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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 UK during 1995 is described. The status of fast reactor studies made in UK is outlined and a description and statement regarding the conclusions of the programme of studies associated with the closure of the Prototype Fast Reactor is included.

1. The UK Nuclear Industry

The latest available statistics on electricity supply relate to 1994 and show that the total electricity generated was 325 TWh, an increase of 0.7% on 1993. Of this 49% was from coal, 28% nuclear, 13% gas, 5% oil and 5% others. Nuclear generation was 88.3 TWh, down 1.2 % on 1993. This was due in part to the closure of PFR, and also to significant increases in generation from combined-cycle gas turbine plant and from renewable sources.

The Sizewell B Pressurised Water Reactor (PWR) station, the newest in the UK, had a very successful initial year in 1995. Criticality was achieved in January, the plant was first connected to the grid in February, and full power operation was reached in June. Since then it has operated reliably with a very high load factor.

In May 1995 the Government's review of the nuclear industry was published. The main points were as follows.

- There is believed to be a significant role for nuclear power in the future.
- The perceived environmental risk of nuclear power can be counteracted by emphasis on safety, a rigorous regulatory regime, and a good safety record.
- Long-term economic interests in the European Union are best served by a free market in energy.
- In the short-term nuclear power is not essential to meet environmental targets.
- In the current UK market the combined-cycle gas turbine system is the most competitive for generation.
- The "nuclear levy" on electricity sales is to be stopped.
- The Advanced Gas-Cooled Reactors (AGR) and PWR stations can be privatised in 1996.
- It is not practical to privatise the Magnox stations.
- The financing of long-term nuclear liabilities is to be separated from power generation.
- Fast reactors were not mentioned specifically.

The nuclear generation industry is being reorganised in preparation for privatisation. A new holding company, "British Energy plc", has been set up to take over the AGR and PWR stations of Nuclear Electric and Scottish Nuclear. It is intended to privatise British Energy when appropriate. The older Magnox stations are to be retained in national ownership in a new company "Magnox Electric plc". Eventually Magnox Electric may become part of BNFL (who already own the two original Magnox stations at Calder Hall and Chapelcross).

As was noted in the UK position paper in 1995, the United Kingdom Atomic Energy Authority has been divided into two organisations. These are AEA Technology plc and UKAEA Government Division. By Act of Parliament, AEA Technology plc. has been vested as a private limited company, wholly owned by the UK Government, with effect from 1 April 1996. It is the UK Government's intent to privatise this company, when appropriate, during 1996. It is the aim of AEA Technology to become a leading scientific and technical services company.

UKAEA, which is to remain in the public sector, is responsible for discharging the decommissioning of nuclear facilities (including the Prototype Fast Reactor (PFR)) and Radwaste management arising out of the past activities of the United Kingdom Atomic Energy Authority. The Intellectual Property arising from the processes of decommissioning PFR will be held by UKAEA.

2. PFR Decommissioning

Since the closure of PFR at end March 1994, UKAEA has been responsible for the decommissioning of the reactor. This is to be divided into three active stages separated by periods of care and maintenance. The first stage comprises initial decommissioning and preparation for care and maintenance and will last until the early 2000s. An initial sub-stage has proceeded to schedule and the removal of core components and their replacement by dummy components was completed in 1995. In addition, the secondary sodium circuits and the NaK-filled decay heat rejection circuits have been drained down, non-essential services have been isolated and redundant plant items have been removed.

During the second sub-stage, a sodium disposal plant is to be constructed and operated on site to dispose of the liquid metal inventory of the reactor. The second sub-stage of decommissioning will also include the conditioning of the primary circuit for long term storage, the draining and cleaning-out of the buffer store and the dismantling non-essential plant and buildings. On the present schedule, the plant will then be placed in long term care and maintenance, including surveillance of the integrity of the secondary containment building, until 2070.

The Stage 2 decommissioning, scheduled to begin in 2070, will comprise the dismantling of the secondary sodium and steam circuits, the construction of a new containment around the primary circuit and irradiated fuel cave and the dismantling of the original secondary containment building. This Stage 2 decommissioning is scheduled to take 6 years. The Stage 3 decommissioning, comprising dismantling of the reactor internals and the concrete shielding, will follow a further period of care and maintenance and will take 20 years.

3. PFR Closure Experiments

A programme of experiments associated with the closure of PFR was carried out. The aim, with only a very modest budget, was to carry out work in some areas for which the European partners were wholly dependent upon the UK facilities and to learn as much as possible of value to future fast reactor development. The programme, as carried out, was in three 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 first two studies were completed on 31 March 1995 and were described in the 1995 paper. The programme to examine materials began in 1994 and the examinations were completed at end March 1996.

These examinations have been restricted to the secondary sodium circuit, since the funding available precluded the study of primary circuit 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.

