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A REVIEW OF THE UK FAST REACTOR PROGRAMME

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Abstract

The general position with regard to nuclear power and fast reactors in the UK during 1996 is described. The main UK Government-funded fast reactor research and development programme was concluded in 1993, to be replaced by a smaller programme which is mainly funded and managed by British Nuclear Fuels plc. The main focus of this programme sustains the UK participation in the European Fast Reactor (EFR) collaboration and the broader international links built-up over the previous decades. The status of fast reactor studies made in the UK in 1996 is outlined and, with respect to the Prototype Fast Reactor at Dounreay, a report of progress with the closure studies, fuel reprocessing and decommissioning activities is provided.

1. The UK Nuclear Industry

The latest available statistics on electricity supply relate to 1995. The general trend of recent years continues, ie. increases in generation from combined cycle gas turbine plant and from renewable energy sources whilst the coal and oil contribution is reduced. Nuclear electricity remains an important contributor in Britain, but it is not expected to grow in the near future. Electricity supplied in 1995 comprised:

Coal	47%	
Nuclear	28%	
Gas	16%	
Oil	5%	
Others	5%	(renewables, imports and hydro)

The nuclear component comprised

Total NPP	35
Magnox Units	20
AGRs	14
PWRs	1

Total Net Nuclear Capacity 14168 MW(e)

A total of ten nuclear power reactors have been taken out of service including the experimental and prototype fast reactors (DFR and PFR) at Dounreay and are in various early stages of decommissioning.

The UK nuclear industry has been restructured as a consequence of the Government's Review which was published in May 1995. The main features of the new structure are:

A new company 'British Energy plc' has been formed but which retains the previous nuclear utilities 'Nuclear Electric' and 'Scottish Nuclear' for licensing reasons. British Energy plc, which was successfully floated on the UK stock exchange in July 1996 owns the fourteen UK Advanced Gas Cooled Reactors and the Sizewell B Pressurised Water Reactor.

The twenty Magnox stations are retained in the newly formed, nationalised 'Magnox Electric' (ME). The intention is that ME will become part of British Nuclear Fuels plc (BNFL) and negotiations are in progress to facilitate this transition.

The former United Kingdom Atomic Energy Authority has been divided into two parts: 'AEA Technology plc' which is a privatised company offering technical services on a commercial basis to other organisations, and the 'UKAEA Government Division' which remains Government owned and which is responsible for discharging the decommissioning of nuclear facilities and for radwaste management arising out of the past activities of the UK Atomic Energy Authority. The fast reactor system 'Intellectual Property' previously owned by UKAEA was vested with AEA-Technology on 31.03.97. AEA Technology plc was subsequently successfully floated on the UK stock exchange in September 1996.

BNFL remains 100% Government owned. The company invests a significant amount annually into fuel cycle research and development and under the restructured nuclear industry has sole exploitation rights to FBR fuel cycle technology.

Other key events in the UK during 1996 included:

- The official opening of Sizewell B PWR
- Life extension to 50 years for BNFL and 37 years for ME Magnox reactors

2. UK Activities on Fast Reactors: Progress in 1996

The UK has continued to participate in the European collaboration. Work has been carried out by the appropriate organisations, AEA-T, NNC and BNFL, with funding mainly from BNFL. As in previous years the main activities have been centred on the continued development of the EFR design, and on the CAPRA programme led by the French CEA, which are reported in detail by our French colleagues. The notes below refer to the main UK contributions.

EFR

Further potential improvements to the design of EFR have been examined. As members of EFR-Associates NNC have investigated the advantages of Gas Expansion Modules ("GEMs") as passive safety devices providing negative feedback. They have concluded that GEMs would be useful in unprotected loss-of-flow accidents, but that other devices, for example providing enhanced expansion of the control rod suspension, are effective in a wider range of accident transients. NNC have also participated in work on the effectiveness of a rectangular containment building for EFR.

