SUBSTATIATION REPORT PROPTOTYPE FAST BREEDER DOUREAY SPARE INTERMEDIATE HEAT EXCHANGER STRUCTURAL INTEGRITY ASSESSMENT OF THE THERMAL SYPHON COIL AND SUPPORT BRACKETS	PFR/LLF/P(85)25 ;FRD/224/DSR/0 13	GREEN D	1985
25202.pdf	DESIGN SUBSTANTIATION REPORT PROTOTYPE FAST BREEDER DOUNREAY SPARE INTERMEDIATE HEAT EXCHANGER STRUCTURAL INTEGRITY ASSESSMENT OF THE THERMAL SYPHON SLEEVE	PFR/LLF/P(85)26 ;FRD/224/DSR/0 14	GREEN D	1985
26021.pdf	THE IHX OUTLET TEMPERATURE TRIP PFR	PFR/TF/P(75)90	BROADLEY D	

Reference No.	Title	Location	Authors	Year
27075.pdf	PFR IHX TOP STRESS ANALYSIS UNDER TRANSIENT THERMAL LOADING CONDITIONS	TN/P(84)709	RITSON DJ;CLARK JS	1984
27464.pdf	EFFECT OF SECONDARY SODIUM CIRCUIT TRIP ON THE PFR IF THE REACTOR IS NOT TRIPPED BT THE IHX OUTLET TEMPERATURE THREMOCOUPLES	PFR/SUWP/15;PFR/TC/P(72)26	BROADLEY D;THOMASSON RK	
27521.pdf	CREEP DAMAGE IN THE BOTTOM TUBE PLATE OF PFR INTERMEDIATE HEAT EXCHANGERS DURING SECONDARY SODIUM PUMP TRIP TRANSIENTS	ND-M-385;PFR/IHX/P(78)16;DCWG/P(78)162	BELL RT	
27614.pdf	THE FEASIBILITY OF COOLING IHX LEAKAGE FLOW WITH DHR LOOPS OPERATING UNDER FORCED FLOW CONDITIONS	TN/P(78)268;PFR/TF/P(78)332	WILKES DJ	
28437.pdf	IHX REMOVAL STRATEGY	PFR/TF/P(81)499	HAYDEN O	1981
31120.pdf	PFR REPLACEMENT IHX PISTON RING SEALS	C5426/TR/010	BOOTH R	1997
31597.pdf	A STRUCTURAL ASSESSMENT OF A PFR IHX SCALE MODEL TEST VESSEL	PFR/SIAG/P(91)52	DANIELS BD	1991
31598.pdf	PFR IHX 0.4 SCALE MODEL PRESSURE TESTS	PFR/SIAG/P(91)53	HARRISON M;DAVENPORTF	1991
31637.pdf	THE DEFECT TOLERANCE OF PFR IHX	PFR/SIAG/P(92)99	PICKER C;GREEN D	1992
31638.pdf	DESIGN AND MANUFACTURING STANDARDS FOR PFR INTERMEDIATE HEAT EXCHANGERS (IHX)	PFR/SIAG/P(92)100	MAIR A	1992
31639.pdf	THE CALCULATION OF CRITICAL CRACK LENGTH OF THE PFR IHX SHELL	PFR/SIAG/P(92)101	GREEN D	1992
31658.pdf	MINIMUM FRACTURE TOUGHNESS REQUIREMENT OF THE PFR IHX SHELL TO RESIST THROUGH WALL CRACK GROWTH OF A 4mm DEEP x200mmLONG SURFACE CRACK	PFR/SIAG/P(92)127	DANIELS BD	1992
31675.pdf	ESTIMATION OF CONDITIONAL FAILURE PROBABILITIES OF PFR IHX'S AND SECONDARY CIRCUIT PIPEWORK	PFR/SIAG/P(92)145	GREEN VR LIDBURY DPG	1992
31686.pdf	PFR IHX POD DAMAGE ASSESSMENT HEAT EXCHANGERS	PFR/SIAG/P(93)154	DANIELS BD	1993
	RATE OF DAMAGE ACCUMULATION IN PFR IHX PODS DURING 2 AND 1 CIRCUIT OPERATION	PFR/LLF/P(85)38;PFR/SWP/P(81)33	ROSE RT	1981

Table 6: Wood Report List: Pumps

Reference No.	Title	Location	Authors	Year
00050.pdf	A STUDY OF THE CONSEQUENCES OF AN OIL LEAK FROM THE MECHANICAL PUMP ON THE RISLEY HIGH TEMPERATURE SODIUM LOOP	CEWP/P(83)356	JONES DG	1983
07252.pdf	PROPOSAL TO CARRY OUT CAVITATION TESTS ON A PFR SECONDARY SODIUM PUMP	FREWG/P(86)287	PREECE	1986
07513.pdf	CAVITATION TESTS ON A PFR SECONDARY SODIUM PUMP	PFR/TC/P(86)106	PREECE	1986
10428.pdf	CAVITATION TESTS ON A PFR SECONDARY SODIUM PUMP	FREWG/P(88)387	CROAD A	1988
16046.pdf	INVESTIGATION OF STEP CHANGE IN CORE FLOW RELATED TO STEP CHANGE IN PRIMARY PUMP MOTOR CURRENT	PFR/TN/P(90)962	MCCRINDLE D;HILL R	
16047.pdf	INVESTIGATION INTO A SECOND STEP CHANGE IN REACTOR CORE FLOW RELATED TO A STEP CHANGE IN PRIMARY PUMP MOTOR CURRENT	PFR TECH NOTE NO(90)970	MCCRINDLE D	1990
19668.pdf	PRIMARY PUMP 3 PERFORMANCE RUN 22.	PFR/TECH/N1017;PE4/1339;	MCCRINDLE D;	1991
21210.pdf	A REVIEW OF OIL INGRESS INTO THE PFR PRIMARY CURCUIT FROM THE PRIMARY PUMPS		HARTLEY DJ;PURSLOW B;BRYANT S	1992
21220.pdf	MANAGEMENT OF THE PRIMARY SODIUM PUMP UPPER BEARING LUBRICATING OIL SYSTEMS	PE1/3234;PFR/TN 1059;PE1/3615	SHIPLEY DF	1992
21267.pdf	PFR PRIMARY CIRCUIT FLOW DURING PRIMARY PUMP VALVE ASSEMBLY REMOVAL	RS/SWP(92)5;PE1/3326;	MCCRINDLE D;CRUICKSHANK A;	1992
21803.pdf	MANAGEMENT OF THE PRIMARY SODIUM PUMP SUPPER BEARING LUBRICATING OIL SYSTEMS	PE1/3716;1059;	SHIPLEY DF;	1992
22246.pdf	INSPECTION OF PRIMARY PUMP VALVE NO 3 FULTERS	PE1/3790;PFR TECH NOTE 1070;	SANDISON A;	1992
22247.pdf	PROPOSAL FOR IRRADIATION OF PFR PUMP OIL SAMPLES IN CONJUNCTION WITH SODIUM AT TEMPERATURE IN THE IFC	PE1/3785;IPSC/P(92)10;	HIGGINSON PR;	1992
22248.pdf	PROPOSALS TO STUDY THE COMBINED EFFECT OF RADIATION AND TEMPERATURE ON THE DEGRADATION OF OIL	PE1/3784;AEMD/P1026;	BURNAY SG;	1992
23047.pdf	ASSESSMENT OF PRIMARY SODIUM PUMP UPPR BEARING LUBRICATION OIL SYSTEM AND PUMP TANK COVER GAS MODIFICATION	PE2/5400;PFR/TN/N 1074	SHIPLEY DF	1992

Reference No.	Title	Location	Authors	Year
23166.pdf	ASSESSMENT OF CARBURIZATION FOLLOWING AN OIL LEAK IN THE PFR PRIMARY CIRCUIT	PFR/SIAG/P(92)118;PE1/4905;PR/MWG/P(92)814;MSD/26/0031;PE1/5684;	THORLEY AW;SKELDON P;PICKER C;HAMER AN;	1992
23179.pdf	PROJECT INSTRUCTION OR PFR PRIMARY SODIUM PUMP NO 2 WORK FOR AEA TECHNOLOGY	C8583/PI	MADDOCKS CD	1992
23203.pdf	THE PFR PRIMARY SODIUM PUMP OIL MANAGEMENT SYSTEM, HISTORY AND STRATEGY FOR THE FUTURE	PE1/4962;UN 150 810 R	SHIPLEY D F	1992
23398.pdf	THE PFR PRIMARY SODIUM PUMP OIL MANAGEMENT SYSTEM HISTORY AND STRATEGY FOR THE FUTURE	PE1/5555;UN 150810R	SHIPLEY DF	1992
23527.pdf	EXAMINATION OF PRIMARY PUMP VALVE FILTERS AFTER REMOVAL FROM PFR	PE1/5924;UN150906R;pe15969	MCKIDDIE R;MUNRO B;PUNNI JS;GOWER SM;HIGGINSON PR;	1992
23739.pdf	PEER REVIEW OF UN150810R THE PFR PRIMARY SODIUM PUMP OIL MANAGEMENT SYSTEM, HISTORY AND STRATEGY FOR THE FUTURE	SDG/S079999/92;	GREGORY CV;	1992
23818.pdf	BURS TEST ON THE FILTER MESH USED IN THE PRIMARY PUMP VALVE	FMD/D(92)112;PE1/6678;	WALTON A;COLLINSON AE;	1992
23894.pdf	BURST TEST ON THE FILTER MESH USED IN THE PRIMARY PUMP VALVE	PE1/6891;PE1/6863;UN150955R;	WALTON A;COLLINSON AE;	1992
23895.pdf	OBSERVATIONS ON THE DEPOSITS FOUND ON THE CLADDING SURFACE OF PINS FROM SUB ASSEMBLIES MAG SNF DKR (IN SUPPORT OF TE OIL INGRESS SAFETY CASE	PE1/6890;UN 151025N;PE1/6862;	GREGORY CV;	1992
23941.pdf	REPLACEMENT FILTERS PRIMARY PUMP VALVES PFR	AEAE/TN(92)562;PE1/6958;PE1/6996;	WARD T;	1992
25283.pdf	PFR SECONDARY PUMP POST-TRIP RUNDOWN CHARACTERISTICS	CFR/CDWG/P(80)82	BUTTERFIED MH;WOFFINDEN J	1980
25285.pdf	PFR SECONDARY SODIUM PUMP RUNDOWN CHARACTERISCTICS FURTHER COMPARISON WITH INCIDENT RECORDER DATA	CFR/CDWG/P(82)84	BUTTERFIED MH	1982

Reference No.	Title	Location	Authors	Year
25428.pdf	EMPA EXAMINATION OF FILTER MESH SAMPLES FROM PRIMARY PUMP VALVES AND FUEL/BREEDER SUB-ASSEMBLIES	FMPM/P(93)6;CFWG /P(93)11;PE4/7706	PUNNI JS	1993
26661.pdf	RAISING OF SECONDARY SODIUM PUMP SPEED RESTIRCTIONS (PFR SAM 467)	PE1/7792;RS/SWP/ P(93)21	HENDER SON JDC	1993
26768.pdf	ASSESSMENT OF PRIMARY SODIUM PUMP UPPER BEARING LUBRICATION OIL SYSTEM AND PUMP TANK COVER GAS MODIFICATIONS	PFR/TN 1074;PE2/5400	SHIPLEY DF	1992
27408.pdf	PFR OPERATIONS CASCADE TRIPPING OF SECONDARY SODIUM PUMPS AND THE REACTOR	PFR/SUWP/46	BAINBRI DGE A;CANEY V	
28020.pdf	SAFETY REPORT FOR THE HARWELL CARBON METER	PFR/SWP/ESC/P(74) 9	HENDER SON JDC	
28087.pdf	CALIBRATION OF PUMP PRESSURE MEASUREMENT INSTRUMENTATION	PFR/OC/P(73)8;PFR /CP/12	HILL EA;DUN COMBE E	
30598.pdf	COMPARISON BETWEEN EFR & PFR MAIN SODIUM PUMPS & INTERMEDIATE HEAT EXCHANGERS	C9809/TR/022	FIRTH GF	1995
31651.pdf	ASSESSMENT OF CARBURIZATION FOLLOWING AN OIL LEAK IN THE PFR PRIMARY CIRCUIT	PFR/SIAG/P(92)118	THORLE Y AW;SKE LDON PPICKER C;HAME R AN	1992
	WATER MODELLING OF PFR PSP CONE OVERFLOW SYSTEM WITH AND WITHOUT OIL INGRESS	UN 150863R;PE1/5088	COLLINS ON A E;WALT ON A	1992

Table 7: Wood Report List Primary Vessel

Reference No.	Title	Location	Authors	Year
03957.pdf	ASSESSMENT OF THE CONSEQUENCES OF PRIMARY TANK FAILURE	R.S.W/P70/21		
15912.pdf	SPECIFICATION FOR THE INSPECTION OF THE PFR ROOF COOLING/GAS BLANKET LUTE SYSTEM		SMALL J	1990
18084.pdf	LEAK-BEFORE-BREAK PROCEDURE FOR SODIUM BOUNDARY COMPONENTS DCRC CONCLUDING REPORT DRAFT REPORT - REVISION 0	FR/SIWG/CBG/P(9 1)162	HOOTON DG	1991
18095.pdf	PFR OUTER POOL TEMPERATURE EXCURSION 2ND FEBRUARY 1981 COLD POOL MODEL TESTS AND TOP STRAKE DAMAGE ASSESSMENT	PFR/SWP/P(82)3;F RD/TN/P(82)498	PURSLOW B;DIXON M	1982
18096.pdf	PFR PRIMARY TANK TEMPERATURE AND DAMAGE ASSESSMENT POST TRIP 19TH MARCH 1981	PFR/SWP/P(82)8;F RD/TN/P(82)501	DIXON M	1982
18097.pdf	THE RELEVANCE OF THE SACLAY SODIUM LEVEL EXPERIMENT TO PFR TOP STRAKE	PFR/SWP/P(82)32; FRD/TN/P(82)535	HOOTON DG;ROSE RT	1982
18098.pdf	ADDENDUM TO PFR SAFETY REPORT PRIMARY VESSEL TOP STRAKE	PFR/SWP/P(81)6;F RD/TN/P(81)438;P FR/LLF/P(85)7	DURSTON JG;ROSE RT	1981
18100.pdf	COVER NOTE TO LIVERPOOL UNIVERSITY REPORT ON THERMAL BUCKLING OF THE PFR TOP STRAKE	PFR/SWP/P(81)40	ROSE RT	1981
18102.pdf	COVER NOTE TO REPORT ON THE SACLAY SODIUM LEVEL EXPERIMENT	PFR/SWP/P(81)39	ROSE RT	1981
21468.pdf	ADDITIONAL THEORETICAL RESULTS FOR PFR TOP STRAKE INSULATION AT FULL POWER INCLUDING EFFECTS OF COOLING FLOW DIVERTOR VALVES	PFR/SWP/P(76)28; PFR/TSPC/P(76)48	DIXON M;HOOTO N DG;GREE N D	
21469.pdf	PFR PRIMARY VESSEL TOP STROKE TEMPERATURE PROFILES	PFR/SWP/P(76)15; PFR/PCR/P(76)4	DURSTON JG	
21470.pdf	PFR PRIMARY VESSEL TOP STRAKE STRUCTURAL INTEGRITY UNDER THERMAL STRESSING	PFR/SWP/P(76)16; PFR/PCR/P)76)5	ROSE RT;WOOD DS	
21471.pdf	DEPENDANCE OF PFR TOP STRAKE TEMPERATURE PROFILE ON ROOF COOLING FLOW	PFR/SWP/P(76)33; PFR/TSPC/P(76)47	DURSTON JG	
21472.pdf	PFR ROOF COOLING SYSTEM ROTATING SHIELD PENETRATION FLOW REVERSAL TESTS	PFR/PCR/P(76)38; PFR/SWP/P(76)29	LUNT AR	
21474.pdf	A REVIEW OF STRUCTURAL VALIDATION FOR THE PFR PRIMARY VESSEL TOP STRAKE FOLLOWING U.S. DISCUSSIONS	PFR/SWP/P(76)40; PFR/TF/P(76)169;P FR/PCR/P(76)44	ROSE RT	
21476.pdf	PFR PRIMARY VESSEL TOP STRAKE REPORT BY PROF. F.A. LECKIE	PFR/SWP/P(76)42	ROSE RT	
21477.pdf	TOP STRAKE DT-PFR	PFR/TSPC/P(76)53 ;PFR/SWP/P(76)44	BROADLE Y D	
22418.pdf	REVIEW OF PFR SAFETY AT AUGUST 1989	PFR/SWP/P(89)9	WASHING TON ABG	1989
25135.pdf	REVISED DAMAGE ASSESSMENT CURVES FOR PFR TOP STRAKE	PFR/SWP/P(78)27; TN/P(78)243	HOOTON DG	

