HEALTH PHYSICS EXPERIENCE IN THE OPERATION AND MAINTENANCE ON AN ON-LOAD REFUELLING PROGRAMME ON UNITED KINGDOM CIVIL POWER REACTORS

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Abstract—On-load refuelling, an inherent factor in the design of the United Kingdom Civil Power Reactors, was first carried out at Bradwell Power Station in April 1963. Although information regarding dose rates around the charge/discharge machine had been measured during the movement of the active absorber material to and from the core, this did not simulate the effect of the delayed neutron spectrum from freshly withdrawn, irradiated fuel, nor could this effect be calculated with any degree of certainty.

Shielding surveys were made using conventional gamma dose rate measuring instruments, whilst the more complex neutron dose contribution was assessed using BF3 counters, Basson intermediate energy detectors, proton recoil counters, and the Andersson-Braun rem counter. At full power, the dose rate from a fuel element being withdrawn from the reactor was measured and the decay curve plotted. The dose rate due to the delayed neutron fraction was shown to decay almost completely within two minutes of withdrawing the element from the neutron flux.

Although measurements inside the shielding close to the fuel element guide tube indicated peak dose rates up to 10^5 R/h, the results of the on-load fuel handling programme have shown that the delayed neutron contribution, although significant, does not limit access to the reactor pile cap during discharge of fuel at full reactor power.

Occupational exposure associated with on-load fuel handling results mainly from activation of various charge machine components. Experience in limiting the accrued dose to maintenance personnel repairing activated fuel handling grabs is discussed in detail, whilst the use of film and thermoluminescent dosimeters for dose control has allowed work to be undertaken on relatively high dose rate components with confidence. The need for a more accurate beta sensitive monitoring device for this type of work is emphasized.

The Health Physics requirements associated with ancillary fuel handling facilities are discussed together with experience in the recovery of damaged irradiated fuel from a charge machine and subsequent decontamination procedures. The need for an integrated approach to the design of fuel handling plant and the associated reactor ancillary equipment is stressed.

1. INTRODUCTION

There are nine nuclear power stations in the first stage of the United Kingdom Nuclear Power Programme. Each station has two gas cooled, graphite moderated reactors, the fuel being natural uranium canned in magnox. The design is based on operational experience gained at Calder Hall and Chapelcross. Station electrical outputs range from 300 MW to 1200 MW. Each new station commissioned has achieved an appreciable increase in electrical output, whilst a significant reduction has been achieved in physical size and capital cost per kW of installed capacity.

Because of their inherently low operating costs, the civil reactor stations are designed to operate on a "base" or constant load principal. To reduce outage time, the reactors are designed for refuelling whilst at full power. The fuel charge/discharge programme is based on an

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equilibrium rate which is related to the energy output of the reactors. At Bradwell Power Station it is necessary to recharge an average of two channels of fuel in each 24 hr period to maintain the equilibrium rate and avoid creating a fuelling back log. This assumes a maximum channel average irradiation of 3600 MW days/Tonne and a fuel dwell time within the reactor of six years.

Although a considerable volume of information existed regarding the radiation levels associated with fuel handling on a shutdown reactor, little information was available regarding the dose rate contribution from delayed neutron capture gammas associated with fuel as it was being removed from the core of an operating reactor, nor could this doserate contribution be calculated with any degree of certainty. Hence the performance assessment of the fuel handling equipment was an important aspect of the station commissioning programme. Due to the novel aspect of on-load fuel handling, a detailed health physics investigation was carried out in conjunction with this assessment. These tests were designed to determine the dose rate contribution associated with the delayed neutron spectrum, the effectiveness of charge machine shielding, and the extent of contamination arising from the fuel handling procedures. With on load refuelling now in routine operation, the main Health Physics problems are associated with maintenance of fuel handling components, although a number of non-routine operations such as the recovery of irradiated components from the machine or reactor pressure vessel have provided useful experience in high level radiation and contamination control.

This paper is based mainly on experience gained at the Bradwell and Trawsfynydd Generating Stations, supplemented where necessary with information from other U.K. Civil Nuclear Stations.

2. FUEL HANDLING SYSTEM

Experience described in this paper may be prefaced by a broad outline of the fuel handling system in use at a typical United Kingdom power station such as Bradwell. New fuel is delivered by road from the U.K.A.E.A. Fuel Fabrication Plant at Springfields and stored in an unirradiated fuel store until required for use. From here it is transferred to a fuel preparation room where it is inspected prior to loading into the charge machine.

