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DECOMMISSIONING OF NUCLEAR FACILITIES

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INTRODUCTION

1. The subject of decommissioning redundant nuclear facilities is one which is receiving increasing international attention and the current application to the problems reflect the responsible attitude of the nuclear industry to provide solutions on which publicly acceptable policies can be formulated.

2. The UKAEA in collaboration with other outside organisations is undertaking a programme of development issues and practical aspects of dealing with nuclear facilities no longer required within the United Kingdom. The programme is designed to cover all types of nuclear installations but has concentrated initially on reactors.

3. At the present time world experience of decommissioning reactors is comparatively limited. A number of early, low power, units have been closed down, and taken to various stages of decommissioning. The largest reactor to be wholly dismantled to date was the Elk River Boiling Water Reactor (22.5MWe) in the USA. Nevertheless there is a wealth of additional experience of working under active conditions during plant maintenance, modification and adaptation which has a direct relevance to the type of work involved in decommissioning.

4. The various types of UKAEA experimental reactors (WAGR; DFR and SGHWR) in support of the nuclear power development

programme, together with the currently operating 26 Magnox reactors in 11 stations totalling some 5GW will probably be retired before the end of the century and attention is focused on these. The actual timing of withdrawal from service will be dictated by development programme requirements in the case of experimental reactors and by commercial and technical considerations in the case of electricity production reactors. Decommissioning aspects were not a primary concern in the design of these facilities, but the lessons to be learned in dealing with them could well contribute to the design without prejudice to safety or performance of future systems to reduce the complexity of decommissioning problems.

5. The UKAEA selected WAGR as the initial reactor for decommissioning studies and a similar study is being undertaken by CEGB of a typical steel pressure vessel magnox station.

RADIOACTIVE INVENTORY

6. To identify the development issues and practical aspects of decommissioning requires, among other data, a knowledge of the total radioactive inventory and its decay together with its distribution within the system. The inventory is made up of the neutron induced activity of the reactor structure, contamination within the system arising from burst fuel or activated corrosion products, and stored operational activated/contaminated waste.

7. An estimate of the activity of the structure can be made on the basis of calculation but the accuracy is dependent upon a knowledge of the abundance of trace elements in the construction materials which are essentially mild steel, stainless steel, concrete and graphite. The composition of the last is known to a high degree of precision due to the nuclear specification and the analytical control. The steels are less well defined and advice was sought on the reasonable contents of inevitable trace elements. The calculations which have been carried out for both WAGR and the commercial station are being validated by measurement of material samples taken from the reactors. They are however at this juncture considered sufficiently accurate to draw general conclusions. However, for future plant it would be clearly advantageous to have available data on the quantities of trace elements in construction materials.

8. Circuit contamination is dependent upon the operational history of the reactor and can only be estimated on the basis of sampling. The activity and quantity of stored operational waste should be identified from records.

MAGNITUDE OF PROBLEM

9. The following table gives the approximate weight of the major materials of construction of the two reactors which have been considered. It does not include generation and other external plant which can be disposed of by conventional means, or operational active waste.

TABLE I - TONNES

ITEM	WAGR			
	MILD STEEL	STAINLESS STEEL	GRAPHITE	CONCRETE
1. PV and internals	600	40	300	
2. Biological Shield	200			4000
3. Heat Exchangers	150 each (4-off)			

ITEM	COMMERCIAL MAGNOX STATION			
	MILD STEEL	STAINLESS STEEL	GRAPHITE	CONCRETE
1. PV and internals	2500	100	2500	
2. Biological Shield	800			16000
3. Heat Exchangers	345 each (4-off)			

10. The major activation occurs in the steel of the pressure vessel and the internal structure. In the case of WAGR the activity is mainly concentrated in the stainless steel components. In the commercial station the activity is more generally distributed and is accounted for by the differences in design and the relative proportions and distribution of stainless and mild steel. The total activity decay, excluding the biological shield, over 350 years as currently assessed for WAGR is shown in Figure 1. The initial decay over the first 40 to 50 years is dominated by Fe 55 (half life 2.6 years) and Co 60 (half life 5.27 years) which are then superseded by Ni 63 (half life 92 years) as the principal isotope. The exponential decay of the system during this second phase is equivalent to a halving time of about 100 years. The β activity decay follows the total curie decay but the γ activity stabilises at a virtually constant value after 100 years.