Reference No.	Title	Location	Authors	Year
25361.pdf	FAILURE MODES OF THE PFR LEAK JACKET FOLLOWING A SODIUM LEAK FROM THE PRIMARY TANK	SIC/226/P(87)12;TAD/RTS/P(87)1703;PFR/LLF/P(87)70;	GREEN D	1987
25704.pdf	EXPLORATORY STUDY OF POSSIBLE EFFECTS ON CIRCUMFERENTIAL TEMPERATURE VARIATION ON THE PFR PRIMARY VESSEL TOP STRAKE	PFR/SWP/P(77)24	OSE RT	
25705.pdf	PFR PRIMARY TANK TOP STRAKE DAMAGE COMPARISON OF TANK DAMAGE IN THE AVERAGE OF THE WORST 3 SECTORS WITH THE WORST SECTOR	PFR/SWP/P(77)3	BROOMFIELD AM;HICKS JL	
25706.pdf	A SAFETY CASE FOR THE CONTINUED OPERATION OF PFR IN VEIW OF THE PRIMARY TANK TOP STRAKE TEMPERATURE GRADIENT (PLUS NOTES AND CALCULATIONS)	PFRSWP/P976)17;PFR/TF/P(76)128	BROADLEY D;ROSE RT	
25801.pdf	PFR PRIMARY VESSEL TOP STRAKE CHANGES IN THERMAL CHARACTERISTICS	TN/P(77)221;PFR/OPS/N473	HUMPHRIES J;DURSTON JG	
25802.pdf	A REVIEW OF PFR TOP STRAKE DAMAGE ASSESSMENT NOVEMBER 1977	PFR/SWP/P(77)70	HOOTON DG;ROSE RT	
25808.pdf	REVISED DAMAGE ASSESSMENT CURVES FOR PFR TOP STRAKE	PFR/SWP/P(78)27;TN/P(78)243	HOOTON DG	
25809.pdf	PFR PRIMARY VESSEL TOP STRAKE TELL TALE REFERENCE READINGS	PFR/SWP/P(77)34	HOOTON DG	
25858.pdf	A REVISED METHOD FOR PFR PV TOP STRAKE DAMAGE ASSESSMENT	PFR/SWP/P(78)4;TN/P(78)227	BROADLEY D;DURSTON JG;HOOTON DG	
25873.pdf	PFR PRIMARY VESSEL POGO STICK READINGS	PFR/SWP/P(80)61;TN/P(80)426	DURSTON JG	1981
25876.pdf	TOP STRAKE INSTRUMENTATION	PFR/OPS/N727	HUMPHRIES J	1983
25877.pdf	ADDENDUM TO PFR SAFTY REPORT PRIMARY VESSEL TOP STRAKE	PFR/SWP/P(81)6;TN/P(81)438	DURSTON JG;ROSE RT	1981
25935.pdf	A RE ASSESSMENT OF TOP STRAKE DAMAGE ACCUMULATION PRIOR TO SODIUM LEVEL FIXING ON 11/3/76	PFR/SWP/P(78)72	HOOTON DG	
27410.pdf	PFR PLANT OPERATING CONSTRAINTS	PFR/SUWP/44	BAINBRIDGE A;CANEY V	
27565.pdf	A CASE FOR THE PROCUREMENT AND INSTALLATION OF GAS DIVERTER VALVES IN THE ARGON ROOF COOLING SYSTEM OF PFR TO REDUCE THE TOP STRAKE TEMPERATURE GRADIENT	PFR/PCR/P(76)36	BROADLEY D;DURSTON JG;SEED G;COOKE B	

Reference No.	Title	Location	Authors	Year
27567.pdf	PFR PRIMARY CIRCUIT THERMAL STRESSES	PFR/PCR/P(76)2	DAYLOR D;HARDI NGHAM RP;	
27575.pdf	A SAFETY CASE FOR THE CONTINUED OPERATION OF PFR IN VIEW OF THE PRIMARY TANK TOP STRAKE TEMPERATURE GRADIENT	PFR/SWP/P(76)17; PFR/PCR/P(76)28	BROADLE Y D	
27575.pdf	A SAFETY CASE FOR THE CONTINUED OPERATION OF PFR IN VIEW OF THE PRIMARY TANK TOP STRAKE TEMPERATURE GRADIENT	PFR/SWP/P(76)17; PFR/PCR/P(76)28	BROADLE Y D	
27588.pdf	CHANGES TO THE PFR ROOF COOLING SYSTEM OPERATION TO ALLOW HIGHER CORE OUTLET TEMPERATURE AND TO AMELIORATE TOP STRAKE THERMAL GRADIENTS	PFR/SWP/P(76)27; PFR/PCR/P(76)27	LUNT AR	
27589.pdf	ENDURANCE OF PFR ABOVE CORE STRUCTURE RELEVANCE OF DFR MK2 SUPPORT STOOL	PFR/SWP/P(78)33; TN/P(78)248	ROSE RT	
27591.pdf	ASSESSMENT OF DAMAGE TO PFR PRIMARY VESSEL TOP STRAKE	PFR/SWP/P(76)46	BROAWN C;MALCO LM PN;HICKS JL	
27593.pdf	EXTERNAL PRESSURE BUCKLING OF PFR PRIMARY TANK STAGE 2	PFR/SWP/P(76)49	GALLETLY GD;AYLW ARD RW	
27596.pdf	SUMMARY OF RESULTS FROM CTS/HARWELL EXPERIMENTS ON THIN CYLINDERS SUBJECTED TO AXIALLY MOVING TEMPERATURE RAMPS	PFR/SWP/P(77)71	BELL RT	
27604.pdf	PFR TOP STRAKE ADDITIONAL INSTRUMENTATION	PFR/TS/P(76)167	LAYCOCK W	
27612.pdf	SPECIFICATION OF STRESS AND STABILITY CALCULATIONS FOR PFR PRIMARY VESSEL	TN/P(78)255	ROSE RT	
27627.pdf	PFR PRIMARY VESSEL TENTATIVE SPECIFICATION FOR AN EXPERIMENTAL INVESTIGATION OF TOP STRAKE THERMAL STRESSES	TSPC/P(76)24	ROSE RT	
27634.pdf	STRESS MODEL FOR THE PFR PRIMARY TANK	TSPC/P(76)29	TAYLOR D	
27635.pdf	PFR PRIMARY VESSEL TOP STRAKE ANALYSIS OF THERMAL STRESS PERFORMANCE ATTAINABLE BY 1/10TH SCALE MODEL TANK	TSPC/P(76)35	BUCKLEY FI;FEWST ER J;KNOWL ES PJ	
27638.pdf	ADDITIONAL STEADY STATE RESULTS OBTAINED FROM PFR SODIUM LEVEL CHANGE EXPERIMENT 6 - 11 APRIL 1976 (ADDENDUM 1)	TSPC/P(76)22	DURSTON JG;GRAH AM D;HUMPH RIES J;LOMAS J;SARGE NT	

Reference No.	Title	Location	Authors	Year
27639.pdf	PFR TEMPERATURE PROFILES DURING SODIUM FILL	TSPC/P(76)33	DURSTON JG;WORT H B	
27640.pdf	PFR VESSEL THERMAL STRAINS CAUSED BY MOVING AXIAL TEMPERATURE RAMP PROGRAMME OF TESTS ON SMALL SCALE CYLINDERS AT HARWELL (JULY/AUGUST 1976)	TSPC/P(76)44	LINNING DL	
27641.pdf	PFR PRIMARY VESSEL TOP STRAKE TRANSIENT TEMPERATURE PROFILES	PFR/CTS/TS(1)	BELL RT	
27641.pdf	PFR PRIMARY VESSEL TOP STRAKE TRANSIENT TEMPERATURE PROFILES	PFR/CTS/TS(1)	BELL RT	
27642.pdf	PFR PRIMARY VESSEL TOP STRAKE DEPENDENCE OF TRANSIENT TEMPERATURE PROFILES ON SODIUM AND GAS HEAT TRANSFER COEFFICIENTS	PFR/CTS/TS(2)	BELL RT	
27643.pdf	PFR PRIMARY VESSEL THERMAL STRESSES MODEL EXPERIMENTAL WORK AT HARWELL INITIAL TEST RESULTS	PFR/CTS/TS(4)	LOVE JB	
27644.pdf	PFR PRIMARY VESSEL THERMAL STRESSES CAUSED BY MOVING AXIAL TEMPERATURE RAMPS PRELIMINARY PROGRAMME OF TESTS ON SMALL SCALE MODEL CYCLINDERS	PFR/CTS/TS(5)	LINNING DL	
27645.pdf	COMPARISON OF EXPERIMENTAL AND THEORETICAL TRANSIENT TEMPERATURES IN THE PFR PRIMARY VESSEL TOP STRAKE	PFR/CTS/TS(6)	BELL RT	
28493.pdf	A STUDY OF THE PFR PRIMARY VESSEL TOP STRAKE TEMPERATURE DISTRIBUTION	TSPC/P(76)37	LUNT AR;MCSW EENEY RN	
28501.pdf	A COMPARISON OF PEDICTED AND MEASURED DIAMETRAL GROWTH DUE TO TEMPERATURE RAMPS WHICH MOVE AXIALLY	TSPC/P(76)46	JOBSON DA	
28503.pdf	AN EXAMINATION OF THE FACTORS INVOLVED IN THE RADIAL RATCHETTING OF A CYLINDER SUBJECTED TO TRAVELLING AXIAL TEMPERATURE RAMP	TSPC/P(76)50	LINNING DL	
28504.pdf	RADIAL RATCHETTING OF A CYLINDER SUBJECTED TO A MOVING AXIAL TEMPERATURE RAMP	TSPC/P(76)51	LINNING DL	
28505.pdf	JET INPRINGEMENT COOLING FOR THE PFR TOP STRAKE 1/10TH SCALE MODEL	TSPC/P(76)52	FEWSTER J	
31174.pdf	STRESS ANALYSIS OF THE PFR PRIMARY TANK AND LEAK JACKET WITH CONSIDERATION AND DEFECT TOLEANCE ON THE BASIS OF TEARING INITIATION AND COVER GAS PRESSURE FOR BUCKLING	PFR/SIAG/P(92)10 2	GREEN D	1993

Table 8: Wood Report List: Roof and Roof Ceiling

Reference No.	Title	Location	Authors
00726.pdf	PFR ROOF ROTATING SHIELD COMPARISON OF PENETRATION THERMOCOUPLE READINGS WITH COMPUTER CALCULATED TEMPERATURE DISTRIBUTION	DM/P(83)447	WALE RJ
03910.pdf	STRESS DISTRIBUTION IN THE PFR VAULT ROOF STEELWORK, MODEL INVESTIGATION USING PHOTOELASTICITY AND STRAIN GAUGES	PFR/SWP/P(70 (4;FPR 3/70	CLABBURN EJ
03911.pdf	AN EXAMINATION OF THE STRESS DISTRIBUTION AT THE CONNECTION OF THE PFR VESSEL SKIRT TO THE ROOF GIRDER	GRO/44/86/37	CLABBURN EJ
03912.pdf	AN EXAMINATION OF STRESS LEVELS OF A PFR ROOF MODEL USING STRAIN GAUGES	PROGRESS REPORT NO.1/69	CLABBURN EJ
03914.pdf	A MODEL INVESTIGATION OF THE STRESS DISTRIBUTION IN THE PFR VAULT ROOF TEST 1 THE STEELWORK UNDER DEAD WEIGHT LOADING	FPR 1/72	CLABBURN EJ
03915.pdf	A MODEL INVESTIGATION OF THE STRESS DISTRIBUTION IN THE PFR VAULT ROOF TEST 1 THE STEELWORK UNDER THE INCIDENT LOAD CONDITION	FPR 3/71	CLABBURN EJ
03917.pdf	A MODEL INVESTIGATION OF THE STRESS DISTRIBUTION IN THE PFR VAULT ROOF TEST 3 THE STEELWORK LINE - LOADED AT THE TUNING FORK POSITION	FPR 8/72	CLABBURN EJ
11136.pdf	CORROSION OF PFR ROOF MATERIALS	TRG MEMO 7095 (R); FRDCC/MWG/ CSG/P(88)150 ;FRDCC/MWG/ P(88)587	GERRARD JF
15912.pdf	SPECIFICATION FOR THE INSPECTION OF THE PFR ROOF COOLING/GAS BLANKET LUTE SYSTEM		SMALL J
15913.pdf	SPECIFICATION FOR THE INSPECTION OF THE PFR VAULT ROOF AND ROTATING SHIELD TO THE MANTENANCE INSPECTION AND TEST SCHEDULE		
21319.pdf	ADDENDUM TO PFR SAFETY REPORT SUB-SECTION H5 REVISED VAULT ROOF COOLING SYSTEM	PFR/SWP/P(74)62	GLASS JAF;BROADLE Y D
21471.pdf	DEPENDANCE OF PFR TOP STRAKE TEMPERATURE PROFILE ON ROOF COOLING FLOW	PFR/SWP/P(76)33;PFR/TSPC /P(76)47	DURSTON JG
21472.pdf	PFR ROOF COOLING SYSTEM ROTATING SHIELD PENETRATION FLOW REVERSAL TESTS	PFR/PCR/P(76)38;PFR/SWP/ P(76)29	LUNT AR
25178.pdf	EXTRACTS FROM THE PFR DESIGN SAFETY REPORT RELEVANT TO LAMELLAR TEARS IN THE ROOF STRUCTURE	PFR/LLF/P(85) 35	
26604.pdf	FOULNESS SAFEX COMMITTEE STATIC TSTS ON 1/16TH SCALE MODEL OF THE PFR VAULT ROOF	SAFEX/P68	WOOD AJ;CARTER HM
27565.pdf	A CASE FOR THE PROCUREMENT AND INSTALLATION OF GAS DIVERTER VALVES IN THE ARGON ROOF COOLING SYSTEM OF PFR TO REDUCE THE TOP STRAKE TEMPERATURE GRADIENT	PFR/PCR/P(76)36	BROADLEY D;DURSTON JG;SEED G;COOKE B

