A Guide to Fast Reactor Core Physics

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1 Introduction

This report supports NNL's Fast Reactor Knowledge Capture contract with BEIS by providing a brief overview of the Core Physics high priority technical area. It is intended to provide an entry point and sufficient overview to guide the reader in any potential areas of interest to facilitate further studies and does not provide a full narrative of the technical area. The discussions and references are based on the data available in Wood's Fast Reactor archives. It is noted that the term 'Reactor Physics' is used interchangeably with 'Core Physics' in the literature and for the purposes of this report, the two terms are considered to be equivalent.

The report is organised into the following sections:

- Section 2 provides a high-level history of the UK Fast Reactor Programme which helps to put the rest of the report into context.
- Section 3 provides a brief overview of the experimental facilities of importance to the Reactor Physics area and an introduction to where this data is now located.
- Section 4 provides a high-level discussion of the Reactor Physics area relating to PFR and identifies some of the relevant references held by Wood in this context. The Wood archives contain very little relating to DFR and thus the majority of the discussion focusses around PFR.
- Section 5 provides a discussion of calculational methods and nuclear data.

2 UK Fast Reactor Programme Summary

Many reactors and facilities were designed, built and operated in the UK during the heyday of the UK Fast Reactor Programme. These included small research reactors at Harwell and Winfrith, large test facilities at Risley and finally the larger and prototype fast reactors, DFR and PFR at Dounreay. This section provides a brief overview of the UK facilities and studies which are of relevance to the Reactor Physics technical area to help put the remaining sections into context.

ZEPHYR was a zero energy reactor based at Harwell in Oxfordshire. It began operation in 1954. It was fuelled with Plutonium and had a natural U breeder. Key outcomes from ZEPHYR were the demonstration that a fast reactor would breed along with providing neutron life-time measurements. Further discussion of the ZEPHYR reactor is provided in Section 3.

ZEUS was a second zero energy reactor also based at Harwell and began operation in 1955. ZEUS was U235 fuelled and provided specific information for the Dounreay experimental Fast Reactor (DFR). Neutron life-time measurements were also carried out.

The Dounreay Materials Test Reactor (DMTR) achieved criticality in May 1958. This reactor was used to test the performance of materials under intense neutron irradiation, particularly those intended for fuel cladding and other structural uses in a fast neutron reactor core. Test pieces were encased in a uranium-bearing alloy to increase the already high neutron flux and were then chemically stripped of this coating after irradiation. DMTR was shutdown in 1969 when materials testing work was consolidated at Harwell Laboratory. DMTR has not been considered further in this knowledge capture exercise.

DFR (Dounreay Fast Reactor) began operation in 1959 and operated successfully for 18 years. DFR had an output of 72 MWt, (15 MWe), used U-Mo metal fuel and was cooled with NaK. DFR was intended to provide the next step towards the design of commercial fast reactors with a step up in power and size from the zero power reactors. The primary objective of the DFR was a demonstration of the feasibility of fast reactor core operation. The heat transfer system was not designed as a suitable prototype of an economical system but instead as the most reliable combination of available components that would accomplish the purpose. Ref [3] provides a comprehensive summary of DFR operation and issues/problems/events which occurred and discusses DFR experiments in some detail.

ZEBRA was a flexible zero energy reactor built at Winfrith and was in operation from 1962 until 1982. ZEBRA was used to study the neutron physics of fast reactors and enabled mock-up assemblies with core dimensions and fuel enrichments similar to those of commercial reactors to be studied. Further discussion of the ZEBRA reactor is provided in Section 3.

Soon after completion of the DFR experimental programme, a consortia consisting of Atomic Power Projects, the Nuclear Power Group, and United Power company Limited, began studying the possibilities of a Commercial Fast Reactor (CFR). In June 1963, these parties produced a report for the UKAEA's reactor policy committee which concluded that the information and experience gained from the DFR provided the necessary confidence that a commercial sized fast reactor could be successfully built and operated. However, because of the large increase in size between the DFR and a commercial plant was necessary, the need for an intermediate plant incorporating the major steps in concept and scale was identified [4]. It is worth noting that NPG later became part of NNC (now Wood). There are numerous documents in the Wood archive discussing many aspects of the CFR design studies.