11. The degree of activation of the biological shield concrete differs between the two reactors due to the differences in their internal shielding features. The thickness of concrete measured from the internal face of the WAGR which will be active after 2 years shut down is calculated to be about 1 metre. The corresponding thickness for a commercial magnox station is about 1.5 metres. Although decay will reduce these thicknesses the weights involved will be large, particularly in the case of commercial stations, and when demolition is undertaken will present the major disposal problem in terms of mass.

12. The degree and distribution of the contamination (Co 60, Cs 137 and Cs 134) within the four WAGR heat exchangers are known. The radiation associated with the contamination, the size of these components (20 metres high by 3.3 metres diameter), and the tube surface area and geometry pose a major problem of decontamination. The magnitude of a similar operation assuming some internal contamination of commercial station heat exchangers which are 23 metres high by 7.3 metres diameter is even more demanding. These items are not activated and decontamination if practical or dilution with inactive material, of all or part of them to the recycling level of 10^{-11} Ci/gm (Ref 1), would reduce the final disposal burden. In this context however the resultant active effluent arising from decontamination would itself require disposal.

DISCUSSION OF DECOMMISSIONING

13. Decommissioning studies are based on a progressive procedure by adoption of the now accepted definitions (Ref 2) of Stage 1, Stage 2 and Stage 3 relating to the status of decommissioned facilities.

14. Whilst describing these as three progressive stages, it should be noted that Stage 1 is an interim phase. Stage 2 is a storage situation, the duration of which may be influenced by environmental pressures or by the economic attraction of reusing an existing site. The degree of decommissioning undertaken within the principles of Stage 2 will be dictated by the time interval, if any, before continuing to Stage 3, which must be the ideal ultimate objective.

Stage 1

15. Stage 1 decommissioning in effect renders the reactor safe and substantially intact on a care and maintenance basis backed by the appropriate degree of monitoring.

Stage 2

16. Stage 2 decommissioning as defined in 13 requires detailed evaluation since the residual structure is essentially the biological shield reinforced where necessary, containing the active components of the reactor. The primary objective is to ensure that the residual structure cannot create a personnel or environmental hazard.

17. The study of the decommissioning of WAGR to Stage 2 assumed the removal of all plant and equipment external to the reactor biological shield including the heat exchangers, the reactor containment shell and all external buildings and facilities. The residual structure would be a 15 metre diameter by 15 metre high cylinder which would occupy about 1/50 of the present WAGR site area and would reduce the visual impact by a factor of 6 based upon the presented vertical area of the cylinder compared with the 41 metre diameter containment shell which currently dominates the WAGR complex. The CEGB will carry out its own assessment for the application of the principles of Stage 2 to commercial reactors but it is unlikely that the conclusions in spite of design differences will differ in principle from those applicable to WAGR. It is worth noting that steel pressure vessel Magnox stations irrespective of output capacity will if reduced to Stage 2 result in cylinders all of approximately 30 metre diameter and heights above ground of between 18 to 30 metres. Each station has twin reactors and the total area occupied by the residual structures would reduce the area occupied by the buildings of the operating station by a factor of about 40.

18. The most economic Stage 2 situation would be to leave the plant with the reactor vessel and the interspace between the latter and the biological shield filled with air and with no external ancillary operating plant except for monitoring facilities. The main factors which are associated with this condition were examined for WAGR and are discussed in subsequent paragraphs.