Reference No.	Title	Location	Authors
27588.pdf	CHANGES TO THE PFR ROOF COOLING SYSTEM OPERATION TO ALLOW HIGHER CORE OUTLET TEMPERATURE AND TO AMELIORATE TOP STRAKE THERMAL GRADIENTS	PFR/SWP/P(76)27;PFR/PCR/P(76)27	LUNT AR
27647.pdf	OPERATIONAL ASPECTS PERTAINING TO PFR ROOF COOLING PROBLEMS	PFR/OC/P(74)3	LUNT AR
27648.pdf	PFR FAULT ROOF COOLING AN ASSESSMENT OF THE EXISTING SYSTEM AND PROPOSALS FOR A NEW SYSTEM	PFR/OC/P(73)32	LUNT AR
28487.pdf	PFR ROOF COOLING SYSTEM FLOW CHARCTERISTICS	TSPC/P(76)27	LUNT AR
29432.pdf	PFR ROOF COOLING AN ASSESSMENT ON THE FEASIBILITY OF CHANGING THE COOLING GAS FROM ARGON TO NITROGEN AND THE PROBABLE REDUCTION IN RUNNING COSTS	TN/P(82)577	BEECH DJ

Table 9: Wood Report List: Absorber Rod & Mechanisms

Reference No.	Title	Location	Authors	Year
00181.pdf	VOID SWELLING IN THE NIMONIC PE16 CONTROL ROD SEG FROM PFR	DFMC(83)P2;CPN/740	FULTON EJ;SINCLAIR WDJ	1983
00495b.pdf	THE PFR Mk2 AND Mk3 CONTROL RODS: LIMITATION OF PIN LIFETIMES BY AXIAL INTERACTION BETWEEN THE PELLET STACK AND CLADDING	NDM-2274; MWP/P(83)1152;FRASG/P(83)202	MOTTERS HEAD D	1983
00513.pdf	AN ESTIMATE OF THE TERMINAL VELOCITY OF A PFR Mk IIIa CONTROL ROD DROPPED FROM THE CHARGE MACHINE	NDM-2516; DRAFT;PFR/FEDWP/P(83)0965	PARDY A;TAYLOR AF	1983
00543.pdf	PROGRAMME FOR THE POST-IRRADIATION EXAMINATION OF PFR CONTROL ROD SEG (BORON CARBIDE0	NDM-1349; MWP/P(80)1116;FRASG/P(80)166	GILCHRIST KE	1980
00567.pdf	THE DESIGN OF A SHORTENED EXTENSION ROD FOR USE WITH A PFR Mk IV CONTROL ROD IN VIBRATION EXPERIMENTS	NDR-716; PFR/FEDWP/P(81)0833	DUTHIE JC	1982
00778.pdf	THE EXPERIMENTAL DETERMINATION OF THE VIBRATION MODES OF A PFR Mk IV CONTROL ROD WITH A DYNAMICALLY SIMULATED EXTENSION ROD	NDR-827;PFR/FEDWP/P(83)0951	WOLSTENHOLME JFR	1984
01807.pdf	PROPOSED MODIFICATIONS TO THE PFR MKIV CONTROL RODS AND GUIDE TUBES	FROC/P(84)124	BROWNE JJ;FORD J	1984
02755.pdf	THE ACCURACY OF CALCULATIONS OF PFR CONTROL AND SHUT-OFF ROD WORTHS USING THE FD4 DATA	RPWP/P(69)4	COLLINS PJ;BAKER AR;ALLON JR	
02756.pdf	PFR CONTROL AND SHUT-OFF RODS	RPWP/P(69)5	SANDISON A;SMITH JC	
02758.pdf	THE ACCURACY OF FD4 CALCULATIONS OF REACTION RATES WITHIN TANTALUM AND BORON CONTROL RODS	RPWP/P(69)29	COLLINS PJ;ALLEN JR	
02980.pdf	A PRELIMINARY ANALYSIS OF A HYPTHETICAL CONTROL ROD RUNAWAY ACCIDENT IN PFR USING THE FRAX 2 COMPUTER CODE	NDM-421;FREYWG/P(78)187	NEWTON TD	
03037.pdf	THE MANUFACTURE AND INSPECTION OF PE16 PFR MK 4 CONTROL ROD WRAPPERS	FR/FMDC/P(85)66;NDM-2928	PARRY P	
03111.pdf	THE MOZART CONTROL ROD EXPERIMENTS AND THEIR INTERPRETATIONS	FRCMWP/P(73)99	BROOMFIELD AM;COLLINS PJ;CARTER MD;MARSHALL J	

Reference No.	Title	Location	Authors	Year
03112.pdf	THE ANALYSIS OF THE ZEBRA 12 CONTROL ROD EXPERIMENTS AND PROPOSED METHODS OF CALCULATING PFR AND CFR CONTROL ROD WORTH	FRCMWP/P(73)100;CPWP/P(73)225	BROOMFIELD AM;COLLINS PJ;CARTER MD;MARSHALL J	
03127.pdf	THE ANALYSIS OF THE SODIUM REMOVAL EXPERIMENTS IN ZEBRA ASSEMBLY 13 - PART II SINGLE SUB-ASSEMBLIES ADJACENT TO A FULLY-INSERTED CONTROL ROD AND TO AN INNER CORE BREEDER SUB-ASSEMBLY	FRCMWG/P(74)150	HARDIMAN JP	
03236.pdf	A METHOD FOR OBTAINING FAST REACTOR CONTROL ROD CROSS SECTIONS AS A COSMOS TASK	FRCMWG/P(78)208	AGAR CWJL	
03637.pdf	AN EXAMINATION OF THE ABILITY OF THE PFR CONTROL AND SHUT-OFF RODS TO MEET THE SAFETY CRITERIA	TC/P(69)42	HENDERSON JDC	
03639.pdf	DIFFERENTIAL EXPANSION BETWEEN PFR CORE AND CONTROL SUPPORTS	TC/P(69)7;TASD/FRPG/P(69)7	MATTHEWS JD	
03680.pdf	SODIUM AEROSOL DEPOSITION ON CONTROL ROD AND SHUT OFF ROD MAGNET FACES	OC/P(85)207;PFR/OPS/N828	MASON L	1985
03788.pdf	CALIBRATION OF PFR CONTROL RODS AT START OF RUN 4	PFR EXPERIMENTAL RESULTS SHEET NO.95	CROWE DS;SUTHERLAND AJ	
03791.pdf	SUB CRITICAL MONITORING DURING RELOAD 3 IN PFR	PFR EXPERIMENTAL RESULTS SHEET NO.97A;OETD.TECH NOTE NO.46	CROWE DS	
03795.pdf	CALIBRATION OF PFR CONTROL RODS AT END OF RUN 4	PFR EXPERIMENTAL RESULTS SHEET NO.104;OETD.TECH NOTE NO.181	CROWE DS;LORD DJ	1980
03796.pdf	CALIBRATION OF PFR ABSORBER RODS AT START OF RUN 5	PFR EXPERIMENTAL RESULTS SHEET NO.106;OETD.TECH NOTE NO.245	CROWE DS;LORD DJ	1980
03799.pdf	CHECK ON CONTROL ROD CURTAIN WORTH DURING RUN 5 OF PFR	PFR EXPERIMENTAL RESULTS SHEET NO.109;OETD.TECH NOTE NO.325	CROWE DS;LORD DJ	1980
03801.pdf	CONTROL ROD WORTHS AT CURTAIN HEIGHT OF 581MM DURING RUN 5 OF PFR	PFR EXPERIMENTAL RESULTS SHEET NO.111;OETD.TECH NOTE NO.393	SUTHERLAND AJ;CROWE DS	1981

Reference No.	Title	Location	Authors	Year
03802.pdf	CONTROL ROD WORTHS AT CURTAIN HEIGHT OF 652MM DURING RUN 5 OF PFR	PFR EXPERIMENTAL RESULTS SHEET NO.112;OETD.TECH NOTE NO.437	SUTHERLAND AJ;CROWE DS	1981
03803.pdf	CONTROL ROD WORTHS AT CURTAIN HEIGHT OF 722MM DURING RUN 5 OF PFR	PFR EXPERIMENTAL RESULTS SHEET NO.113;OETD.TECH NOTE NO.452	SUTHERLAND AJ;CROWE DS	1981
03804.pdf	PFR ZERO POWER FLOW COEFFICIENT AT START OF RELOAD 5	PFR EXPERIMENTAL RESULTS SHEET NO.114;OETD.TECH NOTE NO.466	CROWE DS;LORD DJ	1981
03805.pdf	CONTROL ROD AND SHUT OFF ROD WORTHS AT END OF RUN 5 OF PFR	PFR EXPERIMENTAL RESULTS SHEET NO.115;OETD.TECH NOTE NO.468	CROWE DS;LORD DJ	1981
03806.pdf	PFR ZERO POWER FLOW COEFFICIENT MEASUREMENTS AT THE END OF RELOAD 5	PFR EXPERIMENTAL RESULTS SHEET NO.116;OETD.TECH NOTE NO.487	CROWE DS;LORD DJ;SUTHERLAND AJ	1981
03807.pdf	PFR CONTROL ROD CALIBRATION IN SOURCE REGIME (< 1KW) TO INVESTIGATE RELOAD REACTIVITY DISCREPANCY - START OF RUN 6	PFR EXPERIMENTAL RESULTS SHEET NO.117;OETD.TECH NOTE NO.503	CROWE DS;LORD DJ;SUTHERLAND AJ	1981
03808.pdf	CALIBRATION OF PFR ABSORBER RODS AT START OF RUN 6	PFR EXPERIMENTAL RESULTS SHEET NO.118;OETD.TECH NOTE NO.517	CROWE DS;SUTHERLAND AJ	1982
03809.pdf	BIAS CURVES FOR PFR LOW POWER CHAMBERS	PFR EXPERIMENTAL RESULTS SHEET NO.119;OETD.TECH NOTE NO.530	CROWE DS	1982
03811.pdf	CONTROL ROD AND SHUT OFF ROD WORTHS AT END OF RUN 6 OF PFR	PFR EXPERIMENTAL RESULTS SHEET NO.121;OETD.TECH NOTE NO.552	CROWE DS;SUTHERLAND AJ	1982
03812.pdf	REACTIVITY EFFECTS OF FLOW CHANGES AT THE END OF RUN 6 OF PFR	PFR EXPERIMENTAL RESULTS SHEET NO.122;OETD.TECH NOTE NO.553	CROWE DS;LORD DJ;SUTHERLAND AJ	1982
03813.pdf	PERFORMANCE CURVE CHECKS ON THE PFR LOW POWER CHANNELS	PFR EXPERIMENTAL RESULTS SHEET NO.123;OETD.TECH NOTE NO.557	SUTHERLAND AJ	1982

Reference No.	Title	Location	Authors	Year
03814.pdf	SUB CRITICAL MONITORING DURING RELOAD 5 OF PFR	PFR EXPERIMENTAL RESULTS SHEET NO.124;OETD.TECH NOTE NO.565	CROWE DS	1982
03816.pdf	REACTIVITY EFFECTS OF FLOW CHANGES AT START OF RUN 7 OF PFR	PFR EXPERIMENTAL RESULTS SHEET NO.126	LORD DJ;SUTHERLAND AJ;CROWE DS	1982
03817.pdf	CALIBRATION OF PFR ABSORBER RODS AT START OF RUN 7	PFR EXPERIMENTAL RESULTS SHEET NO.127;OETD TECH NOTE NO.628	CROWE DS;SUTHERLAND AJ	1982
03818.pdf	MEASUREMENT OF THE ISOTHERMAL TEMPERATURE COEFFICIENT IN PFR AT THE START OF RUN 7	PFR EXPERIMENTAL RESULTS SHEET NO.128;OETD TECH NOTE NO.633	CROWE DS;LORD DJ	1982
03819.pdf	SUB-CRITICAL MONITORING DURING RELOAD 6 IN PFR	PFR EXPERIMENTAL RESULTS SHEET NO.129;OETD TECH NOTE NO.653	CROWE DS	1982
03820.pdf	MEASUREMENT OF THE ISOTHERMAL TEMPERATURE COEFFICIENT IN PFR DURING RUN 7	PFR EXPERIMENTAL RESULTS SHEET NO.130;OETD TECH NOTE NO.697	CROWE DS	1983
03821.pdf	INTERCALIBRATIONN OF POWER CHANNELS IN PFR	PFR EXPERIMENTAL RESULTS SHEET NO.131;OETD TECH NOTE NO.702	SUTHERLAND AJ	1982
03824.pdf	CONTROL ROD AND SHUT OFF ROD WORTHS AT END OF RUN 6 OF PFR	PFR EXPERIMENTAL RESULTS SHEET NO.134;OETD TECH NOTE NO.781	CROWE D S;SUTHERLAND AJ	1983
03825.pdf	REACTIVITY EFFECTS OF FLOW CHANGES AT THE END OF RUN 7 OF PFR	PFR EXPERIMENTAL RESULTS SHEET NO.135;OETD TECH NOTE NO.782	CROWE DS;LORD DJ;SUTHERLAND AJ	1983
03842.pdf	CALIBRATION OF PFR ABSORBER RODS AT END OF RUN 1	PFR EXPERIMENTAL RESULTS SHEET NO.74	CROWE DS;LORD DJ;SUTHERLAND AJ	

Reference No.	Title	Location	Authors	Year
03843.pdf	REACTIVITY FEEDBACK EXPERIMENTS AT THE END OF RUN 1 (FEBRUARY 1978)	PFR EXPERIMENTAL RESULTS SHEET NO.73	LORD DJ;DICKSON AK	
03844.pdf	RESULTS FROM SIGNALS RECORDED ON MAGNETIC TAPE FOR REACTOR TRIP ON 23 JANUARY 1978	PFR EXPERIMENTAL RESULTS SHEET NO.72	CROWE DS;SUTHERLAND AJ	
03849.pdf	MEASUREMENTS OF PFR POWER COEFFICIENT TO INVESTIGATE COMPONENT DUE TO EXPANSION OF CONTROL ROD SUPPORTS	PFR EXPERIMENTAL RESULTS SHEET NO.67	LORD DJ	
03851.pdf	MEASUREMENTS OF PFR ISOTHERMAL TEMPERATURE COEFFICIENT WITH PARTICULAR REFERENCE TO COMPONENT DUE TO RELATIVE MOTION OF CORE AND CONTROL RODS	PFR EXPERIMENTAL RESULTS SHEET NO.65	CROWE DS;SUTHERLAND AJ	
03852.pdf	MEASUREMENT OF POWER COEFFICIENTS AND REACTIVITY BURN-UP IN PFR FOR AUGUST TO OCTOBER 1977	PFR EXPERIMENTAL RESULTS SHEET NO.64	SUTHERLAND AJ	
03885.pdf	MEASUREMENT OF THE RELATIVE WORTH OF EACH SHUT-OFF ROD AT LOW POWER	PFR EXPERIMENTAL RESULTS SHEET NO.30	CROWE DS;SUTHERLAND AJ	
03887.pdf	MEASUREMENT OF THE SHUT OFF RODS SHAPE FUNCTION	PFR EXPERIMENTAL RESULTS SHEET NO.28	CROWE DS;SUTHERLAND AJ	
03889.pdf	PRELIMINARY CONTROL ROD CALIBRATION AT LOW POWER TO MONITOR FOR LOSS OF ABSORBER	PFR EXPERIMENTAL RESULTS SHEET NO.26	SUTHERLAND AJ;CROWE DS	
03945.pdf	SYSTEMS RELIABILITY SERVICE - PROTECTION OF THE PFR AGAINST CONTROL-ROD RUNAWAY - A RELIABILITY ANALYSIS	SRS/ASG/2007/1	UKAEA	
04339.pdf	CONTROL OF SODIUM VAPOUR PENETRATION INTO THE PFR CONTROL AND SHUT-OFF MECHANISM	TRG REPORT 1939	HIGSON J	
04343.pdf	SIMPLE STRESS ANALYSIS OF THE BORON CARBIDE CONTROL RODS IN PFR	TRG REPORT 2834;MWP/P(76)245;FRASG/P(76)45	KELLY BT	
04345.pdf	THE VARIATION OF REACTION RATES AND SUBASSEMBLY POWERS WITH CONTROL ROD CURTAIN POSITION IN THE PFR	TRG REPORT 2838	WEBSTER EB	
04512.pdf	FAST REACTOR ABSORBERS/CONTROL RODS - EXPERIMENTAL PROGRAMME STATUS REPORT - APRIL 1983	FRASG/N(83)47	GILCHRIST KE	1983
04513.pdf	THE PIE OF THE PFR TANTALUM MINI-ROD EXPERIMENT (13/01)	NDM-1079;MWP/P(83)1148;FRASG/P(83)198	MOTTERSHEAD D	1983