In 1966, approval was given for building a Prototype Fast Reactor (PFR) on land adjacent to DFR, to incorporate lessons learned from the operation of the DFR. Prototype Fast Reactor (PFR) was operated at Dounreay from 1974-1994. DFR had used a NaK alloy as coolant; PFR used sodium, which was cheaper and easier to handle. PFR operated at 600 MWt (250MWe) and utilised MOX fuel and breeder elements appropriate to commercial fast reactors. PFR fuel was a mixed plutonium–uranium oxide in sealed stainless steel-clad pins to achieve higher burnup than DFR and to keep the coolant relatively clean.

A new large breeder design for Commercial Demonstration Fact Reactor (CDFR) was studied by NNC and UKAEA in the late 1970's and early 1980's culminating in a proposed design in 1981. The proposed CDFR design comprised a plant of around 1300MWe and was designed as a very compact plant to save on materials costs. It is beyond the scope of the current exercise to delve into extensive detail about the political nuclear landscape at that time. However, suffice to say that the upsurge in interest in fast reactor technology in Europe and beyond, resulted in the formation of the European Fast Reactor Utilities Group (EFRUG) and ultimately lead to the formal European Fast Reactor Project of which the UK was an active participant. For the interested reader, a full history of the evolution of the UK, French and German Fast Reactor programmes into the EFR project can be found in [5].

The European Fast Reactor Project (EFR) was launched in March 1988 and initial, large scale design work ended in 1993 after the concept validation phase. The project was undertaken by EFR Associates, which was a consortium of vendors consisting of NNC, Interatom (later Siemens, then AREVA GmbH) and Novatome (later Areva NP) and sub-contract support from Arnaldo and Belgonucleaire. The goals of the EFR project at the outset were the demonstration of the following design aspects;

- for the design to be achievable,
- to provide economic performance i.e. with the generating cost of a commercial series of EFR to be competitive with contemporary pressurised water reactors,
- licensable in all participating countries
- achieve a safety level meeting the ambitious targets of (then) future nuclear plants.

The end of the EFR project came about gradually as the funding from the EFRUG members was simply not renewed when the agreements expired. This allowed the EFR project to be phased out and documented in a structured manner [5]. NNC (now Wood) have access to much of this documentation. As the EFR records date from a later period than those of the early UK Fast Reactor programme, and phase out of the EFR programme was in a structured manner, no further discussion of EFR is included here. UK government support for EFR ended in 1993. The EFR programme was continued at a much reduced scale by Framatome (now AREVA), and NNC continued to participate with funding support from BNFL until 1998, when the project ended.

The EFR programme was completed and closed in an organised manner by EFR Associates, such that the detailed design work, design options, safety case, and outstanding research and development options were comprehensively documented. A set of 47 synthesis documents were produced which gave the final statement of the status of the design studies, together with remaining open issues. The syntheses also presented summaries of abandoned options and innovative concepts. A head document [33] acts as an introduction and index to the suite of design documentation and synthesis reports.

It is worth noting that following the closure of all the UK Fact Reactor Facilities and the end of the EFR project, Fast Reactor studies continued in the UK via participation in international programmes, some of which are listed below for completeness. As these are international programmes and the reactors studied were never built these are not considered further here in the context of reactor physics.

Following the end of the EFR programme, three organisations in the UK (BNFL- now NNL and NNC Limited and AEA-T – now both part of Wood) actively participated in the CEA led international CAPRA/CADRA programme. Whilst CAPRA/CADRA cannot be badged as a UK programme, the UK did make a valuable contribution to this international programme. This is discussed elsewhere [6] and is thus not discussed in any further detail here.

In addition to the experimental and commercial demonstration concept reactors described above, the key thermal hydraulics rigs which operated within the UK to support the fast reactor programmes are described below:

- HIPPO (located at Risley) was a water test rig on a 1:5 scale, which allowed the flow patterns at the core exit plane and between the wrappers to be simulated. Flow velocities in the interstitial gaps could be measured by means of sophisticated instrumentation, such as laser Doppler anemometry.
- THOR (located at Risley) was commissioned in October 1990. It represented an 0.3 scale water model used primarily to observe the hot pool flow and quantify free surface conditions and potential gas entrainment.
- The CHARDIS 111 rig (located at Risley) was commissioned as late as in December 1991. It consisted of 61 subassemblies for studies of distortions and loads occurring during charging and discharging operations. The dummy fuel elements were provided by the French CEA, modified by Interatom in Bensberg, and finally used in CHARDIS 111 [79].
- SUPERNOAH (located at Dounreay) was used to study medium sized sodium-water leaks in steam generators. It did not become operational until 1993 [80].