Temperature

19. The residual heating in the system associated with activity after 2 years shut down is less than 1KW. Assuming no forced cooling of the graphite and applying a simplified model based on pessimistic assumptions it was calculated that the graphite temperature would not exceed 40°C. This temperature is well below the minimum graphite operating temperature (230°C) and hence there is no possibility of spontaneous energy release from the graphite or of its combustion. Graphite temperature monitoring would be maintained as a safety measure.

Structural Integrity

20. The structural integrity of the reactor is basically dependent upon the corrosion of steel. Since the site is coastal a pessimistic corrosion rate of .075 mms per year on each exposed surface (ie .15 mms total thickness) was assumed. Ignoring corrosion retardant factors such as temperature due to residual activity, major component failure periods have been estimated. It is concluded that the integrity of the reactor pressure vessel and its supports would be satisfactory for at least 100 years.

21. The sealing of penetrations through the biological shield, eg after cutting through gas ducts to release the heat exchangers, is essential. The top of the biological shield would be capped with concerete with an access provided to enter the reactor top void space for inspection.

22. The biological shield is a large reinforced concrete structure and based on available experience of this material no significant problems of deterioration should arise for at least 50 years if it was left exposed to weather. This period could be substantially extended by the construction of a relatively light weight structure around and braced to the biological shield to afford protection.

Radiological Aspects

23. No radiation hazard should exist at the external face of the biological shield after fuel removal and all penetrations have been sealed and checked.

24. Throughout the Stage 2 condition corrosion both within the pressure vessel and external to it could produce loose particulate activity. This is all contained within the sealed biological shield the penetration of which by particulates to create an environmental hazard is discounted. The contamination of ground water from this source which is an alternative route to the environment is referred to in para 26.

25. The possibility of radiolytic chemical reactions within the pressure vessel between the air constituents and materials

of construction, eg graphite, cannot be discounted absolutely. They are however unlikely to occur to any significant degree since the radiation fields are relatively low. The reactor vessel and the interspace between the pressure vessel and its biological shield would however be equipped with sampling points for routine atmosphere monitoring.

26. The barrier to any contamination reaching the surrounding ground water is the steel diaphragm floor beneath the reactor, which would require to be made weather-tight at ground level with provision made for access, maintenance, and monitoring.

27. Access to within the biological shield of the reactor under Stage 2 conditions would only be available through facilities engineered to permit monitoring or inspection. These entrances would be secured and only used for a permit system by authorised persons. A fence would be erected around the structure to define the boundary of the installation.

28. An engineering logic sequence taking account of the factors outlined above has been prepared but has not been detailed or costed. The primary objective of the study was to assess the technical considerations required to establish a safe long term Stage 2 condition. The major problem in achieving this is the handling and disposal of contaminated heat exchangers.

Stage 3

29. Whatever period elapses before Stage 3 is commenced after reactor shut-down the two factors which dominate the technical approach are the residual activity which could govern disposal routes of components, and the associated radiological hazards to persons and the environment which would require to be countered during dismantling operations. Delay must assist in reducing the problems due to the natural decay of the radioactive inventory of the system and in this context it should be noted that the extended period of fuel discharge (2-4 years) and the preparatory work for dismantling will automatically introduce a useful decay period particularly in relationship residual short lived high energy nuclides.

30. An inspection of Figure 1 which refers to the total system indicates that after about 100 years delay no further advantage would be gained in relationship to the reduction of γ activity. This however should not be accepted as a criterion for decommissioning since the system design will provide varying degrees of internal self shielding which together, where necessary using remote handling facilities and shielding, reduce radiation levels to personnel engaged in dismantling to acceptable values.

31. The UKAEA is at the preliminary stage of examining the practical problems of decommissioning WAGR to Stage 3 as a

continuing process from reactor closure. The approach which is being applied is to prepare a demolition plan on the basis of engineering logic. The plan is then examined against the known or assessed magnitude of the radiation/contamination problems which will arise at the various demolition stages and consideration then given to their solution by methods, eg remote handling, shielded working or controlled access, which do not entail modification to the engineering logic. Only if at particular stages such methods prove impracticable will there be a departure from the strict engineering logic and the overall plan amended accordingly.