Reference No.	Title	Location	Authors	Year
04514.pdf	EXAMINATION OF BORON CARBIDE PINS FROM THE PFR MK2 CONTROL ROD SEG AFTER OPERATION FOR 180 E.F.P.D.	NDR-944;MWP/P(83)147;FRASG/P(83)197	MOTTERSHEAD D;GILCHRIST KE;BROCKLEHURST JE;KELLY BT	1983
04515.pdf	THE IRRADIATION BEHAVIOUR OF BORON CARBIDE ABSORBER PINS IN PFR EXPERIMENT 13/06	NDR-711;MWP/P(81)1132;FRASG/P(81)182	MOTTERSHEAD D;GILCHRIST KE;KELLY BT	1983
05556.pdf	X-RADIOGRAPHIC EXAMINATION OF SHUT-OFF RODS VVX,TYX,JWS,JHD AND CONTROL ROD CXH	PFR/FEDWP/P(85)1128;PFR/TC/P(86)5;DFMC/P(85)28;IFC REPORT 87	McLOUGHLAN D;LILLEY RJ	
06028.pdf	CONTROL AND DYNAMICS REVIEW 1985/86	FRDCC/PPWG/P(86)65	BUTTERFIELD MH	1986
06715.pdf	THE COMPUTER CODE PEBBLE AND A REVIEW OF ITS USE IN PREDICTING ABSORBER ROD EXERCISING LOADS	PFR/FEDWP/P(86)1204;EPD/TN(86)270	LIGHTOWLERS RJ	1986
06819.pdf	THE MANUFACTURE AND INSPECTION OF PE16 PFR MK4 CONTROL ROD WRAPPERS	ND-M-2928(S)	PARRY P	1986
07758.pdf	REVIEW OF ABSORBER ROD EXERCISING DURING RUNS 8,9, AND 10	PFR/TC/P(86)9	MELHUISH KR	
08274.pdf	BORON CARBISE CONTROL ROD PIN DEVELOPMENT	FRDCC/P(86)202	KELLY BT	1986
08988.pdf	PEBBLE*: A COMPUTER PROGRAM FOR ANALYSING STRUCTURAL INTERACTIONS BETWEEN FAST REACTOR ABSORBER COMPONENTS	NPD/TN(84)200	RIDING DJ	1985
09056.pdf	IMPROVING THE PERFORMANCE OF B4C CONTROL ROD PINS	ND-M-3066; FRASG/N(85)50	BROCKLEHURST JE;GILCHRIST KE	
10158.pdf	PFR ABSORBER ROD EXERCISING A STATUS REPORT FROM THE WORKING GROUP - FEBRUARY 1987	PFR/TC/P(87)158	LORD DJ;HENDERSON JDC	
11058.pdf	COMMENTS ON CONTROL AND DYNAMICS REVIEW 1986/7 PPWG/P(87)115	PFR/TC/P(87)203	BUTTERFIELD MH	1988
12225.pdf	RECOMMENDED LIFE LIMIT FOR MK IIIE CONTROL RODS BASED ON A 70 DPA NFE DOSE LIMIT FOR AUSTENITIC COMPONENTS	PFR/FEDWP/P(89)1443;PFR/TC/P(89)316	NELLIGAN DJ	
12328.pdf	MEASUREMENT OF THE FRICTION FORCES WITHIN THE PFR CONTROL GEAR SYSTEM DUE TO MISALIGNMENT BETWEEN THE MECHANISM AND CORE	PFR/FDWP/P(72)162	KNOWLES PJ	
12341.pdf	THERMAL STABILITY TESTING OF FAST REACTOR FUEL ELEMENT AND CONTROL ROD COMPONENTS	PFR/FDWP/P(72)178	ROSE KM;BENTLEY JW	
12342.pdf	PROGRAMME FOR THERMAL STABILITY TESTING PFR ABSORBER ROD COMPONENTS	PFR/FDWP/P(72)179	BENTLEY JW;ROSE KM	

Reference No.	Title	Location	Authors	Year
12364.pdf	ENDORSMENT OF PFR Mk IV CONTROL ROD DESIGN	PFR/FDWP/P(73)236	GATLEY JA	
12369.pdf	PFR MK 1V CONTROL ROD DESIGN EVALUATION OF THERMAL STABILITY	PFR/FDWP/P(73)231	BENTLEY JW	
12383.pdf	THE PFR MK 1V CONTROL ROD DESIGN	PFR/FDWP/P(73)214	GATLEY JA	
12411.pdf	PROGRAMME FOR ENDORSEMENT OF PFR MK1V CONTROL ROD DESIGN	PFR/FDWP/P(74)258	GATLEY JA	
12440.pdf	THERMAL AND IRRADIATION RELAXATION OF PE 16 SPRINGS FOR PFR MK 1V CONTROL RODS	PFR/FDWP/P(74)289	GATLEY JA	
12444.pdf	PRELIMINARY DESIGN OF A EUROPIA CONTROL ROD	PFR/FDWP/P(73)192	GATLEY JA	
12583.pdf	REVISED PROGRAMME FOR PFR MK 1V CONTROL RODS	PFR/FDWP/P(75)352	GATLEY JA	
12861.pdf	PFR CONTROL ROD MK11 WRAPPER ASSEMBLY - THERMAL STABILITY TEST	PFR/FDWP/P(75)369	BENTLEY JW	
13100.pdf	THERMAL STABILITY TESTING OF A PFR MK III CONTROL ROD SHELL AND A SIMULATED PFR MK 1V CONTROL ROD	ND M 820;FR/FEDWP/P(79)669	BENTLEY JW	
13160.pdf	PRELIMINARY EXAMINATION OF THE BOWING OF THE PFR CONTROL AND SHUT OFF RODS AND THEIR ASSOCIATED GUIDE TUBES	PFR/FEDWP/P(77)529	SIMPSON A	
13252.pdf	EXTENDING THE LIFE OF PFR CONTROL ROD PINS	NRL-M-2055;FRASG/P(88)230	OAKDEN MML;HARRISON WR	1988
13255.pdf	RECOMMENDED LIFE LIMITS FOR PFR CONTROL RODS BASED ON BORCON PREDICTIONS OF ABSORBER PIN BEHAVIOUR	NRL-M-2053; FRASG/P(88)229	OAKDEN MM;HARRISON WR	
13489.pdf	PFR CONTROL RODS A PRELIMINARY ASSESSMENT OF ROD GUIDE C0NE SWELLING	FEDWP/P(79)692	TRIGGS GW	
13490.pdf	MHD SHOCK ABSORBER FOR PFR AUXILIARY SHUT -OFF RODS FEASIBILITY STUDY AND DEVELOPMENT PROPOSAL	FEDWP/P(79)691	THATCHER G;DAVIDSON DF	
13558.pdf	ENDURANCE TEST AT RNL ON PFR MKIV CONTROL ROD EXPERIMENTAL SPECIFICATION	ND-M-1180; PFR/FEDWP/P(80)725	MEREDITH BE	1980
13594.pdf	A DISTORTION ANALYSIS OF THE PFR MK4 CONTROL ROD AND GUIDE TUBE	CFR/FEDWP/P(81)767;RTD/TN/P(81)150	SIMMERS DA	1981
13616.pdf	THE DESIGN OF A SHORTENED EXTENSION ROD FOR USE WITH A PFR MKIV CONTROL ROD IN VIBRATION EXPERIMENTS	ND-R-716; PFR/FEDWP/P(81)833	DUTHIE JC	1981
14075.pdf	OPERATIONAL LIFE OF PFR MK IIIC CONTROL ROD PINS	NRL-M-2149(S); PFR/SWP/ESC/P(89)22;FRASG/P(89)236	KELLY BT	1989
14401.pdf	BORCON PREDICTIONS FOR THE PFR MKIIIE CONTROL ROD PINS AT 650MW REACTOR POWER	NRL-M-2027(S); CFWG/FPSG/P(89)3	KELLY BT	1989
15242.pdf	COMPARISON OF BORCON VERSION V5A WITH POST IRRADIATION EXAMINATION OF PFR CONTROL RODS SEG AND DKA	NRL-R-2035	KELLY BT	1989

Reference No.	Title	Location	Authors	Year
17347.pdf	THE POST IRRADIATION EXAMINATION OF BORON CARBIDE PINS FROM THE PFR MK III CONTROL ROD DKA	NRL-M-2133; FRDCC/FRASG/P(89)234;FRDCC/FEWP/P(89)22	OAKDEN MM; ADAM RW; MUNRO B; HARRISON WR; HANLEY D; BROCKLEHURST JE	
17628.pdf	RECOMMENDED LIFE LIMITS OF PFR CONTROL ROD PINS AND ABSORBER MATERIALS CLUSTERS	AEA-TRS-5041; CFWG/FSPG/P(90)36	KELLY BT	1990
18080.pdf	EVALUATE THE REACTIVITY WORTH OF REMOVING TWO CONTROL RODS	PFR/SWP/P(89)27	HENDERSON JDC	1989
18774.pdf	THE PROPOSED ABSORBER ROD EXERCISING PROGRAMME FOR RUN 22	PFR/SWP/P(91)1;	WASHINGTON A	1991
19196.pdf	DRAFT 1 PFR EXPERIMENTS DEMONSTRATION OF CONTROL ROD ENHANCED EXPANSION DEVICE (CREED) IN PFR		DOSTAL M;HARTLEY DJ;SHERLOCK P;	1991
19940.pdf	APPLICATION FOR A SUSPENSION OF MITS ITEM DEALING WITH PFR MAGNET SCREW ASSEMBLIES	PE1/1843;DNSC/P(91)65	SANDISON A	1991
19996.pdf	APPLICATION FOR A SUSPENSION ON MITS ITEM DEALING WITH PFR MAGNET SCREW ASSEMBLIES	PE1/1843;DNSC/P(91)65	SANDISON A	1991
20546.pdf	A MEASUREMENT OF THE PFR CONTROL ROD S-CURVE AT THE END OF PFR RUN 22	PE1/2408;PFR/TC/P(91)490;PFR/ER/203	NEWTON TD	1991
20952.pdf	PFR CONTROL ROD ENHANCED EXPANSION DEVICE (CREED) CONCEPT PAPER FOR EFR END-OF-LIFE EXPERIMENTS		BARROWMAN GR	1992
21483.pdf	TEMPERATURE FLUCTUATIONS WITHIN PFR CONTROL ROD SHROUD TUBES REML WATER TEST RESULTS	ACSCM/P(77)14; PFR/SWP/P(77)14	FEWSTER J	
21490.pdf	CONTROL ROD DRIVE INCIDENT IN PFR	PFR/SWP/P(77)46	GRAY J;BROOMFIELD AM	
21491.pdf	CASE FOR A POSSIBLE FURTHER DELAY IN SAFETY ROD COMMISSIONING	PFR/SWP/P(77)51	HENDERSON JDC	
21492.pdf	SAFETY REQUIREMENTS FOR AN ALTERNATIVE SHUT-DOWN DEVICE FOR PFR	PFR/SWP/P(77)62	BROADLEY D; McDonald DR	
21997.pdf	OPERATIONAL LIFE OF PFR MK III C CONTROL ROD PINS	NRL-M-2149(S); PFR/SWP/ESC/P(89)22;FRASG/P(89)236	KELLY BT	1989
22022.pdf	REVIEW OF ABSORBER ROD EXERCISING RESULTS FROM RUNS 8 TO 11 (INCL) AND PROPOSALS FOR RUN 12	PFR/SWP/P(86)34	HENDERSON JDC;MELHUISH KR	1986
22057.pdf	PROPOSED ABSORBER ROD EXERCISING DURING RUN 10	PFR/SWP/P(85)40	MELHUISH KR	1985

Reference No.	Title	Location	Authors	Year
22065.pdf	PFR ABSORBER ROD EXERCISING A STATUS REPORT FROM THE WORKING GROUP	PFR/SWP/P(87)7	LORD DJ;HENDERSON JDC	1987
22090.pdf	PFR ABSORBER ROD EXERCISING A FINAL REPORT FROM THE WORKING GROUP	PFR/SWP/P(88)1	LORD D;HENDERSON JDC	1988
22092.pdf	ROD EXERCISING - RUNS 15 AND 16A	PFR/SWP/P(88)3	HENDERSON JDC	1988
22434.pdf	EVALUATE THE REACTIVITY WORTH OF REMOVING TWO CONTROL RODS	PFR/SWP/P(89)27	HENDERSON JDC	1989
22435.pdf	SHORT-TERM CONCESSION ON OPERATING DIRECTIVE ON SHUTDOWN MARGIN	PFR/SWP/P(89)28	HENDERSON JDC	1989
23067.pdf	ABSORBER ROD EXERCISING PLANS	PFR/SWP/P(88)22	HENDERSON JDC	1988
24249.pdf	OPERATIONAL MOVEMENTS OF PFR BORON CARBIDE CONTROL RODS 1979 - 1980	ND-M-1835	MOTTERSHEAD D	1982
24251.pdf	OPERATIONAL MOVEMENTS OF PFR BORON CARBIDE CONTROL RODS 1974-1978	ND-M-1070	MOTTERSHEAD D	1980
24464.pdf	THE PROPOSED ABSORBER ROD EXERCISING PROGRAMME FOR RUN 29	RS/SWP/(93)2	WASHINGTON A	1993
24657.pdf	BORCON (V5B) CONTROL ROD PIN MODEL CALCULATIONS FOR THE PFR MK III F DESIGN	NRL-M-2039;CFWG/FPSG/P(89)8	KELLY BT	1989
24659.pdf	COMPARISON OF BORCON VERSION V5A WITH POST-IRRADIATION EXAMINATION OF PFR CONTROL RODS SEG AND DKA	NRL-M-2035	KELLY BT	1989
24660.pdf	RECOMMENDED LIFE LIMITS FOR PFR SHUT OFF RODS BASED ON BORCON PREDICTIONS OF ABSORBER STACK BEHAVIOUR	NRL-M-2134; FRDC/FRASG/P(89)235;FRDCC/FEWP/P(89)21;PFR/ESC/P(89)25	OAKDEN MM	1989
25490.pdf	SPECIFICATION FOR PROTOTYPE CONTROL ROD MECHANISM	UKAEA (EG)47045A		
25491.pdf	SPECIFICATION OR THE CONTROL ROD MECHANISMS	UKAEA (EG)47053A		
25492.pdf	SPECIFICATION OR THE OSCILLATOR AND SECONDARY SHUT DOWN MECHANISM	UKAEA (RG)47060A		
25493.pdf	SPECIFICATION FOR MAGNET AND WEIGHT SENSING DEVICE FOR CONTROL ROD MECHANISM	UKAEA (EG)14917		
25493.pdf	SPECIFICATION FOR MAGNET AND WEIGHT SENSING DEVICE FOR CONTROL ROD MECHANISM	UKAEA (EG)14917		
25495.pdf	THE PFR PROTOTYPE MAGNET TESTS (ENGLISH ELEC WHETSTONE REPORT)	W/AT 1632;	SAAGI R	
25495.pdf	THE PFR PROTOTYPE MAGNET TESTS (ENGLISH ELEC WHETSTONE REPORT)	W/AT 1632;	SAAGI R	
25520.pdf	THE PROPOSED ABSORBER ROD EXERCISING FOR RUN 29	RS/SWP/P(93)2;PE1/7732	WASHINGTON A	1993
25863.pdf	POBABILITY OF SHUTDOWN FAILURE DUE TO ABSORBER ROD BOWING IN THE PROTOTYPE FAST REACTOR	NJH 84/12	HOLLOWAY NJ	1984
26000.pdf	ABSROD A COMPUTER CODE FOR CALCULATING WRAPPER TEMPERATURES IN CONTROL RODS AND GUIDE TUBES IN PFR AND CFR DESCRIPTION AND USERS GUIDE	ND-M-1121	WILLIAMS BD;MCAREAVEY G	1980