NNC have also contributed to an investigation of radical alternatives to the EFR design by examining the potentiality of a gas-cooled fast reactor, based on past UK studies and on the extensive UK experience of AGR thermal reactors. There are advantages over liquid metal coolants in terms of in-service inspection and repair, but there may be difficulty in meeting safety requirements.
BNFL is an active member of the European Fast Reactor Utilities Group (EFRUG) and in 1996 hosted the annual exchange meeting between EFRUG and the Japan FEPC. BNFL is currently participating in an EFRUG study of the relative safety of fast and thermal reactors, in which the specification of EFR and the users' requirements for a European PWR are being compared. The work is continuing, but preliminary conclusions are that identical safety criteria could be met by the two systems.

AEA-T and NNC have undertaken several activities to make available operating experience from PFR to the designers of EFR and as a contribution to information exchanges within international agreements involving the European collaboration.

CAPRA

The CAPRA reference core utilises mixed oxide fuel with a high plutonium concentration to maximise the Pu consumption rate. As a result the ratio of cladding damage to fuel burnup is lower than in conventional fast reactor cores, and it may be possible to make use of this by increasing the discharge burnup. NNC have optimised a high-burnup CAPRA core in which the fuel is irradiated until a cladding dose of 180 dpa is reached, and shown that there are potential advantages in terms of improved reactivity control and longer fuel residence time, and that safety criteria can be met. Irradiation testing would of course be needed to confirm that burnups of 25% are possible.

The performance of CAPRA cores has so far been calculated using nuclear data adjusted and validated for use with cores of the Super-Phenix and EFR type, in which the Pu concentration is relatively low. AEA-T have been investigating the use of unadjusted data from the JEF 2.2 set processed via the ECCO cell code. It is important to trace the origin of differences in the calculated reactivity and performance parameters, and to propose routes for validating the calculation procedures.

The highest Pu consumption rates can be achieved only if uranium is eliminated from the core. Nitride appears to be a possible non-uranium fuel material, and the performance of a core fuelled with pure PuN has been studied. AEA-T have studied the vaporisation behaviour of nitride fuels, surveyed the extant data on the physical and chemical properties of PuN and (U,Pu)N, and set up a calculational model of a nitride fuel pin. Preliminary results indicate that acceptable burnups can be achieved provided potential problems of fuel swelling can be solved.

Fuel Cycle Studies

In parallel with the work done in collaboration with the European partners BNFL has conducted studies of the potential role of fast reactors in the UK and elsewhere. It is important to consider the fuel cycle as a whole and to make use of fast reactors in the optimum way to maximise safety and economic advantage while minimising environmental impact and proliferation risks. To this end accelerator-based systems as alternatives to critical reactors, and the thorium cycle as an alternative to the uranium-plutonium cycle, have been examined with particular reference to the implications for fuel fabrication, reprocessing and waste disposal. This work continues but the initial conclusion is that the critical Pu-fuelled fast reactor, properly integrated with reactors of other types, and with optimised arrangements for Pu recycling, has many attractive advantages.

IAEA IWGFR Meetings

The UK continues to support the activities of the IWGFR when appropriate UK expertise is available. In 1996 this included attendance at:

- The Annual IWGFR meeting in Kazakhstan in May
- The TCM on 'Creep Fatigue Damage Rules to be used in Fast Reactor Design', hosted in the UK in June
- The TCM on 'Evaluation of Radioactive Materials Release and Sodium Fires in Fast Reactors'.

3. PFR Closure Experiments

The programme of tests associated with the closure of PFR made use of the opportunity to gain information that could not have been obtained from rig experiments or modelling. The programme was described in some detail in the paper presented at the 1995 Annual Meeting of IWGFR. The studies were in 3 parts:

i) Steam generator leak detection studies in PFR
ii) Sodium-water reaction tests in the Super Noah rig
iii) Destructive examination of materials from PFR

The studies of steam generator leak detection and tube overheating were completed on 31 March 1995 and were described in the paper presented at the 1995 Annual Meeting of IWGFR. A programme examining materials removed from the reactor was concluded at end March 1996. These examinations were restricted to areas outside the primary circuit, since the funding available precluded the study of active components such as the Above Core Structure or Intermediate Heat Exchangers. The secondary circuit studies were in three parts:

i) Examination of carbon steel components exposed to sodium environments.
ii) Study of delayed reheat cracking in austenitic steel weldments.
iii) Study of secondary pipework transition welds.