32. The radiation levels which will arise as dismantling proceeds have been estimated for a 7 year decay period. Since the radiation will arise almost exclusively from Co 60, correction factors for longer delay periods are relatively easy to calculate. Conceptual engineering studies for specialised equipment and the modification of existing facilities to dismantle the reactor have been initiated. It is, however, relevant to comment that demolition will probably not require the development of any new technology but rather the adaptation of existing techniques.

33. It is important to appreciate that the engineering logic differs between leaving decommissioning at Stage 2 for an unspecified period and an on-going approach to Stage 3 particularly in the retention and adaptation of existing plant facilities. Hence if the policy relating to the fate of a reactor can be declared well in advance of retirement it should be possible to select the optimum plan for decommissioning.

DECOMMISSIONING WASTES AND DISPOSAL

34. The activity of decommissioning waste from reactors is virtually all neutron induced which is exclusively β γ although some contamination may be present. This activity coupled with the mass and bulk of the waste poses major disposal problems. The methods available are limited to land disposal or sea dumping. However no firm statement can be made at this juncture in relationship to UK policy for disposal of decommissioning wastes. This topic is under consideration in the current review of the Government White Paper - Control of Radio Active Wastes (Cmnd 884) which is an advisory document and forms the basis of UK practices. The recommendations following this review will undoubtedly have taken into account both national and international opinion. In addition a Royal Commission on Environmental Pollution recommends that a Nuclear Waste Disposal Corporation should be established charged with the responsibility for the safe disposal of all wastes arising at Nuclear Sites (Ref 3).

35. Graphite is of particular interest since it is combustible and is a large inventory item of commercial gas-cooled reactors. The disposal of this material by burning requires further investigation to assess whether the C14 activity would prove acceptable on the basis of dispersed atmospheric pollution.

36. It has been suggested that redundant reactors should be used as disposal facilities using the void space between the pressure vessel and the biological shield plus any available space inside the pressure vessel. The storage space could be used to accommodate heat exchangers after cutting these into suitable sizes. Disposal by definition is a permanent situation, ie there is no intent to recover from the disposal site. This in turn requires that the long term integrity of the concrete forming the biological shield can be guaranteed since some residual activity will always be contained within it. If however there is an intention to recover items ultimately then the reactor effectively becomes a silo.

REGULATORY CONTROL

37. The controls under the site licence would remain in force until it could be shown that there was no danger from ionising radiations from anything on any part of the site; at that point surrender of a licence would be accepted.

COMMENT

38. This paper has been confined to the general aspects of decommissioning of early graphite moderated - carbon dioxide cooled reactors which will be the first redundant reactors arising from the UK power programme. Costs have not been discussed since studies in more depth will be required before valid estimates can be derived.

39. Whilst decommissioning has not been a primary consideration in the past, more attention should be given in future to the design of reactors for dismantling, and in power station layouts to optimise land re-utilisation.

40. No responsible nation would wish deliberately to hand on to successive generations the legacy of unsolved problems of dealing with redundant nuclear plant. In the case of nuclear reactors on the evidence available to date, these can be rendered safe and ultimately removed and disposed of. Although the cost of decommissioning could be substantial it does not present as great an uncertainty in the lifetime economics of a nuclear power station as do other factors, eg the precise period over which the reactor will be operable at full load factor.

REFERENCES

(1) A W KENNY., N T MITCHELL., United Kingdom Waste Management Policy (IAEA-SM-137/62).

(2) Decommissioning of Nuclear Facilities, IAEA-179(Section 2).

(3) Royal Commission on Environmental Pollution - SIXTH REPORT - SEPT 1976, Recommendation No. 41 (P 204).

TOTAL SYSTEM DECAY

FIG. No. 1

CURIE & MeV