Reference No.	Title	Location	Authors	Year
26002.pdf	PFR CONTROL AND SHUT OFF RODS (MK 3) DROP TESTS IN WATER	TRG-M-5087	BARRETT WI	
26383.pdf	A PROGRAMME TO STUDY THE LOSS OF ABSORBER MATERIAL FROM FAILED PINS IN FAST REACTOR CONTROL RODS	FRDC/P(78)305;FRASG/P(78)103;ND-M-235	KELLY BT	
26387.pdf	PROGRAMME FOR THE POST IRRADIATION EXAMINATION OF PFR CONTROL ROD TET (BORON CARBIDE)	FRDC/MWP/P(78)310;FRASG/P(78)110;ND-M-330	KELLY BT	
26388.pdf	PROGRAMME FOR THE POST IRRADIATION EXAMINATION OF PFR CONTROL ROD LWG (TANTALUM)	FRDC/MWP/P(78)311;FRASG/P(78)111;ND-M-328	KELLY BT	
26389.pdf	EXAMINATION PROGRAMME FOR PFR IRRADIATIONS 13/01-06	FRDC/MWP/P(78)312;FRASG/P(78)112;ND-M-329	KELLY BT	
26390.pdf	BORCON CALCULATIONS OF THE LIFE OF PFR MK II AND MK III CONTROL ROD PINS	FRDC/MWP/P(78)313;FRASG/P(78)113;ND-M-510	KELLY BT;PRSTON SD	
26391.pdf	STORAGE AND WASTE DISPOSAL ASPECTS OF ALTERNATIVE FAST REACTOR ABSORBER MATERIALS	FRDC/MWP/P(78)316;FRASG/P(78)116;RD/B/N4319	SIMPSON KA	
26400.pdf	PROPOSALS TO INCLUDE 40% ENRICHED BORON CARDIBE PINS IN MKIV PFR CONTROL RODS	FRDC/MWP/P(78)327;FRASG/P(78)127;ND-M-577	KELLY BT	
26417.pdf	COMPUTER MODELLING OF THE PERFORMANCE OF FAST REACTOR CONTROL ROD PINS BORCON MK3	FRDC/MWP/P(79)1130;FRASG/P(79)153;ND-M-1053	KELLY BT;HOY CJ	
26719.pdf	DESIGN REPORT ON PFR CONTROL RODS SHUT-OFF RODS AND THE SAFETY ROD	PFR/SWP/P(71)36	STAMFORD S	
27780.pdf	PFR HOLDING MAGNET DESIGN DATA CONTROL ROD MECHANISMS		ENGLISH ELECTRIC COMPANY	
27780.pdf	PFR HOLDING MAGNET DESIGN DATA CONTROL ROD MECHANISMS		ENGLISH ELECTRIC COMPANY	
27781.pdf	4TH DRAFT: DESCRIPTIVE MANUAL CONTROL ROD & SHUT OFF ROD MECHANISMS	G.1.2	FRDO RISLEY	
27963.pdf	BOW LIMITS FOR PFR MK1A SHUT-OFF RODS AND GUIDE TUBES	RTD/TN(79)40	SIMMERS DA	
27964.pdf	THERMAL STRESSES IN THE EXTENSION ROD OF A PFR CONTROL ROD	PFR/SWP/P(79)14;TN/P(79)312	PRICE JWH	
27965.pdf	AN EXAMINATION OF THE POSSIBILITY OF THREEPOINT CONTRACT BETWEEN PFR SHUT-OFF RODS AND GUIDE TUBES	RTD/TN(79)41	TRIGGS GW	
27970.pdf	BOW LIMITS FOR PFR MK1A SHUT OFF RODS AND MK3 CONTROL RODS IN A MK 3 GUIDE TUBE	RTD/TN(79)66	SIMMERS DA	
27971.pdf	THE EFFECT OF IRRADIATION CREEP ON PFR Mk 1A SHUT OFF RODS	rtd/tn(79)70	RIDING DJ	
27973.pdf	PFR ABSORBER RODS GUIDE TUBES AND ASSOCIATED ITEMS APPROACHING LIMITS AT RELAOD 4 DURING RUN 5	RTD/TN(79)73	SIMMERS DA	
28079.pdf	PFR CONTROL AND SHUT OFF RODS FUNCTIONAL RELIABILITY AND COMMISSIONING TESTS	PFR/OC/P(72)11;PFR/CP/16	HENDERSON JDC	

Reference No.	Title	Location	Authors	Year
29608.pdf	THE INGRESS OF SOIUM INTO MK 2 PFR VENTED CONTROL ROD PINS	ND-R-584;FRDC/MWP/P(80)1118;FRASG/P(80)168	WALKER DEY;MURGATROYD RA;BLAND JT	1982

Table 10: Wood Report List: Failed Fuel Detection

Reference No.	Title	Location	Authors	Year
00137.pdf	PROPOSAL FOR THE IN-CORE SECTION OF A RADIAL BREEDER BPD SYSTEM	TN/P(83)605;CTWG/P(83)36	SHERWOOD D	1983
00540.pdf	COMMENTS ON THE DETECTION OF FAILED FUEL ELEMENTS IN CDFR	NDM-1357;DRAFT;CFR/SWP/P(80)20;FFWG/P(80)11	CARTWRIGHT DK	1980
00540.pdf	COMMENTS ON THE DETECTION OF FAILED FUEL ELEMENTS IN CDFR	NDM-1357;DRAFT;CFR/SWP/P(80)20;FFWG/P(80)11	CARTWRIGHT DK	1980
01230.pdf	WORK PACKAGE 322 N4: BPD	FREWG/P(84)028	LAMBERT B;BARROWMAN G	1984
01724.pdf	FUEL ELEMENT FAILURE DATA	MWP/P(81)1353; FIXSG/P(81)189	TAYLOR AF; NUTTER NR	1981
01900.pdf	DEVELOPMENT WORK FOR AN ULTRASONIC SYSTEM FOR IDENTIFICATION OF PFR CORE COMPONENTS	TRG/R/2313;EST/P(74)190	MCKNIGHT JA; WILLIS P;CARTWRIGHT DK	
03200.pdf	FISSION GAS ESCAPE FROM FAILED FUEL PINS	FFWG/P(77)13;FRAX NOTE 45;FRGN/528	HILL DJ	
03204.pdf	THE PRINCIPLES OF FAILED FUEL DETECTION IN FAST REACTORS (LMFBRS)	FFWG/P(77)24;FRSBWG/N(77)34	DIGGLE WR;CARTWRIGHT DK	
03208.pdf	FISSION-PRODUCT MEASUREMENTS IN THE SCARABEE MONO AND SEVEN-PIN LOSS-OF-COOLANT EXPERIMENTS	FFWG/P(78)17	CARTWRIGHT DK	
03223.pdf	AN ASSESSMENT OF BPD SIGNALS FROM THE FIRST FUEL FAILURES IN PFR	NDM-897;FFWG/P(79)10	CARTWRIGHT DK DINGTON P	
03620.pdf	THE CALCULATIONAL BASIS FOR THE CALIBRATION OF THE PFR BCD SYSTEM	TRG MEMO 6153	CARTWRIGHT DK;DIGGLE WR	
03783.pdf	HIGH SIGNAL LEVELS IN THE PFR BURST PIN DETECTION EQUIPMENT: 23.8.78-17.10.78	PFR EXPERIMENTAL RESULTS SHEET NO.90	LENNOX TA; MACLEOD DJ; CATHRO IS; CARTWRIGHT DK	
03792.pdf	THE HISTORY OF SIGNALS IN THE PFR BPD EQUIPMENT DURING RUN 3	PFR EXPERIMENTAL RESULTS SHEET NO.101;OETD.TECH NOTE NO.96	LENNOX TA; MACLEOD DJ; CATHRO IS	

Reference No.	Title	Location	Authors	Year
03815.pdf	A COMPLETE SCAN AT 180 MW(TH) WITH THE MARK IV LOCATION LOOP DELAYED NEUTRON MONITOR	PFR EXPERIMENTAL RESULTS SHEET NO.125;OETD.TECH NOTE NO.579	CROWE DS;SUTHERLAND AJ; LENNOX TA; MACLEOD DJ	1982
03827.pdf	MONITORING OF THE FAILED FUEL DETECTION SIGNALS AT THE RUN 7 SHUTDOWN	PFR EXPERIMENTAL RESULTS SHEET NO.137;OETD TECH NOTE NO.798	LENNOX TA; MACLEOD DJ; STEELE KB; MORRISON NS	1983
03828.pdf	GAMMA SPECTROSCOPY MEASUREMENTS ON THE MK IV LOCATION LOOP DN COIL AT THE END OF RUN 7	PFR EXPERIMENTAL RESULTS SHEET NO.138;OETD TECH NOTE NO.799	MACLEOD DJ; LENNOX TA; SUTHERLAND AJ	1983
03829.pdf	INITIAL MEASUREMENT OF CONTROL ROD CURTAIN WORTH OF PFR AT START OF RUN 8	PFR EXPERIMENTAL RESULTS SHEET NO.139;OETD TECH NOTE NO.825	SUTHERLAND AJ; LORD DJ	1984
03830.pdf	CALIBRATION OF PFR ABSORBER RODS AT START OF RUN 8	PFR EXPERIMENTAL RESULTS SHEET NO.140;OETD TECH NOTE NO.826	CROWE DS;SUTHERLAND AJ	1984
03831.pdf	REACTIVITY EFFECTS OF FLOW CHANGES AT START OF RUN 8 OF PFR	PFR EXPERIMENTAL RESULTS SHEET NO.141;OETD TECH NOTE NO.832	CROWE DS; SUTHERLAND AJ; LORD DJ	1984
03833.pdf	CALIBRATION OF PFR ABSORBER RODS AT START OF RUN 2	PFR EXPERIMENTAL RESULTS SHEET NO.83	CROWE DS; SUTHERLAND AJ;LORD DJ	
03835.pdf	HIGH FISSION PRODUCT ACTIVITY IN THE ARGON GAS BLANKET 16.7.78-17.7.78	PFR EXPERIMENTAL RESULTS SHEET NO.81	LENNOX AT; CATHRO IS; MACLEOD DJ	
03837.pdf	RESULTS FROM SIGNALS RECORDED ON MAGNETIC TAPES FOR REACTOR TRIP ON 20 FEBRUARY 1978	PFR EXPERIMENTAL RESULTS SHEET NO.79	CROWE DS;SUTHERLAND AJ	

Reference No.	Title	Location	Authors	Year
03838.pdf	THE TRANSIENT SIGNAL IN THE IHX DELAYED NEUTRON MONITOR FOLLOWING A REACTOR TRIP	PFR EXPERIMENTAL RESULTS SHEET NO.78	CROWE DS; SUTHERLAND AJ	
03840.pdf	CHECKS ON THE IHX DNM COUNTING STACK	PFR EXPERIMENTAL RESULTS SHEET NO.76	LENNOX TA	
03871.pdf	NOTES ON BPD TESTS AUGUST-DECEMBER 1976	PFR EXPERIMENTAL RESULTS SHEET NO.45	WHEELER RC; LENNOX TA	
03875.pdf	PRELIMINARY DATA FROM BPD BULK AND LOCATION THIMBLE TESTS	PFR EXPERIMENTAL RESULTS SHEET NO.41	LENNOX TA; SOMERVILLE AC	
03877.pdf	ANALYSIS OF PFR GAS BLANKET BETA PRECIPITATOR SIGNAL FOLLOWING A REACTOR TRIP	PFR EXPERIMENTAL RESULTS SHEET NO.39	SOMERVILLE AC	
03901.pdf	STEADY STATE SIGNALS IN THE PFR BPD EQUIPMENT DURING RUN 1 (OCTOBER 1977)	TC/P(77)22	LENNOX TA; MACLEOD DJ	
03958.pdf	AN ASSESSMENT OF PFR SUB-ASSEMBLY PROTECTION	PFR/SWP/P(73)66	MAIDMENT L	
04066.pdf	MEASUREMENTS OF BPD PIPE GROWTH RELATIVE TO GUIDE-TUBE WRAPPERS FOR GUIDE-TUBES FROM RELOADS 6,7 AND 8	DFMC/P(85)17;PFR/FEDWP/P(85)1077	LILLEY RJ; CARFRAY J	1985
04344.pdf	DEFECTED PIN EXPERIMENTS IN SCARABEE, 1973 AND 1974	TRG REPORT 2835	CARTWRIGHT DK; DIGGLE WR; MANENT G	
04344.pdf	DEFECTED PIN EXPERIMENTS IN SCARABEE, 1973 AND 1974	TRG REPORT 2835	CARTWRIGHT DK; DIGGLE WR; MANENT G	
04484.pdf	THE MARK IV LOCATION LOOP FOR THE PFR BPD SYSTEM	TN/P(78)225;TF/P(78)288;PFR/SWP/P(78)1	COOKE B	
04571.pdf	A REVIEW OF BPD EQUIPMENT FOR CDFR	TN/P(80)405;PDR/22;ASDSWP/P(80)11	SUKER JH	1980
04651.pdf	ROAMING MONITOR FOR LOCATING SUB-ASSEMBLY FAILURES	PPWP/P(82)355; CFR/SWP/P(82)7	BURTON EJ; SMITH DCG	1982
04900.pdf	A PROPOSED FAST REACTOR PROGRAMME FOR FAILED FUEL DETECTION AND LOCATION	FREWG/P(85)169	LENNOX TA;GREGORY CV	1985
05709.pdf	FAILED FUEL PIN MODELLING	AERE 3532; FEWP/P(86)3;FRSWG/SAFSG/P(86)1	MATTHEWS JR	