UK involvement in Fast Reactor studies continued at a low level throughout the intervening years, for example, via European Commission funded Framework programmes which can be accessed via the CORDIS website and for a short period via engagement with the Generation IV International Forum.

3 UK Fast Reactor Experimental Facilities (ZEPHYR, ZEUS and ZEBRA)

This section discusses the UK experimental facilities and notes the contributions that they made to the UK Fast Reactor Programme in the context of Reactor Physics.

The ZEPHYR (Zero power Experimental PHYsics Reactor) began operation in 1954 at Harwell in Oxfordshire. ZEPHYR was a small low-power fast reactor with a core composed of plutonium and natural uranium sections surrounded by a uranium envelope. The reactor was controlled by moving sections of uranium near the core to vary the fraction of neutrons that may escape. Power level measurements were carried out by pulse-operated fission chambers with both linear and logarithmic ratemeters as indicators. No cooling was required since the maximum operating power was around 2 W. The reactor became critical for the first time on the 5th February 1954. Studies carried out included the investigation of the delayed neutrons and their influence on the reactor kinetics. Of the many aspects investigated, particular attention was given to a determination of the breeding characteristics of the system and the measurement of the nuclear parameters required for a theoretical interpretation of fast reactors. ZEPHYR was considered a success as it managed to demonstrate a breeding factor of two (i.e. two Pu nuclei were created for each one destroyed). Despite its success, extrapolating from ZEPHYR to a full size 100 MW reactor was considered a step too far and a second reactor, ZEUS, was designed [7,8].

ZEUS began operation in 1955. ZEUS was fuelled with U235 and was essentially a zero-power mock-up of the later DFR reactor. It was designed to simulate as closely as practicable the size and composition of the then proposed DFR. ZEUS provided specific information for the Dounreay experimental Fast Reactor (DFR) including neutron life-time measurements.

ZEBRA was a flexible zero energy reactor built at Winfrith. It operated from 1962 until 1982 at the Winfrith site of the United Kingdom Atomic Energy Authority (the Atomic Energy Establishment, Winfrith, AEEW). ZEBRA was used to study the physics of fast reactors in mock-ups with material volume fractions and a neutron spectrum similar to those of commercial reactors. The core of ZEBRA was made of many small plates which could be taken apart and re-assembled to test out new designs. It could accommodate either highly enriched uranium or plutonium in its core. Mock-up type assemblies representing the fast reactors PFR and MONJU and a European fast reactor (which was being studied in the 1970s) were modelled, as well as simple assemblies for validation of nuclear data and methods of calculation. Some experiments were designed to validate the methods used to treat the heterogeneity of the fuel assemblies and to calculate control rod effectiveness. ZEBRA was used to provide information to help fill the wide gaps in the theoretical methods and data needed for calculating reactor performance parameters especially for reactor core compositions with the softer neutron energy spectra associated with large power fast reactors.

It is worth noting that in the design and commissioning phase of PFR, an exchange agreement with USA provided access to information from USA research reactors such as ZPRIII and, from 1963, ZPRVI. In particular, the latter was used to study the reactor physics of systems having the relatively soft neutron energy spectra typical of a large fast reactor. The ZEBRA and the ZPR reactor programmes were complementary. The data obtained were used together with more basic nuclear physics data derived from nuclear physics experiments such as particle accelerators to guide the development of a basic fast neutron data base as well as fast reactor calculational methods.

The data from the UK experimental facilities was given to the NEA and entered into the NEA databank and the knowledge from them has thus been captured in a usable format and is available to those who subscribe to the NEA Databank.

4 PFR operation in the context of Core Physics

This summary serves to identify the data which still exists within the Wood archives relating to the core physics area and puts a high-level context around this. The majority of reports in the Wood archive in the context of Reactor Physics relate to the PFR, the design of the planned CDFR and the PFR Plant Performance working group (PPWG); an almost complete set of these papers exists.

A report published in July 1996 provides an outline review of PFR development and operation [1] and includes a somewhat anecdotal discussion of the Reactor Physics area, but with very few references to support or help the reader to go back to the original information. The following paragraphs draw heavily on this reference to summarise some of the activities which were carried out during the operating life of PFR. In the early days of PFR operation, physics data available to support the project was limited mainly to those which had been derived from low power experimental facilities such as ZEPHYR and ZEUS (see Section 3), plus that which emerged from the DFR project. The written record of the DFR project within the Wood archive is sparse and thus is not discussed in any greater detail in this account. A good summary exists in the open literature [3].