The programme and the preliminary results were outlined at the 1996 Annual Meeting of IWGFR.

Examination of as-welded joints in the PFR carbon steel sodium storage tanks that were exposed to sodium vapour were carried out to provide some assurance that similar steels, likely to be used for fast reactor roof structures, would not be susceptible to the type of cracking observed in the molybdenum-containing 15Mo3 steel. Carbon steel could then be used in the as-welded state for roof constructions, avoiding the need for costly stress relief heat treatment. The study included ultrasonic inspection of welds on the selected storage tank and removal of three weld samples for dye penetrant and metallurgical examination. No cracking was found. It was concluded that, within the relatively restricted range of temperature (≤150°C) and time (several months) of exposure examined, carbon steels in the as-welded condition can operate successfully in contact with sodium without cracking.

Three welds from a length of secondary sodium circuit (superheater) pipework that had operated at about 550°C and three samples containing known defects in the Superheater 3 vessel shell (all of these being in Type 321 steel welded with Type 347 consumable) were removed for metallurgical examination. This was primarily to study delayed reheat (relaxation) cracking. No cracking attributable to this mechanism was found in the pipework welds. It was concluded that this gave added confidence that this type of cracking is unlikely to occur in the less susceptible Type 316L(N) steel, in conditions of low restraint, during a reactor lifetime.

For the known defects in the Superheater 3 vessel shell, the crack size measured by ultrasonic inspection was, in each case, very similar to the actual crack size found by metallography, although the positioning of the crack was not in good agreement for one location. There was some evidence of crack initiation by the fracture of inclusions and embryonic cracking of the matrix from these. It was not clear if the fracture of inclusions had taken place during operation or prior to operation. Progressive growth of a small defect over 3 to 4 years of operation had been by intergranular cracking. It was not possible to distinguish whether this was by delayed reheat cracking or conventional creep crack growth, as this characteristics is common to both mechanisms. A large crack that had remained stable with little growth over some 4 years of operation was confirmed to be at the weld centreline and has characteristics consistent with a hot tear that occurred during fabrication of the vessel.

A secondary pipework transition weld between Type 321 steel pipework from the superheater and a $2^1/_4$Cr1Mo nozzle at the inlet of the evaporator on secondary sodium circuit 1 was examined metallurgically, including use of transmission electron microscopy. This weld had operated at 490°C. No evidence was found of any mechanism that was likely to have led to failure of this type of weld during longer term operation.

A notable finding of the overall materials study was that there was a significant content of tritium in the secondary circuit components. This was over three orders of magnitude above the 'free issue' level. Very marked partitioning of the tritium content away from the ferritic steel to the austenitic steel took place at the Evaporator 1 nozzle transition weld. The level on the austenitic steel side of the transition weld was three times higher than in the Superheater 3 vessel and Circuit 3 pipework austenitic steel and was at least 5000 times higher than in the adjacent $2^1/4$Cr1Mo steel.

4. PFR Decommissioning

The PFR core has been removed and replaced by a dummy core as part of the first stage decommissioning. Most of the sub-assemblies with failed fuel cladding have had the pins removed. Although problems were expected with this operation, the pins have so far been pulled cleanly. Decommissioning work is presently concentrating on the preparation of the sodium disposal plant to dispose of the sodium inventory of PFR, and the supporting transfer system, to be ready for operation by end 1998. Construction of the sodium disposal plant began in January 1997. All present major plant engineering work is in support of the sodium disposal phase. A flask is being prepared to transfer components out so that a pump to be put in place to effect the sodium transfer.

5. **PFR Fuel Reprocessing**

The reprocessing of fuel from PFR has continued. Successful reprocessing of all categories, fuel, axial and radial breeder and residues, has been demonstrated. During 1996, 24 radial breeder and 4 core assemblies were reprocessed. In total, approximately 127 kg Pu was separated from 3654 kg U.

A dissolver leak occurred in September 1996 and an assessment is being made on how this can be dealt with. It is, however, possible to by-pass the head end using the residues recovery plant in order to reprocess non-irradiated fuel.