Reference No.	Title	Location	Authors	Year
06468.pdf	THE LOCAL SUB-ASSEMBLY ACCIDENT AND FAILED FUEL BEHAVIOUR IN PFR	FRSWG/SAFSG/P (86)22	BURTON EJ	1986
07656.pdf	CIRCUIT ACTIVITY LEVELS WHEN OPERATING WITH FAILED FUEL	FRDCC/SCWG/P(86)90	EVANS	1986
07747.pdf	PFR FAILED FUEL AND GASEOUS DISCHARGES	PFR/TC/P(86)40	GRIFFITHS JH	
07759.pdf	REVIEW OF MK IV BURST PIN DETECTION SYSTEM	PFR/TC/P(86)10	HODGSON D	1986
08419.pdf	PFR DESIGN REVIEW - ABSORBER RODS,GUIDE TUBES AND ABSORBER DMSA EXPERIMENTS.BCD FOILS	FRDCC/FEWP/P(87)10	FORD J	1987
09543.pdf	A PROPOSED FAST REACTOR PROGRAMME FOR FAILED FUEL DETECTION AND LOCATION	DPC/P(84)22;ND-M-2396;FRDCC/FEWP/P(84)31;FRSWG/SISG/P(84)4	LENNOX TA;GREGORY CV	1984
10091.pdf	PFR HALIP BPD MK IV - STATUS	FRDCC/P(87)251	CLARE A	1987
12309.pdf	AXIAL MISALIGNMENT OF THE BPD TAKE OFF	PFR/FDWP/P(71)139	MITCHELL CH	
12517.pdf	PFR IHX DELAYED NEUTRON MONITOR VALIDATION EXPERIMENTS ON A 1/5 SCALE WATER MODEL	NRL-R-1010;PFR/TC/P(88)265	PARDY A	1987
12519.pdf	THE HISTORY OF SIGNALS IN THE PFR CLAD FAILURE DETECTION SYSTEMS DURING RUNS 13A AND 13B	PFR/TC/P(88)261;DFMC/P(88)18	MACLEOD DJ	1988
13649.pdf	PFR ABSORBER ROD GUIDE TUBE BPD PIPE BUSH PULL OUT TEST	PFR/FEDWP/P(83)957	RIDEALGH F	1983
14874.pdf	CLAD FAILURE DETECTION AND THE SINGLE SUBASSEMBLY FAULT A PROPOSED PROGRAMME OF WORK	FRSWG/SAFSG/P(87)7;FRSSG/N316	LENNOX TA	
17449.pdf	PRELIMINARY INVESTIGATION OF METHODS OF LOCATING FAILED FUEL SUB ASSEMBLIES IN CDFR AFTER SHUTDOWN	ND-M-1838	CARTWRIGHT DK	1982
17510.pdf	OPERATIONAL EXPERIENCE OF BPD SYSTEMS WITH IMPLICATIONS FOR CDFR DESIGN	TC/P(80)401	SHAW SG	1980
17925.pdf	SPECIAL SAFETY EXPERIMENTS IN PFR	FRSWG/SAFSG/P (88)25	TAIT D; LORD DJ; LENNOX TA	
18527.pdf	INITIAL DATA ASSUMPTIONS FOR THE FAILED FUEL DETECTION LOOP (FFDL) - DESIGN STUDY		LENNOX T	1991
19214.pdf	CDFR BURST PIN DETECTION (BPD) SYSTEM SAMPLE SELECTOR. PROPOSAL FOR A WATER TEST MODEL. (PROJECT NO. C93/40/41)	CBG/P(82)454; RES.INT. 2670;	BOUABDALLAH S;	1982
19379.pdf	CFR BPD SYSTEM SAMPLE SELECTOR PROPOSAL FOR A WATER TEST MODEL	CBG/P(82)434	BARROWMAN GR	1982
20150.pdf	FAILED FUEL DETECTION CALIBRATION TECHNIQUES	FRDCC/SISG/P(85)19	LENNOX TA	1985
20214.pdf	A PROPOSED FAST REACTOR PROGRAMME FOR FAILED FUEL DETECTION AND LOCATION	FRSWG/SAFSG/P (84)9	LENNOX TA; GREGORY CV	1984

Reference No.	Title	Location	Authors	Year
20640.pdf	FAILED FUEL DETECTION CALIBRATION TECHNIQUES	FRSWG/SISG/P(85)19	LENNOX TA;WEBSTER R	1985
21921.pdf	RUNNING ON WITH EXPOSED FUEL FAILURES IN PFR	PFR/SWP/P(84)39;OETD TEC NOTE NO 891	LENNOX TA	1984
22051.pdf	PFR OPERATING LIMITS WITH FAILED FUEL	PFR/SWP/P(85)33	GREGORY CV	1985
22051.pdf	PFR OPERATING LIMITS WITH FAILED FUEL	PFR/SWP/P(85)33	GREGORY CV	1985
22055.pdf	A PROPOSED ALARM LEVEL FOR THE BULK IHX DN SIGNAL FROM A DMSA IN THE PFR CORE	PFR/SWP/P(85)37	GREGORY CV	
23661.pdf	ASSESSMENT OF THE INITIAL SIGNALS FORM THE PFR MARK 4 BPD LOCATION LOOP	ND-M-1845	CATWRIGHT DK;POVEY V	1983
23662.pdf	A COMPARISON OF THE PREDICTED PERFORMANCE OF THE PFR DELAYED AND FISSION GAS LOCATION SYSTEMS	TRG-M-6934	CARTWRIGHT DK;DIGGLE WR	
24108.pdf	THE PROTOTYPE FAST REACTOR FAILED FUEL PIN DETECTION SYSTEMS	ND-M-3477	MASON L;TREVILLION EA	1987
24699.pdf	FLUIDIC BPD SYSTEM FOR FAST REACTORS SINGLE-STAGE TESTS AND CIRCUIT ANALYSIS	HIC 288	TIPPETTS JR	
25893.pdf	FUEL ELEMENT FAILURE DATA	FRDC/MWP/P(81)1353;FRDC/MWP/FIXSG/P(81)189	TAYLOR AF;NUTTER NR	1981
26175.pdf	THE MAGNITUDE AND MECHANISM OF DN EMISSION FROM FAST REACTOR FUEL PINS WITH PARTICULAR REFERENCE TO ENDURANCE FAILURES	FFWG/P(82)12	CARTWRIGHT DK;DIGGLE WR	1982
26683.pdf	PFR BREEDER TRANSIT TIMES FROM CORE OUTLET TO THE ABOVE CORE STRUCTURE	ACSCM/P(76)9	WINN RW	
29868.pdf	DETERMINATION OF SAMPLE TRANSIT TIMES FOR THE PFR IHX BULK MONITOR BURST PIN DETECTION SYSTEM-USING A 1/5 SCALE WATER MODEL	ND-M-1051 (R)	FRANCE J;WINN WR;ELLABY GM	1980
29869.pdf	PRELIMINARY EXPERIMENTS ON THE PFR INTERMEDIATE HEAT EXCHANGER BURST PIN DETECTION SYSTEM-USING A 1/5 SCALE WATER MODEL	ND-M-1681 (R)	ROBINSON RGJ; PARDY A; HILES RIW; TAYLOR AF	1982
31599.pdf	FURTHER CONSIDERATION OF MAXIMUM FLAW SIZES IN PFR CORE SUPPORT STRUCTURE (CSS) AT BEGINNING OF LIFE	PFR/SIAG/P(91)56	PICKER C	1991

Table 11: Wood Report List: Decay Heat Rejection System

Reference No.	Title	Location	Authors	Year
1	**Reliability of Decay Heat Rejection**			
00708.pdf	INVESTIGATION OF AC FAILURE TRANSIENT WHEN ONLY TWO DHR'S OPERATE WITH A SINGLE FAN EACH	TN/P(84)682	THOMSON AF	1984
01387.pdf	AVAILABILITY OF THE PFR THERMAL SYPHON EDHR SYSTEM FOLLOWING LOSS OF GRID SUPPLIES	TECH.MEMO P&S(R)927;PFR/SWP/P(79)10	BLAND PS	
01431.pdf	RECOMMENDED CONSTRAINTS ON THE USE OF PFR DECAY HEAT REMOVAL LOOPS	PFR/SWP/P(78)45;TN/P(78)252	WILKES DJ	
03959.pdf	RELIABILITY OF THE PFR DECAY HEAT REJECTION SYSTEM USING STEAM DUMPING	PFR/SWP/P(71)38	MARSHALL F	
03960.pdf	RELIABILITY ASSESSMENT OF THERMAL-SYPHON DECAY HEAT REJECTION SYSTEM FOR PFR	SRD M 5	DAVIES FM	
03961.pdf	PFR SAFETY ASSESSMENT - DECAY HEAT REJECTION	SRD M 16	DAVIES FM	
2	**Metallurgical Examination of Failed AHX Pulled Tees**			
00500.pdf	METALLURGICAL EXAMINATION OF FAILURES ON INLET HEADER PULLED TEE REGIONS OF THERMAL SYPHON C, JULY 1981 AND SEPTEMBER 1982	NDM-2339	FRASER AS;ROGER RJC;IRONS H	1983
01406.pdf	METALLURGICAL EXAMINATION OF A FAILURE ON PFR THERMAL SYPHON B	PFR/SWP/P(75)23;REV.1	FRASER AS;PORTER JD;ROGER RJC	
03016.pdf	METALLURGICAL EXAMINATION OF DEFECTS DETECTED ON THE SURFACES OF PULLED TEE 20 FROM THE OUTLET HEADER OF PFR DHR THERMAL SYPHON A, JULY/AUGUST 84	NDM-2950;PFR/SWP/P(84)73;FRDCC/MWG/P(84)89;FRDCC/MWG/MPSG/P(84)52	FRASER AS ;IRONS H;ROGER RJC;MACDONALD GJ	1985
3	**Diagnosis of NaK Flow Conditions in AHX Tubes**			
01400.pdf	TESTS ON THERMAL SYPHON C NAK/AIR HEAT EXCHANGER	OPS NOTE 732	HERRICK AR;MCCRINDLE D	
4	**The Condition of the AHX Tube Cleats**			
00618.pdf	A QUALITATIVE ASSESSMENT OF FLOW INDUCED VIBRATION OF THE PFR NaK-AIR HEAT EXCHANGER WITH FAILED BRACING CLEATS	NDM-2126;DRAFT;OC/P(83)50	COLLINSON AE;FRANCE NJ	1983
01408.pdf	VISUAL INSPECTION OF NAK AIR HEAT EXCHANGER IN PFR THERMAL SYPHON LOOP "A" OCTOBER 1983	OETD TECH NOTE.789	CASTLE P	1983
5	**Measurements of Flow on the Air Side of the AHXs**			

Reference No.	Title	Location	Authors	Year
03822.pdf	A PFR DECAY HEAT (LOOP A) REMOVAL TEST AT 250C PRIMARY POOL TEMPERATURE	PFR EXPERIMENTAL RESULTS SHEET NO.132;OETD TECH NOTE NO.773	CROWE DS; DICKSON AK; DISBURY WH; GRAHAM DK; LORD DJ; SUTHERLAND AJ	1983
03826.pdf	PERFORMANCE CHECKS ON PFR DECAY HEAT REMOVAL LOOPS AT END OF RUN 7	PFR EXPERIMENTAL RESULTS SHEET NO.136;OETD TECH NOTE NO.790	CROWE DS	1983
6	**Overall DHR Loop Performance Tests**			
01401.pdf	REPORT ON THE PERFORMANCE CHECKS ON PFR DECAY HEAT REMOVAL LOOPS (PROJECT NO.RES/C87/41)	RES.INT.2697;PFR/SWP/P(82)19	AUSTIN NM; BLAND PS	1983
01419.pdf	PERFORMANCE TESTS ON PFR DECAY HEAT REMOVAL SYSTEM - CIRCUITS B AND C UPTO 400 C PRIMARY SODIUM TEMPERATURE	FRD/5643/DN179	SEDDON F; GRUNDY I	
7	**Laboratory AHX Tests**			
02067.pdf	SUMMARY OF THE OVERALL ANALYSIS OF SEVEN CDFR AND PFR AIR-COOLED HEAT EXCHANGER FINNED TUBE BUNDLES TESTED AT NEL	FREWG/P(85)082	LEISHMAN P	1985
02543.pdf	NEL AIR-COOLED HEAT EXCHANGER TESTS - SUMMARY OF CONCLUSIONS	FREWG/P(85)098	LEISHMAN P	1985
8	**Gas Locks in Horizontal Pipe Runs**			
00983.pdf	STATUS OF THE AIR HEAT EXCHANGERS DECAY HEAT REMOVAL SYSTEM, PFR DOUNREAY	OC/P(84)109	WEBB J	1984

Table 12: Wood Report List: Stream Generators

Reference No.	Title	Location	Authors	Year
1	**Metallurgical examination of evaporator welds**			
00445.pdf	METALLURGICAL EXAMINATION OF TUBE TO TUBE PLATE WELD NO. 408 FROM PFR EVAPORATOR CELL 2	NDM-1488;TF/P(81)459;MWP/CCCSG/P(81)897;MWP/P(81)997	KIRKLAND GR;DAVIES ER;HURLEY JC	1981
00452.pdf	METALLURGICAL CHARACTERISATION OF PFR EVAPORATOR TYPE WELD	NDM-1610	LINEKAR GAB;HUGHES B;ROGER RJC	1981
00454.pdf	THE METALLURGICAL EXAMINATION OF TUBE-TO-TUBEPLATE WELD NUMBER 280, REMOVED FROM PFR EVAPORATOR TUBE BUNDLE WU2 IN 1977	NDM-1696	FRASER AS;ROGER RJC;IRONS H	1983
00457.pdf	THE METALLURGICAL EXAMINATION OF TUBE-TO-TUBEPLATE WELD NUMBER 101 FROM PFR EVAPORATOR WU3 IN CIRCUIT 2	NDM-1726;CCCSG/P(82)948;MWP/P(82)1048	YATES G;IRONS H	1982
00459.pdf	PFR SPARE EVAPORATOR UNIT - METALLURGICAL CHARACTERISATION OF A TEST WELD	NDM-1735	LINEKAR GAB;HUGHES B	1981
00771.pdf	METALLURGICAL EXAMINATION OF TUBE TO TUBE PLATE WELD NO. 480 FROM PFR EVAPORATOR WORKS UNIT 3 CELL 2	NDR-741; OC/P(82)4;MWP/P(82)1030;CCCSG/P(82)930;FIXSG/P(84)237	KIRKLAND GR;DAVIES ER;HURLEY JC	1982
00776.pdf	METALLURGICAL EXAMINATION OF TUBE PLATE WELD NO. 365 FROM PFR EVAPORATOR UNIT 3 CELL 2	NDR-812	KIRKLAND GR;DAVIES ER;HURLEY JC	1982
02199.pdf	PROVISIONAL RESULTS FROM THE METALLOGRAPHIC EXAMINATION OF WELD 235 FROM PFR EVAPORATOR, WORKS UNIT 3, CIRCUIT 2	FTD EVAP.PROV.REPORT NO.16	JAMES G;CHATWIN WH	1982
06185.pdf	A NEUTRON DIFFRACTION STUDY OF THE RESIDUAL STRESS DISTRIBUTION IN PFR EVAPORATOR FUSION WELDS	AERE-R-11978; FRDCC/MWG(85)P233;FRDCC/MWG/FSG/(85)P42;	ALLEN AJ;BOWEN PH;HUTCHINGS MT	1986
2	**Identification of leaking welds from the water side**			
00453.pdf	THE INSPECTION OF PFR EVAPORATOR WORKS UNIT 3 IN CIRCUIT 2, 4 TO 10 JULY 1981	NDM-1613	FRASER AS;LEYLAND KS;SHARP RKY;ROGER RJC	1981
00455.pdf	THE INSPECTION OF PFR EVAPORATOR WU3 IN CIRCUIT 2 AUGUST 1981	NDM-17188	FRASER AS;ROGER RJC;SHIPLEY DF	1981