When PFR first went critical in 1974, the shape of the softer neutron energy spectrum induced by the use of mixed-oxide fuel (rather than the metal fuel used in DFR) and lower enrichment was not well established. In addition to this, as the cross-section data was imprecisely known, uncertainties in reactivity and power distribution calculations, but more especially in the reactivity worth of control rods and the components of the temperature coefficients, were introduced. The full significance of the Doppler and sodium voiding effects was not then appreciated. This is discussed further in Sections 4.1 and 4.3.

The PFR reactor was first made critical uneventfully. Ref [1] describes the initial loading strategy, but provides no references which may have included the original data. Subassemblies containing solid steel pins were replaced, in a pre-planned sequence, by subassemblies containing fuel in the form of mixed uranium/plutonium oxide pellets clad in sealed stainless-steel pins. The objective of the first loading was to check theoretical reactivity prediction; 54 subassemblies were loaded and the reactor found to be within 0.6% of critical with the absorbers at maximum height and therefore with minimum effect. 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 the use of nominal data; for example, the measured positioning of the subassembly clusters in the diagrid, and the actual isotopic composition and measured density of the individual fuel subassemblies differed marginally but significantly from the nominal figures.

Smith and Broomfield [1] mention a number of reactor physics experiments which were conducted during the start-up period and early operation of PFR. These included a series of low power reactor physics experiments which were conducted using fission and ionisation chambers and activation foils [9,10]. The results confirmed the pre-calculated neutron flux and power density distributions, the reactivity invested in the absorber rod system and the principal temperature coefficients of reactivity. Subsequent operation at power with a variety of coolant flow rates endorsed the methods used to predict the power coefficients of reactivity.

The Doppler broadening of reaction cross-sections as temperature is increased, has an important stabilising effect in normal operation and also under theoretically postulated abnormal power and/or flow conditions. The effect had been measured in a series of experiments up to 30% power: the results compared well with calculated values, confirming the physics data of the reactions and the heat transfer processes involved.

Early operation showed the reactor to be very stable and easily manoeuvrable. It was designed at a time when experience of fast reactors was limited to smaller power units using metal fuel; unexpected reactivity effects arising from subassembly bowing in thermal gradients had been uncovered in these reactors and, as a consequence, this possibility was given close attention by the PFR designers. The subassemblies were grouped in clusters of six round a hexagonal leaning post and cantilevered to lean against low friction pads on the faces of the leaning post in an attempt to counteract the effects of bowing which could, in other circumstances, have introduced positive feedback. The effect of thermal or irradiation creep was not expected to nullify this concept and subsequent experience on PFR demonstrated this.

As an insurance, in case abnormal feedback had been found, a reactivity oscillator was provided to assist diagnosis but it was not needed. Normal operational manoeuvres provided some early information about the feedback; the benign behaviour of the reactor was reassuring. Furthermore, the background reactivity noise level was, at 0. 1 %, at the bottom end of the anticipated range. This permitted the use of small reactivity injections to excite a measurable dynamic response which was analysed to provide the same information that the oscillator would have provided. Ramp rates of 0.6 cents per second have been generated by control rod movement and negative steps of up to 15 cents have been injected by the deliberate dropping of a pre-positioned control rod.

Analysis of the results of these dynamic tests was compared with theoretical models of reactivity feedback. The principal component was associated with the Doppler effect, but with a significant contribution from the expansion of the fuel; the latter was 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 (e.g. from power or flow changes) the expansion of the control rod insertion mechanism was important. For changes in coolant inlet temperature (e.g. during the early stages of the approach to power or in some hypothetical accident situations) the radial expansion of the core support system imposed a large reactivity change. Each of these feedback components was negative and had a relaxation type time constant and was dynamically stabilising. These effects are discussed in more detail in the later sub-sections of this document.

The plant responses to transients, deliberately imposed or adventitious, were compared with those generated by computer models and the agreement was sufficient to give confidence that no significant feedback factors had been overlooked.

There were more than 30 refuelling campaigns. The reloading was pre-planned to provide additional reactivity, sufficient to compensate for burn-up during the next operational period and to accommodate any changes in the experimental fuel loading. The reactivity target lay within a small window. Too much reactivity would have required control rod insertion to an extent which would have left an unacceptably small reactivity input per rod (shutdown margin): too little would have compromised the length of the next operating run. A further and much tighter criterion which was also embodied in the operating directives, required prediction of the post re-load reactivity to within about 40 cents of the observed value. This criterion was equivalent to about 13mm in the control rod curtain height and was designed to ensure that a significant mis-loading could not have passed unnoticed.