3.1 Examination of carbon steel components exposed to sodium environments

Following cracking of 15Mo3 steel in the Superphénix storage drum and Kalkar sodium storage vessels, questions remained over the use of as-welded carbon steels for structures in contact with sodium or sodium vapour. Examination of as-welded joints in the PFR carbon steel sodium storage tanks that have been exposed to sodium vapour were carried out to provide some assurance that similar steels, likely to be used for EFR 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.

A suitable vessel that had operated at up to 150°C for a period of several months, prior to the sodium filling of PFR, was selected. The section thicknesses of the vessel were generally 12.7 mm. Following removal of lagging, the weld seams and a number of nozzle welds were ultrasonically inspected from the exterior of the vessel. No defect indications of significance were detected. The man-way nozzle, a smaller temperature connection nozzle and a section containing the junction of a circumferential weld seam and a longitudinal weld seam were removed for examination. These were bagged-off to prevent air ingress and the sodium was removed using an ethanol wash. A bolted patch arrangement was used to effect a repair at each sample removal site. The sections removed were examined with dye penetrant. No indications of significance were found. The metallurgical examination of these samples has yet to be reported but preliminary studies suggest that there is no cracking of the type observed in the 15Mo3 steel, implying that carbon steels in the as-welded condition can operate successfully in contact with sodium without cracking.

3.2 Study of Delayed Reheat Cracking in Austenitic Steel Weldments

Delayed Reheat Cracking (DRC) was the main mechanism attributed to the cracking occurring at 500°C to 520°C in the weldments of the PFR reheater and superheater shells, between 1987 and 1993. It was also a major contributory factor in two earlier cracking incidents in PFR secondary pipework. All of these incidents have involved the Ti-stabilised Type 321 stainless steel.

DRC is expected to be exacerbated by large section thicknesses, high levels of constraint and high heat input welding. The published literature indicates that Type 316 steel is less susceptible to DRC than the Ti-stabilised Type 321 steel of PFR, however, the data base of experience of DRC on Type 321 steel is itself meagre. In particular there was little information on section thickness effects and relatively low temperatures (≤600°C) of operation. Suitable welds chosen for examination in PFR were those of the secondary pipework and a superheater vessel shell. The aim was to look for cracking, if any, in both good geometry and poor geometry welds of the thin section pipework and to characterise known cracks detected by ultrasonic inspection in the shell of Superheater 3.

A section of pipework on the line from the IHX to the Superheater 3 inlet was removed and bagged off. Repair of the remaining pipework was effected using welded end caps. Three sections, containing known defects, were removed from the Superheater 3 vessel shell and cleaned in ethanol. Repairs to the vessel were made using bolted patches.

The metallurgical examination of the pipework and superheater samples has not yet been reported formally. Some very limited cracking has been found at one of the pipework welds, suggesting that more extensive cracking, possibly leading to leakage, might have occurred over a much longer period than the 56,000 hours of high temperature operation experienced.

A large crack in a circumferential weld of Superheater 3, detected during the first ultrasonic inspection in 1988, and showing negligible growth in the intervening period, was found to be a hot tear in the weld formed during the vessel fabrication. A smaller crack at a longitudinal seam weldment at the junction with a circumferential weld was also examined. This crack had shown progressive growth over a three year period. It had an intergranular characteristic consistent with either DRC or conventional creep crack growth.

3.3 Study of a Secondary Circuit Transition Weld

At present, design rules for transition welds are not well validated. Such welds will be required between the roof and the primary vessel and between the secondary sodium pipework and the steam generators of future fast reactors. There is a history of problems with transition welds in conventional power stations, but this is usually associated with thick section welds operating at typically 600°C, where creep-related failures have occurred in a decarburised zone in the ferritic steel.

A transition welds in PFR which operated at high temperature was selected for examination for microstructural changes. This weld was one between the 2.25Cr1Mo steel evaporator inlet nozzle and the Type 321 secondary pipework steel. These welds were made using Inconel filler and operated at or below 490°C.

Following removal of the lagging from the lower region of Evaporator 1, which was the unit with the longest operational exposure, ultrasonic inspection was carried out on two of the three inlet nozzles on lines from Reheater 1 and two of the three on lines from Superheater 1. One of the two inspected nozzles from the Superheater line was removed, and a repair effected, in the same way as the sodium pipework. The nozzle weld removed was subjected to destructive metallurgical examination. This examination has not yet been reported formally, but no characteristics were found that suggest the onset of any changes which might ultimately lead to failure. This weld had operated for about 67,000 hours.

4. Fast Reactor Development

Participation in the European collaboration continued under the leadership of BNFL with support from Nuclear Electric, NNC, and AEA Technology. This involved

- BNFL and NE in the activities of the European Fast Reactor Utilities Group (EFRUG),
- NNC in EFR Associates' reactor design studies, mainly in the areas of rector physics, safety, in-service inspection and repair, containment, decay heat rejection, and feedback of experience from PFR,
- BNFL, NNC and AEA Technology in R&D studies, mainly in connection with the CAPRA project and associated fuel cycle strategies (reported in more detail below), and
- Secondees with EFR Associates at Lyon and CEA at Cadarache (MASURCA).