Reference No.	Title	Location	Authors	Year
00458.pdf	THE VISUAL INSPECTION OF PFR EVAPORATOR WU1 IN CIRCUIT 1 SEPTEMBER 1981	NDM-1734	FRASER AS; MCDONALD J; LEYLAND KS; TELFORD D	1981
00486.pdf	VISUAL INSPECTIONS CARRIED OUT ON THE PFR EVAPORATOR TUBE BUNDLE WU1 IN THE CIRCUIT 1 EVAPORATOR, PLANT EVENT NUMBER 33, OCTOBER 1982	NDM-2073;MWP/P(84)1524;MWP/FIXSG/P(84)238	FRASER AS; ROGER RJC; MACDONALD J	1983
05009.pdf	RNL RESULTS FOR THE ULTRASONIC EXAMINATION OF PFR EVAPORATOR WU1 TUBE TO TUBEPLATE WELDS BETWEEN SEPTEMBER 1981 AND JUNE 1982	NDR 965	HUDGELL RJ; BIRCHALL PD; TURNER NA	1983
07876.pdf	THE INSPECTION OF PFR EVAPORATOR WORKS UNIT 3 IN CIRCUIT 2 16 AND 17 FEBRUARY 1981	ND-M-1519(D)	FRASER AS; ROGER RJC; McKEAGUE R; LEYLAND KS	1981
09756.pdf	ULTRASONIC TECHNIQUES FOR THE ISI OF PFR EVAPORATOR TUBE TO TUBE PLATE WELDS	ND-R-886(R)	HUDGELL RJ;BIRCHALL PD	1982
3	**The evolution of a leak in WU1**			
04823.pdf	PFR EVAPORATOR 1 INLET LEAK ON 1490479	PFR/TF/P(79)370;EST/P(79)447	CURRIE R	
4	**Pure water stress-corrosion cracking**			
04933.pdf	THOUGHTS ON THE RAMIFICATIONS OF THE FIRST LEAK IN PFR EVAPORATOR TUBE BUNDLE 2 ORIGINALLY IN CIRCUIT 2.	NDM 1125	SMEDLEY JA;HALE JC;FOLEY J;BUXTON K	1980
04934.pdf	REVIEW OF EARLY TUBE TO TUBEPLATE LEAK BEHAVIOUR IN THE PFR EVAPORATORS, OCT74-FEB77.	NDM 1128	SMEDLEY JA	1980
5	**Shot-peening**			
08464.pdf	PFR EVAPORATOR UNIT WELDS - METALLURGICAL CHARACTERISATION OF A SHOT-PEENED TEST WELD	ND-M-1671	LINEKAR GAB; HUGHES B;IRONS HW	1981
6	**The condition of the magnetite layer in evaporator tubes**			
00416.pdf	THE VISUAL EXAMINATION OF MAGNETITE CONDITION ON SELECTED TUBES OF PFR EVAPORATOR WU3 AT PEN 28, AND THEIR BORE CONDITION AFTER CHEMICAL CLEANING	NDM-1963;MWP/P(84)1535;MWP/FIXSG/P(84)249	FRASER AS;ROGER RJC;PORTER JD	1983

Reference No.	Title	Location	Authors	Year
00464b.pdf	PFR EVAPORATOR WORKS UNIT 3 TUBE BORE MEASUREMENTS TO ASSESS THE LEAKAGE OF INLET SODIUM INTO THE INLET HEADER REGION (JANUARY-FEBRUARY 1982)	NDM-1812; MWP/P(84)1532; MWP/FIXSG/P(84)246	LEYLAND KS;GUNN T	1983
01044.pdf	WATERSIDE AND SODIUM SIDE CORROSION OF PFR EVAPORATOR TUBING REMOVED FROM WORKS UNITS 1 AND 3 BETWEEN MAY 1981 AND NOVEMBER 1982 - WATERSIDE CORROSION SODIUMSIDE CORROSION - STEAM GENERATORS	AERE-R-11134;EDCC/P(83)96	ASHMORE CB; TOMLINSON L; HURDUS MH	1984
02229.pdf	PFR EVAPORATORS : ON-LOAD CORROSION BY ACID SULPHATE AND SUGGESTED METHODS OF PREVENTION	EDCC/P(84)108	TOMLINSON L	1984
02302.pdf	EXAMINTION OF TUBE 470/1470 REMOVED FROM PFR EVAPORATOR WORKS UNIT 2 PART 2 INVESTIGATION OF OXIDE DEPOSTION AND METAL CORROSION ON THE WATER AND SODIUM SIDES	AERE-R-11323; FRDCC/MWG/P(84)26;MWG/CSG/P(84)8	ASHMORE CB; TOMLINSON L; HURDUS MH	1984
02528.pdf	EXAMINTION OF TUBE 470/1470 REMOVED FROM PFR EVAPORATOR WORKS UNIT 2. PART 1. GENERAL SURVEY OF THE TUBE SURFACE AND INVESTIGATION OF SPALLED OXIDE REGIONS - WATERSIDE CORROSION - STEAM GENERATORS 2.1/4CR1MO	AERE-R-11374;FRDCC/MWG/CSG/P(84)10;FRDCC/MWG/P(84)28	HURDUS MH; TOMLINSON L; ASHMORE CB	1984
7	**Sodium flow-patterns affecting magnetite formation+B64**			
00466.pdf	A REPORT ON THE SODIUM LEVEL MEASUREMENTS MADE IN THE SPARE INLET-SIDE POCKET OF EVAPORATOR WORKS UNIT 1 (MARCH 1982)	NDM-1818	JAMES PR;LEYLAND KS;SMEDLEY JA	1982
01564.pdf	AN INVESTIGATION INTO THE JETTING OF SODIUM ONTO THE UNDERSIDE OF THE TUBEPLATE OF PFR EVAPORATOR WORKS UNIT 3	NDM-2597	MACKENZIE E M	1984
02182.pdf	THE SODIUM-SIDE THERMAL HYDRAULICS OF THE PFR EVAPORATORS AND THE WATER-SIDE DEPOSITION OF MAGNETITE	PTWG/P(83)18	WEBSTER R;DAWSON C;FOLEY J	
11452.pdf	TEMPERATURE DATA FROM A PFR EVAPORATOR DURING PLANT TRIP CONDITIONS	ND-M-1176	INNES NJ; WIDDOWSON IR	1980
8	**Fretting and galling of evaporator tubes**			
00004.pdf	ULTRASONIC WALL THICKNESS MEASUREMENT OF THE PFR EVAPORATOR TUBES	NDR 808	HUDGELL RJ	
00607.pdf	TUBE GALLING ON EVAPORATOR WU2	OC/P(84)135	BAINBRIDGE H	1984
01054.pdf	THE DETECTION OF FRETTING WEAR FOR PFR SGU TUBING	EDCC/P(83)45	GRAY BS; HUDGELL RJ;NETTLEY PT	1983
01057.pdf	NON-DESTRUCTIVE EXAMINATION OF EVAPORATOR TUBES FOR FRETTING WEAR AND GALLING DAMAGE	EDCC/P(83)40	WALFORD JD	1983

Reference No.	Title	Location	Authors	Year
02016.pdf	TUBE INSPECTION OF WORKS UNIT 3 - JULY-AUGUST 1982	EDCC/P(82)14	TELFORD D;GREGORY CV	
04772.pdf	EXAMINATION OF THE SODIUM SIDE OF PFR EVAPORATOR TUBES	ND-R-1161(R);FRDCC/MWG/P(85)200	LONGSON B	1985
9	**Tubeplate washing**			
00081.pdf	THE VISUAL EXAMINATION OF THE UNDER-SURFACE OF THE TUBEPLATE ON PFR EVAPORATOR TUBE BUNDLE WU3, THROUGH THE TUBE 299 ACCESS HOLE, MARCH 1982	NDM 1890;MWP/P(84)2002;CCCSG/P(84)1002;CEWP/P(84)423	FRASER AS; ROGER RJC	1982
00468.pdf	THE EXAMINTION OF THE UNDER SURFACE OF THE TUBEPLATE OF PFR EVAPORATOR TUBE BUNDLE WU3, OBSERVED THROUGH THE TUBE 172,241 AND 365 SAMPLE HOLES,JUNE 1981	NDM-1853	FRASER AS; ROGER RJS; MACDONALD M	1982
00470.pdf	THE VISUAL INSPECTION OF THE UNDER-SURFACE OF THE TUBEPLATE OF PFR EVAPORATOR TUBE BUNDLE WU3 THROUGH VENT LINE ACCESS HOLES, AUGUST 1981	NDM-1873	FRASER AS; ROGER RJC; ROBERTSON D	1982
00471.pdf	THE VISUAL EXAMINATION OF THE UNDER-SURFACE OF THE TUBEPLATE OF PFR EVAPORATOR TUBE BUNDLE WU1 THROUGH THE VENT LINE ACCESS HOLES, SEPTEMBER 1981	NDM-1874	FRASER AS; ROGER RJC; TREVILLION EA	1982
00472.pdf	THE VISUAL EXAMINATION OF THE INSIDE OF THE INLET HEADER REGION ON THE PFR CIRCUIT 2 EVAPORATOR CONTAINING TUBE BUNDLE WU3, THROUGH TUBE 487 ACCESS HOLE, FEBRUARY 1982	NDM-1887	FRASER AS; ROGER RJC; MACDONALD J; LEYLAND KS	1982
00473.pdf	THE VISUAL EXAMINATION OF LOCALISED AREAS OF THE UNDER-SURFACE OF THE TUBEPLATE OF PFR EVAPORATOR TUBE BUNDLE WU3, THROUGH THE TUBE 235, 3 AND 396 ACCESS HOLES, APRIL 1982	NDM-1891	FRASER AS; ROGER RJC	1982
00474.pdf	THE EXAMINTION OF THE UNDER SURFACE OF THE TUBEPLATE ON PFR EVAPORATOR TUBE BUNDLE WU1, OBSERVED THROUGH THE TUBE 1127 ACCESS HOLE MARCH 1982	NDM-1918	FRASER AS; ROGER RJC	1982
00475.pdf	THE VISUAL EXAMINTION OF LOCALISED AREAS OF THE UNDER-SURFACE OF THE TUBEPLATE OF PFR EVAPORATOR TUBE BUNDLE WU3, THROUGH THE TUBE 255,268,374, AND 1261 ACCESS HOLES	NDM-1934	FRASER AS; ROGER RJC; MACDONALD J	1982
00477.pdf	THE VISUAL EXAMINATION OF LOCALISED AREA OF THE UNDER-SURFACE OF THE TUBEPLATE ON PFR EVAPORATOR TUBE BUNDLE WU1, THROUGH THE TUBE 16 ACCESS HOLE , JUNE 1982	NDM-1948	FRASER AS;ROGER RJC;MCDONALD J	1982

Reference No.	Title	Location	Authors	Year
00487.pdf	THE VISUAL INSPECTION OF THE UNDER-SURFACE OF THE TUBEPLATE OF PFR EVAPORATOR TUBE BUNDLE WU1 THROUGH THE TUBE 6 AND 122 ACCESS HOLES, NOVEMBER/DECEMBER 1982	NDM-2074	FRASER AS; ROGER RJC;MACDONALD J	1983
02202.pdf	EXAMINATION OF A LOCALISED AREA OF THE UNDERSURFACE OF THE TUBEPLATE OF WU1 THROUGH THE TUBE 1127 ACCESS HOLE	FTD.EVAP.PROV. REPORT NO.12	FRASER AS;ROGER RJC	1982
02203.pdf	EXAMINATION OF LOCALISED AREAS OF THE UNDERSURFACE OF THE TUBEPLATE OF WU3 THROUGH THE TUBE 235, 3 AND 296 ACCESS HOLES	FTD.EVAP.PROV. REPORT NO.11	FRASER AS;ROGER RJC	1982
02318.pdf	TESTS ON THE EFFICACY OF REMOVAL OF SODIUM HYDROXIDE FROM CRACKS IN STEAM GENERATOR MATERIAL BY SODIUM WASHING	NDM-2487;FRDCC/MWG/P(85)34;FRDCC/MWG/P(85)114	BUXTON K;MACKINNON DJ	1983
10	**Rig tests to replicate after-leak conditions on the sodium side of evaporator welds**			
00469.pdf	THE METALLURGICAL EXAMINTION OF WELD TEST BLOCK NDT 33 AFTER EXPOSURE IN THE SMALL WATER LEAK RIG	NDM-1858	FRASER AS;IRONS H;PORTER JD;ROGER RJC	1982
00740.pdf	THE WELD CRACKING TESTS - A SERIES OF TESTS IN THE DOUNREAY SMALL WATER LEAK RIG TO INVESTIGATE SODIUM-SIDE INITIATION OF TUBE-TO-TUBEPLATE WELD FAILURE MECHANISMS	CCCSG/P(82)938;MWP/P(82)1038	EDGE DM	
11	**The state of the evaporator welds**			
00449.pdf	TUBES PLUGGED OR SLEEVED IN THE PFR EVAPORATOR TUBE BUNDLES BECAUSE OF PITTING, LEAKAGE OR OTHER DEFECTS BETWEEN SEPTEMBER 1973 AND DECEMBER 1980	NDM-1517	FRASER AS;ROGER RJC;MCKEAGUE R	1981
00537.pdf	THE APPLICATION OF THE DNE INSPECTION PROCEDURE TO THE IDENTIFICATION OF DEFECTS IN PFR EVAPORATOR BUNDLES TUBE TO TUBE PLATE WELDS IN PREPARATION FOR CARRYING OUT SHOT-PEENING OPERATIONS	NDM-1365;DRAFT;TF/P(80)443	MCKEAGUE R;LEYLAND KS	1980
00579.pdf	RNL RESULTS FOR THE ULTRASONIC EXAMINATION OF PFR EVAPORATOR WU 1 TUBE TO TUBEPLATE WELDS BETWEEN SEPTEMBER 1981 AND JUNE 1982	NDR-965	HUDGELL C	1983
00583a.docx	PFR EVAPORATOR WU 1 : A FINAL REPORT ON THE RESULTS OF THE RNL ULTRASONIC ISI CARRIED OUT AT PLANT EVENT NUMBER 33, NOVEMBER 1982	NDR-1008;MWP/P(84)1527;MWP/FIXSG/P(84)241;MWP/MPSG/P(84)427;EDCC/P(84)110	LEYLAND KS;HUDGELL RJ;WILLETTS AJ;BIRCHALL PD;TURNER NA	1983
01063.pdf	EVAPORATOR WU1 RE-INSPECTION: JUNE 1982	EDCC/P(82)10	WALFORD JD	1982