Originally, a critical approach was made after a reload, with the charge machine still in position. Improved reactor physics models were developed which allowed the sub-critical margin to be monitored as the reload proceeded [9,10]. The models allowed for the perturbation of reactivity, neutron source magnitude and distribution, and flux distortion introduced at each fuel element interchange. Typically, a set of fifteen 2-dimensional calculations, representing fifteen interchanges, sufficed to provide the operators with adequate prediction of count rates on each of the low power counting channels· during the complete reload operation.

The calculated sub-critical reactivity margin varied during the reload campaign as did the accuracy of its assessment which 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 operations the shutdown margin was controlled administratively to ensure that the reactor would remain sub-critical in the event of the worst two mis-loadings. 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 [11-26]. 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 [28,29,30]. 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 [31,32].

4.1 Doppler Effect

A change in reactor fuel temperature may produce a reactivity change. This, known as the Doppler effect, arises through the change in the mean thermal agitation velocity of atoms. The shape of a reaction resonance is broadened as temperature is increased and this increases the effective reaction cross-section. The reactivity change may be positive or negative; usually a fission resonance will lead to a positive Doppler coefficient and a capture resonance will provide a negative coefficient. This Doppler broadening of reaction cross-sections as temperature is increased is an important phenomenon for Fast Reactor operation.

The importance of the Doppler effect in the overall temperature coefficient, was first coming to light in the design phase of PFR. In the early days of PFR operation this was still not well understood nor was its significance fully appreciated. In fact, there was concern, on theoretical grounds, that the

Doppler effect from plutonium might generate a fast acting, destabilising, temperature coefficient. Smith and Broomfield [1] mention a series of experiments in PFR up to 30% power which were used to measure the Doppler effect: the results compared well with calculated values, confirming the physics data of the reactions and the heat transfer processes involved. Data on these experiments is not present in the Wood archive.

A direct measurement of the Doppler effect was planned for ZEBRA. A hot and a cold sample were to have been oscillated into the ZEBRA core but this was subsequently modified when it became apparent that adequate accuracy would be obtained only if a heated zone, rather than a small heated sample, were installed. With the heated sample experiment the neutron energy spectrum driving the sample experiment would have been almost unaltered by the hot sample and sensitivity would have been lost.

A number of the papers are identified in the Wood Document Index relating to the analysis and calculations to further understand the Doppler effect and the impact of various core materials on this and of the contemporary nuclear datasets on the prediction accuracy. These are highlighted in the Document Index [34-54, 78].

4.2 Temperature and Reactivity Coefficients

An important safety feature of PFR, which was revealed by operational measurements, was the self-stabilising nature of the power level. The strongly stabilising negative reactivity/temperature coefficient ensures that under all conditions, including accident situations, the reactor power is self-limiting. To calculate this phenomenon in detail requires a knowledge of the reactivity coefficients and the time and spatially dependent temperature distribution within the reactor core, the subassembly support diagrid, and the hot sodium pool. Determining the detailed spatially and time dependent temperature distribution requires an understanding of the complex hydrodynamics of the sodium in the various regions of the core and the hot pool. Considerable progress was made in developing and verifying methods for dealing with these thermal hydraulics questions. The topic of thermal hydraulics is explored elsewhere in the current project. Putting all this together made a convincing case for arguing that the reactor was self-protecting against a variety of accidents including extremely improbable ones such as complete failure of sodium pumping together with failure to trip the reactor. A number of reports identified in the Wood Document Index discuss temperature and reactivity coefficients and provide information on the measurements of these. [16-27, 55-60].

4.3 Sodium Void

The sodium void coefficient is an important phenomenon for Fast Reactor control and this is an effect that was also not fully appreciated in the design phase of PFR. The sodium void coefficient is considered important as it results in a reactivity insertion as the core voids (i.e. it produces a positive feedback). Smith and Broomfield [1] mention that sodium void measurements could not be made in ZEBRA in time for the design of PFR but data was made available from US ZPR VI experimental reactor, following the signing of the US/UK agreement which covered this aspect. They cite a sodium void experiment in PFR that was designed but not implemented. Subsequent improvements in calculational methods came largely from experiments in the ZEBRA reactor during the early operational phase of

PFR thus rendering the proposed experiment unnecessary. A number of papers are included in the Wood Document Index which provide an entry point for further examination of the sodium void effect calculational methods and related experiments [34,52,53,61-67].