Collaboration with Japan takes the form of a continued attachment at the MONJU site, and participation in studies of the safety of advanced reactor systems.

In addition the activities of the IWGFR have been supported.

5. UK Activities in Support of the CAPRA Project

The CAPRA project is described in detail by our French colleagues. This report concentrates on the UK contribution.

At the end of 1994 the feasibility of a mixed oxide core designed to consume around 70 kg of plutonium per TWh of electrical generation was established. In 1995 the work was extended to refine the design of the oxide core, to examine the feasibility of a high-consumption core containing no uranium, and to investigate the effect of plutonium-consuming on nuclear fuel cycle logistics in Western Europe.

5.1 Reactor Physics

The oxide core has considerable flexibility which can be exploited for various purposes. One attractive possibility is to increase the fuel residence time. This may be possible because, compared with a conventional fast reactor core such as EFR, the ratio of clad dose to burnup is low. If the limit on irradiation is set by dose rather than the fraction of heavy atoms fissioned the fuel residence time could in principle be increased by about 30%. The sodium void coefficient would be increased because of the higher concentration of higher plutonium isotopes, but this can be compensated for by increasing the amount of moderator in the core. Studies by NNC have shown that, provided the fuel performance can be demonstrated experimentally, the high burnup version of the mixed-oxide CAPRA core appears to be feasible and has attractive features.

An obvious disadvantage of a core without uranium is the loss of the negative Doppler coefficient associated with U 238. This is offset partly by the negative Doppler of the higher Pu isotopes, and partly by the fact that positive feedback from the sodium void reactivity is lower. Studies by AEA Technology have shown, however, that there are difficulties in calculating the Doppler effect in non-uranium cores, and have indicated how standard calculation routes may need to be improved. There is a world-wide lack of experimental data on the Doppler effect of Pu 240.

5.2 Reactor Safety

Because of the heterogeneity there are difficulties in predicting the performance of the mixed-oxide core in a core-disruptive accident. The lead in tackling this problem has been taken by our German colleagues, and UK organisations have participated in the discussions to set the context of these aspects of the safety assessment.

NNC have made a detailed study of the transient response of a preliminary non-uranium core design to a range of design-basis and beyond-design-basis accidents. As expected the response of the intact core is strongly affected by the small Doppler coefficient, especially in the case of "high-quality" plutonium which has lower Pu 240 content. Nevertheless for a range of initiating conditions core damage and fuel melting can be avoided provided there is adequate negative feedback from a source such as the expansion of the control rod supports . The reliability and acceptability in a safety case of this form of negative feedback would of course require further investigation.

5.3 Fuel Performance

It is an objective of the CAPRA project that the fuel should be compatible with the PUREX reprocessing route. For the mixed-oxide core this has limited the plutonium fraction of the fuel to 45 % in order to preserve solubility in nitric acid. For the non-uranium core it has ruled out oxide as a fuel material, so that interest has centred on plutonium nitride fuel.

Compared with oxide, nitride has high thermal conductivity and high density. The high conductivity results in lower fuel temperatures. This is advantageous in that it tends to guard against dissociation of the nitride which may be a problem at high temperature, but may also have the drawback that more fission-product gas is retained within the fuel matrix.

The high density is a disadvantage to a CAPRA core. Compared with a conventional fast reactor core the power per unit mass of fuel is high because the diluting effect of the uranium has been lost. Fuel pin and subassembly designs have to be adapted to accommodate the very high power density, and high fuel density exacerbates the problem.

AEA Technology have been involved in determining the properties of nitride fuel materials and its behaviour in the reactor, both in normal operation and in off-normal transients.

5.4 Fuel Cycle Scenarios

It is necessary to support the studies of plutonium-burning reactors of the CAPRA type by a clear understanding of the effect the deployment of such rectors would have on inventories of plutonium and other actinides and waste products. Studies of this type have been undertaken in the UK and other countries for some time. During 1995 a combined West-European assessment was started in connection with the CAPRA project.

NNC have set up a model of the flows of fuel cycle materials within the nuclear power industry. Initially this was applied to the UK, but it has been applied to Western Europe to show the effect of plutonium burners on plutonium stocks and waste arisings, in comparison with other options such as recycle of plutonium in LWR reactors in the form of MOX fuel. Plutonium stocks can be reduced by CAPRA reactors but it is clear that, if nuclear power generation is assumed to continue indefinitely into the future, there is a minimum plutonium stock level which has to be maintained.

5.5 The Third International CAPRA Seminar

In November 1995 BNFL hosted a seminar at Lancaster at which the results of the work done in many countries in connection with the CAPRA project were exchanged and discussed. About 60 delegates from 8 countries attended. 29 papers were presented in the course of two days of discussion. The meeting concluded with a visit to the BNFL THORP reprocessing plant at Sellafield.