Reference No.	Title	Location	Authors	Year
01064.pdf	WORKS UNIT 3 - COMPARISON OF DNE ULTRASONIC INSPECTION RESULTS - AN UPDATE	EDCC/P(82)9	LEYLAND KS	1982
02019.pdf	SUMMARY REPORT ON THE ULTRASONIC EXAMINATION OF THE TUBE TO TUBEPLATE WELDS IN EVAPORATOR WORKS UNIT 1	EDCC/P(82)3	NETTLEY PT;HUDGELL RJ;GRAY BS	1982
02025.pdf	NON DESTRUCTIVE TESTING OF THE TUBE TO TUBE PLATE WELDS IN PFR EVAPORATORS	EDCC/P(82)1	GRAY BS;NETTLEY PT	1982
02726.pdf	RNL RESULTS FOR THE ULTRASONIC EXAMINATION OF THE TUBE-TO-TUBEPLATE WELDS IN PFR WU2 EVAPORATOR (JULY-SEPTEMBER 1982)	NDR-1122	BIRCHALL PD;TURNER NA;HUDGELL RJ	1985
12	**Sleeving of evaporator welds**			
00481.pdf	THE EXAMINATION OF PFR EVAPORATOR SLEEVE 16 REMOVED FROM WORKS UNIT 1	NDM-1998	YATES G;IRONS H	1982
00498.pdf	IN-SERVICE INSPECTION OF THE PFR EVAPORATOR TUBE-TO-TUBEPLATE WELDS AFTER SLEEVING	NDM-2300;MWP/P(84)1530;MWP/FIXSG/P(84)244	PEAT TS;TELFORD DW	1983
00768.pdf	METALLURGICAL EXAMINATION OF PFR BRAZED STEAM TUBE ASSEMBLY (TUBE 365, EVAPORATOR CELL 2)	NDR-722;RWP/P(83)140;MWP/P(83)1503;FIXSG/P(83)234	JOHNSON R	1983
00851.pdf	ENDORSEMENT OF PFR EVAPORATOR SLEEVES	EDCC/P(83)30;RWP/P(83)119;MWP/P(83)1494;MWP/FIXSG/P(83)227	NETTLEY PT	1983
01560.pdf	THE DEVELOPMENT OF EXPLOSIVE WELDING TECHNOLOGY FOR PFR	ND-R-1166(R);FRPDC/P(84)66;FRDCC/MWG/P(84)40;FRDCC/MWG/FSG/P(84)3;FRDCC/MWG/MPSG/P(84)29;FRDCC/MWG/I	HAMER AN	1984
01732.pdf	EXPLOSIVE WELDING OF TUBE-TUBEPLATE JOINTS FOR PFR EVAPORATOR TUBE BUNDLES	MWP/P(79)881;FIXSG/P(79)141	JACKSON PW;COMBE G;FRYER D;GRAHAM BL;SHAW MP	
13	**Review of evaporator leaks**			
00904.pdf	A REVIEW OF THE PFR EVAPORATOR TUBE TO TUBEPLATE WELD PROBLEM	OC/P(82)24;PODG/P(82)34;MWP/P(82)1046;CCCSG/P(82)946	BROOMFIELD AM	1982
14	**The destructive examination of WU2**			
08353.pdf	PFR WORKS UNIT 2 EVAPORATOR BUNDLE OPERATING HISTORY	OETD/TN 1359	HAWKSLEY C	1986

Reference No.	Title	Location	Authors	Year
08360.pdf	PFR EVAPORATOR TUBE BUNDLE WU2 CLEANING AND DISMANTLING:STAGE 1	OETD/TN 1411	HAWKSLEY C	1987
15	**Hydrogen detection systems**			
00536.pdf	ANALYSIS OF H2D SIGNALS LEADING TO IDENTIFICATION OF AN INLET GAS SPACE LEAK IN PFR EVAPORATOR IN CIRCUIT 2 - NOVEMBER 1980	NDM-1368;DRAFT;TF/P(80)444	BELL AC;EDGE DM	1980
01144.pdf	EXAMINATION OF THE HYDROGEN INGRESS RATES TO CIRCUIT 1 SODIUM DURING START-UP PERIUODS FROM 1980 TO 1983	EDCC/P(84)117; OC/P(84)113;OPS/N782;MWG/P(84)2;CSG/P(84)1; CEWP/P(84)431	WALLACE DM	
01192.pdf	IMPROVEMENTS TO THE AVAILABILITY OF THE STEAM GENERATOR UNIT GAS PHASE HYDROGEN DETECTION SYSTEM	OC/P(82)29	WILLIAMS R	1982
16	**Crack growth and leak-before-break**			
00074.pdf	THE CRACKING OF EVAPORATOR WELDS IN PFR,MWG,CSG	EDCC/P(83)51;NDM 2199;CCCSG/P(83)985;MWP/P(83)1085;CEWP/P(83)412	VOICE E;EDGE DM;LINEKAR GA	1983
00074.pdf	THE CRACKING OF EVAPORATOR WELDS IN PFR,MWG,CSG	EDCC/P(83)51;NDM 2199;CCCSG/P(83)985;MWP/P(83)1085;CEWP/P(83)412	VOICE E;EDGE DM;LINEKAR GA	1983
00460.pdf	PFR EVAPORATOR WELD FAILURES: STUDIES OF CRACKS INCLUDING CROWTH RATE, MORPHOLOGY, CORROSION AND SPACE-TIME DISTRIBUTION	NDM-1753	BUTLER JK	1982
00476.pdf	THE LEAK EVENT IN EVAPORATOR WU2 IN CIRCUIT 3, 8-19 NOVEMBER 1981	NDM-1947	EDGE DM;BELL AC;CURRIE R	1982
01061.pdf	NOTES ON EVAPORATOR WELD CORROSION AND EMBRITTLEMENT,MWG,CSG	EDCC/P(82)19	BUTLER JK	1982
02204.pdf	A REVIEW OF EVAPORATOR WU1 - JUNE 1982	FTD.EVAP.PROV. REPORT NO.10	BUTLER JK	1982
02206.pdf	A REVIEW OF THE PFR SPARE EVAPORATOR'S FITNESS FOR SERVICES	FTD.EVAP.PROV. REPORT NO.5	BUTLER KJ	1982
02209.pdf	STRESSES IN HEAT EXCHANGER WELDS	FTD.EVAP.PROV. REPORT NO. 2	BUTLER JK	1982
02210.pdf	CRACK GROWTH RATE INFERRED FROM FAILURES IN SERVICE	FTD.EVAP.PROV. REPORT NO. 1	BUTLER JK	1982
25471.pdf	LEAK BEFORE BREAK IN EVAPORATOR TUBE/TUBEPLATE WELD FAILURE	PFR/SWP/P(81)55	BUTLER KJ;BELL AC	
17	**Local post-weld heat treatment**			
00074.pdf	THE CRACKING OF EVAPORATOR WELDS IN PFR,MWG,CSG	EDCC/P(83)51; NDM 2199; CCCSG/P(83)985;MWP/P(83)1085;CEWP/P(83)412	VOICE E;EDGE DM;LINEKAR GA	1983

Reference No.	Title	Location	Authors	Year
01580.pdf	LOCAL POST-WELD HEAT TREATMENT OF WU2 TUBE-TO-TUBEPLATE WELDS	NDM-2582	YATES G;ARMSTRONG M;HUGHES B;IRONS HW;ROGER RJC	1984
02722.pdf	THE STRESS RELIEF OF PFRE EVAPORATOR FUSION WELDS BY RF INDUCTION HEATING	AERE-R-11470; FRDCC/MWG/P(84)63;MWG/FSG/P(84)7	CROAD JP;AYRES CF;EDWARDS BC	1984
18	**Rattling tubes and tie-rods**			
24467.pdf	EXPERIMENT TO INVESTIGATE NOISE IN PFR EVAPORATORS 1&2 AT HIGH PUMP SPEED (SAPEX 627)	RS/SWP/P(93)8; PE1/7531;PE1/7730	HENDERSON JDC	1993
25452.pdf	TUBE EXCITATION BY JOGGLE-GAP FLOW IN PFR EVAPORATORS	FMD/D(92)104	COLLINSON AE	1992
25453.pdf	ANALYSIS AND INTERPRETATION OF A ACOUSTIC SIGNALS ROM PFR CELL 2 EVAPORATOR FEBRUARY/APRIL 1992	PFR/TN 1083	ROWLEY R;RICHARDSON K;COLLINSON AE	1992
26662.pdf	PFR EVAPORATOR TUBE BUNDLE NOISE PROBLEMS	PE1/7799;PFR/TN 1123	COLLINSON A	1993
19	**Superheater 2 tubeplate cracks**			
25833.pdf	A REPEAT INSPECTION OF THE DAMAGED REGION OF THE INNER TOROID OF PFR SUPERHEATER 2 NOVEMBER 1979	FRDC/MWP/P(80)1266;FRDC/MWP/FIXSG/P(80)169;ND-M-1252	BIRCHALL PD;HUDGELL RJ;HALE JC;SARGENT TH	1980
20	**The safety case for operating superheater 2 in its cracked state**			
25810.pdf	ADDITIONAL RESULTS TO THE RE ANALYSIS OF THE CRITICAL DEFECT SIZE AND CRACK GROWTH RATE OF THE PFR SUPERHEATER TUBEPLATE	PFR/SWP/P(80)7; TN/P(80)367	GREEN D	1980
25811.pdf	THERMAL ANALYSIS OF SUPERHEATER NUMBER 2 TUBEPLATE DURING A TRIP TRANSIENT	PFR/SWP/P(80)7; TN/P(80)367	DIXON M	1980
27581.pdf	OPERATION OF PFR SUPERHEATER 2 WITH A CRACKED TUBE PLATE FUTURE OPERATION OF ALL PFR STAINLESS STEEL; STEAM GENERATOR	PFR/SWP/P(76)22;PFR/TF/P(76)142	SMEDLEY JA	
27590.pdf	HIGH PRESSURE TESTING & OPERATION OF PFR SUPERHEATER NO 2	PFR/SWP/P(75)36	ABLITT JF	
27592.pdf	RECONSIDERATION OF THE SAFETY CASE TO RUN PFR SUPERHEATERS WITH CRACKED TUBE PLATES FOLLOWING THE FINDING OF CRACKING IN PFR REHEATER 3	PFR/SWP/P(76)48	SMEDLEY JA	
27597.pdf	REVIEW OF THE SAFETY CASE FOR CONTINUED OPERATION OF PFR SUPERHEATER 2 FOLLOWING TUBE PLATE EXAMINATION OCTOBER/NOVEMBER 1979	PFR/SWP/P(79)73	SMEDLEY JA	
21	**RTB design tests**			

Reference No.	Title	Location	Authors	Year
26959.pdf	PFR REPLACEMENT SUPERHEATER TUBE BUNDLES HALF SCALE AIR MODELS OF INLET/BEND AND OUTLET REGIONS PRESSURE LOSS MIXING AND FLOW DISTRIBUTION TESTS ON THE OUTLET REGION AIR MODEL FITTED WITH BUSHED TYPE TUBE SUPPORT GRIDS	RD/P/996	LITTLE AJ	1981
22	**High temperatures caused by the under-sodium steam leak**			
22471.pdf	EVIDENCE OF SEVERE OVERHEATING DURING THE PFR SUPERHEATER 2 EVENT	PFR/SW/P(90)21	LINEKAR GAB;FRASER AS;YATES G	1990
27172.pdf	SUMMARY OF TEM INFORMATION ON TUBES IN TUBEBUNDLE WU3 FROM THE PFR CIRCUIT 2 SUPERHEATER TUBE 1	SH2/TEM8	GOWER SM	
27174.pdf	SUMMARY OF TEM INFORMATION ON TUBES IN TUBEBUNDLE WU3 FROM THE PFR CIRCUIT 2 SUPERHEATER TUBE 18	SH2/TEM4	GOWER SM	
27175.pdf	SUMMARY OF TEM INFORMATION ON TUBES IN TUBEBUNDLE WU3 FROM THE PFR CIRCUIT 2 SUPERHEATER TUBE 27	SH2/TEM1	GOWER SM	
27176.pdf	SUMMARY OF TEM INFORMATION ON TUBES IN TUBEBUNDLE WU3 FROM THE PFR CIRCUIT 2 SUPERHEATER TUBE 124	SH2/TEM2	GOWER SM	
27177.pdf	SUMMARY OF TEM INFORMATION ON TUBES IN TUBEBUNDLE WU3 FROM THE PFR CIRCUIT 2 SUPERHEATER TUBE 152	SH2/TEM9	GOWER SM	
27178.pdf	SUMMARY OF TEM INFORMATION ON TUBES IN TUBEBUNDLE WU3 FROM THE PFR CIRCUIT 2 SUPERHEATER TUBE 200	SH2/TEM5	GOWER SM	
27179.pdf	SUMMARY OF TEM INFORMATION ON TUBES IN TUBEBUNDLE WU3 FROM THE PFR CIRCUIT 2 SUPERHEATER TUBE 325	SH2/TEM6	GOWER SM	
27180.pdf	SUMMARY OF TEM INFORMATION ON TUBES IN TUBEBUNDLE WU3 FROM THE PFR CIRCUIT 2 SUPERHEATER TUBE 363	SH2/TEM7	GOWER SM	
27181.pdf	SUMMARY OF SEM INFORMATION ON TUBES IN TUBEBUNDLE WU3 FROM THE PFR CIRCUIT 2 SUPERHEATER TUBE 17	SH2/SEM2	MURRAY AL	
27182.pdf	SUMMARY OF SEM INFORMATION ON TUBES IN TUBEBUNDLE WU3 FROM THE PFR CIRCUIT 2 SUPERHEATER TUBE 27	SH2/SEM3	MURRAY AL	
27183.pdf	SUMMARY OF SEM INFORMATION ON TUBES IN TUBEBUNDLE WU3 FROM THE PFR CIRCUIT 2 SUPERHEATER TUBE 124	SH2/SEM4	MURRAY AL	
27184.pdf	SUMMARY OF SEM INFORMATION ON TUBES IN TUBEBUNDLE WU3 FROM THE PFR CIRCUIT 2 SUPERHEATER TUBE 325	SH2/SEM5	MURRAY AL	
27185.pdf	SUMMARY OF SEM INFORMATION ON TUBES IN TUBEBUNDLE WU3 FROM THE PFR CIRCUIT 2 SUPERHEATER TUBE 363	SH2/SEM6	MURRAY AL	
23	**Escalation of an under-sodium leak in a steam generator**			

Reference No.	Title	Location	Authors	Year
22478.pdf	RE-EVALUATION OF THE PFR SUPERHEATER 2 EVENT AND ITS IMPLICATIONS IN THE LIGHT OF RESULTS FROM THE FOLLOW-UP DEVELOPMENT PROGRAMME	PFR/SWP/P(90)30	TOMKINS B	1990
THE FOLLOWING REPORTS ARE NOT IN THE WOOD ARCHIVE				
	HEAT TRANSFER FROM A SODIUM-WATER REACTION JET	PFR/SGSG(89)26	JUDD AM	1989
	OVEHEATING OF PFR SUPERHEATER TUBES	PFR/SGSG(89)54	JUDD AM	1989
	OVEHEATING OF PFR EVAPORATOR TUBES	PFR/SGSG(89)63	JUDD AM	1989
24	**An Account of the Superheater 2 Under-Sodium Leak Event and its Consequences**			
27124.pdf	PFR SUPERHEATER 2 LEAK	AGT7(87);AGT8(87)P1	GREGORY CV	1987

Table 13: Wood Report List: Secondary Cold Trap Loops

Reference No.	Title	Location	Authors	Year
00893.pdf	INVESTIGATION OF SCTL BASKET NUMBER 6	OC/P(82)10	HUMPHRIES J; MORRISON N	
00939.pdf	SCTL BASKET LIFE	OC/P(83)69	HUMPHRIES J; MORRISON N	1983
00973.pdf	SCTL COLD TRAP VESSEL EXCHANGE	OC/P(83)99	MARSHALL W; GRAY J	1983
02555.pdf	SIMULATION OF FLOW IN THE PFR COLD TRAP	FR/THSG/P(85)68	HICKMOTT S	1985
02556.pdf	COMMENTS ON LARGE COLD TRAP DEVELOPMENT AND OPERATION	FR/THSG/P(85)69;FRDCC/SCWG/P(84)9	HUMPHRIES J	1984
02695.pdf	CURRENT STATUS OF THEORATICAL MODELLING OF COLD TRAPS USING THE VICSEN CODE	FR/THSG/P(85)66	HULME G	

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