4.4 Control Rods

Control rod design and materials are important in terms of reactivity control during reactor operation and in fault conditions. A brief discussion of the design of the PFR control rods and selected control rod materials is thus included here.

The reactivity control rod material selected for PFR was initially tantalum in the form of discs protected from the sodium by a molybdenum coating. The shut-off rods contained enriched boron carbide, the un-needed level of enrichment reflecting the uncertainty in the calculation methods then available for absorber effectiveness, especially in the core/breeder boundary region where the shut-off rods are located. The long-term acceptability of the tantalum rods was put in some doubt by concern over the integrity of the molybdenum coatings and, more importantly, by the increased emphasis on minimising highly active waste material such as would be generated by tantalum irradiation. Ultimately an alternative design based on enriched boron carbide was adopted. They had enhanced pellet-to-clad and wrapper-to-guide-tube clearances to accommodate wrapper swelling and bowing; the designs differ in the length and diameter of the pins and the clearance sizes. Smith and Broomfield [1] cite an extended life version of the design successfully increased the absorber life from 371 to 530 equivalent full power days.

For most of the PFR operational life, control of reactor power was achieved by vertically positioning the five boron carbide control rods (worth collectively around 13$). There are also five boron carbide shut-off rods (worth around 13$) which provided additional shutdown margin during refuelling. 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, whether this data still exists or not is not currently confirmed. It is worth noting the subsequent international Fast Reactor designs all use enriched boron carbide rods so in this respect the experience from PFR made a significant contribution.

PFR operating experience is also cited by Smith and Broomfield [1] as having endorsed the reliability of individual absorber rods. However, one aspect of the design, the relatively limited diversity in the shutdown system, proved difficult to resolve at the time in spite of repeated attempts to produce a viable alternative to supplement the existing system. Hence a key parameter in the absorber rod reliability assessment was the possibility of a common mode failure. A Working Group of Experts examined the limited diversity in the shutdown rods and concluded that the claimed figure of 6 to 10 rod failures per demand was justifiable, given the rod age distribution and the rod exercising programme. A small number of reports in the Wood Document Index examine the merits of Boron Carbide as an absorber [68-72].

Physics is also cited by Smith and Broomfield [1] as making important contributions in many of the multi-disciplinary areas during PFR development, especially in the fields of mechanics, thermal hydraulics and instrumentation. Fast reactor physics was a more specialist field requiring the development of many new methods and much new nuclear data.

5 Calculational Methods

No summary of reactor physics would be complete without a discussion of the calculational methods and nuclear data. A Fast Reactor Super Archive report on neutron physics [2] aims to provide an entry point to all fast reactor neutron physics and shielding records considered of significance in 1989. The majority of these relate to nuclear data and calculational methods. The following sections provide an overview of this information and a discussion of this in the context of the current state of the art.

5.1 Neutronics Codes and Methods

Neutronics codes focus their applicability on phenomena that are present in the reactor core by solving the neutron transport equation. This equation is a balance equation for the production and destruction of neutrons, through the absorption of neutrons in materials or escape of neutrons from the analysed domain. They are capable of predicting reactions between the atoms of materials in the reactor core and neutrons or fission products, which result from the fission process. Through these codes, one is able to evaluate the most important parameters, from the reactor physics point of view, such as the reactivity of the core, the multiplication factor, neutron flux, isotopic changes in the core fuel or the fuel burnup. Calculations for fast reactors are necessarily complex due to the 3D complexity of the reactor core and complexity of the calculation process.

Neutronic codes can be divided into a number of types, depending on the problem-solving method. These comprise deterministic methods or the Monte Carlo methods based on the probability density functions. Neutronic codes are dedicated to different types of applications, such as

1. lattice assembly calculations for the creation of homogenous multigroup constants for deterministic reactor power simulation,
2. whole core calculations to investigate the fuel cycle,
3. codes validation, which solve the neutron transport equation for the fuel assemblies,
4. neutronic calculations and fuel burnup for the research reactors on the whole core level.

In the early days of the UK Fast Reactor programme it was necessary to develop the capability to carry out the types of modelling described above to predict the behaviour of the core to support continued operation and to help design the then postulated future commercial sized plants. The UK methods were widely reviewed in 1973 in [73]. Methods development continued in the UK beyond this time as described below.

In the early days of PFR, the UK COSMOS modular code system was developed to support operational predictions, and the development of reactor physics methods took place within the COSMOS suite of modules. COSMOS was used as the main tool for reactor physics analysis for many years during the operation of PFR, beyond into studies for EFR and in the early days of the CAPRA/CADRA programme. In parallel to the development of COSMOS a French code scheme, ERANOS was developed by CEA. In the mid-1990's ERANOS was adopted in the UK for use on the CAPRA/CADRA project and development of COSMOS ceased. Ongoing development of ERANOS continued over the intervening years, by CEA, to the extent that COSMOS is now obsolete. Development of ERANOS ceased when the French fast reactor fleet (the Phenix and Superphenix reactors) were shutdown so ERANOS is now also largely obsolete. Fast Reactor functionality has been developed within Wood's ANSWERS code WIMS with the incorporation of the ECCO cross-section processing capability.

With modern advances in computing power and understanding of reactor physics from world-wide operational facilities, calculational methods and their accuracy have moved on significantly since the Fast Reactor Super Archive report was produced in 1989. A number of the reference documents reviewing the calculation methods are still available in the Wood archive and these have been identified as part of the database deliverable [74-77].

A key aspect which affects the accuracy of any neutronics calculation is the basic nuclear data. This is discussed in the following sections.

5.2 Basic Cross Section Data

Basic cross-section data is of great significance to the accuracy of any reactor physics predictions and is generally the largest source of uncertainty in any reactor physics prediction. The UK fast reactor programme made significant contributions to this, contributing to the contemporary international efforts on developing cross-section data.

The measurement and evaluation of microscopic cross-section and other nuclide-specific data has long been an international activity and the most comprehensive data collections are held by the international agencies. The data file used by the European Fast Reactor partners was called 'JEF'. In the days of PFR this was the "Joint Evaluated File" but in subsequent years this has evolved in the JEFF "Joint Evaluated Fission and Fusion" Nuclear Data Library suite. Ongoing development of JEFF continues to the present day and the current public release is JEFF 3.2, although a test version 3.3 is under development. JEFF is maintained by the NEA Data Bank and currently is a collaborative project between NEA Data Bank participating countries, including the UK. The JEFF library combines the efforts of the different JEFF Working Groups to produce common sets of evaluated nuclear data, for fission and fusion applications. The JEFF suite of nuclear data libraries contains a number of different data types, including neutron and proton interaction data, radioactive decay data, fission yields, and thermal scattering law data. As discussed in a previous section any data which was derived from the UK fast reactor programme is available via the NEA databank and the nuclear data and lessons learned from the UK FR programme in the context of reactor physics is included in this. Whist this data is of significance and is of historical interest it is already well preserved and available to NEA Databank subscribers and thus no further discussion is included here.

5.3 Cross-section Processing

The basic data described in the previous sections is provided at thousands of energy points. Cross-section processing techniques have continued to be developed in the years since the end of the UK fast reactor programme with development of the fine group ECCO cross-section library which is now included and available in the Wood ANSWERS WIMS software. Again, whilst the UK's contribution to this from the UK fast reactor programme is of historical interest it is not worth dwelling on in the context of the current Fast Reactor knowledge capture exercise.

6 Summary

This report has provided a summary of the Core Physics high priority technical area, with a focus on PFR and the UK test reactors which supported the early UK Fast Reactor programme. Whilst a significant amount of effort was expended in subsequent studies, including EFR and the subsequent CAPRA / CADRA programmes these are well documented. Therefore, with the limited effort available within the current knowledge capture project, priority has been given to the operating reactors.

A Abbreviations

Abbreviation	Complete Term
CDFR	Commercial Demonstration Fast Reactor
CFR	Commercial Fast Reactor
DFR	Doureay Fast Reactor
EFR	European Fast Reactor
EFRUG	European Fast Reactor Utility Group
FR	Fast Reactor
FRCG	Fast Reactor Consultants Group
FRDCC	Fast Reactor Design Coordination Committee
NEA	Nuclear Energy Agency
NNC	National Nuclear Corporation Limited
NNL	National Nuclear Laboratory
PFR	Prototype Fast Reactor
PPWG	Plant Performance Working Group
UKAEA	United Kingdom Atomic Energy Authority

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