The charge/discharge equipment consists of a single machine, 9 ft in diameter and 55 ft high, weighing 430 tons, together with a number of auxiliary components for coupling the machine to the reactor vessel and providing adequate shielding during fuel movement. It is designed to provide a shield against radiation and a seal to prevent loss of coolant gas from the reactor during the removal and insertion of components into the reactor vessel. The machine travels along rails on a moveable gantry which spans the reactor pile cap and an enclosed maintenance bay is provided adjacent to this area.

The charge machine consists of a pressure vessel, mounted inside a biological shield. This vessel contains a rotatable inner core or turret extending down through the vessel, fitted with vertical storage and guide tubes, somewhat analogous to the chamber of a revolver, with hoists and turret drives located on the top.

A chute head box is used to form a gas tight connection between the charge machine and any one of the reactor vessel charge standpipes. When connected to the reactor, additional shielding is provided between the pile cap floor and the bottom of the charge machine by means of an annular shield consisting of an upper and lower section, each filled with iron shot concrete. This shield encloses the chute head box, and the upper section is raised telescopically to mate with the base of the charge machine, thus forming a complete biological shield around the base of the charge machine assembly.

The upper sections of the charge machine shielding are of Barytes concrete, whilst the lower sections are of iron shot concrete with additional steel shielding around the base. A slide valve through which all components entering or leaving the machine must pass is fitted in the base of the machine. This valve also acts as a shield and a gas-tight seal when closed, allowing the machine to be disconnected from the reactor while maintained at reactor pressure and containing active components.

After a refuelling cycle, irradiated fuel is lowered from the charge machine through a shielded discharge tube into a skip, positioned in the cooling pond beneath the reactor building. Facilities are available at this point for selecting and bottling damaged elements. Undamaged fuel is desplittered to remove the magnox splitter fins and braces, after which it is stored in the pond for approximately 100 days. It is then loaded into a shielded transport flask and returned via road and rail to the U.K.A.E.A. reprocessing plant at Windscale. Outline diagrams of the charge machine, make-up shielding and interconnections to the reactor are shown in Figs. 1, 2 and 3.

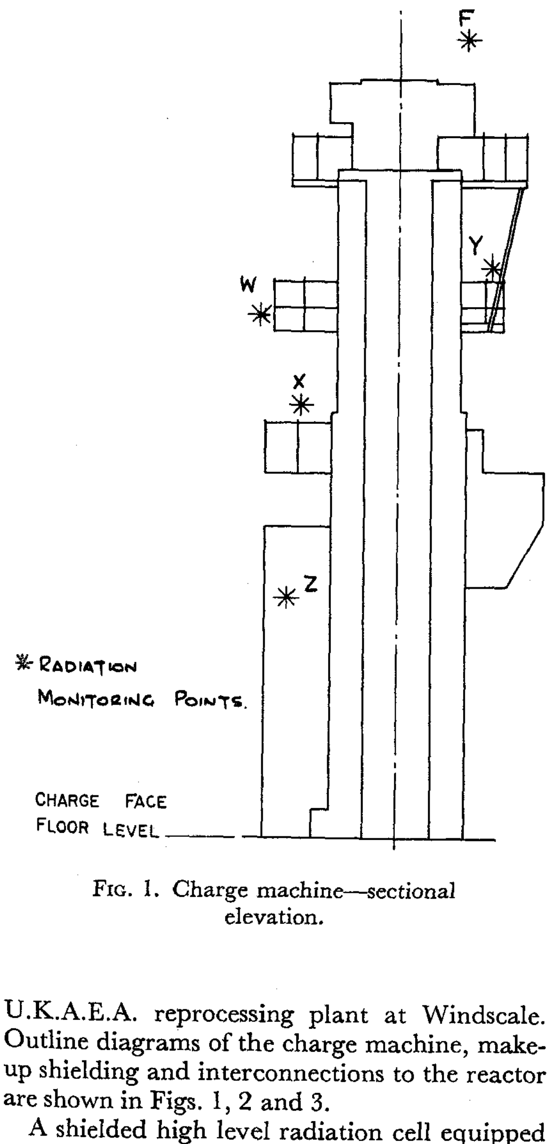

FIG. 1. Charge machine—sectional elevation.

A shielded high level radiation cell equipped with remote control manipulators is provided for maintenance of fuel handling equipment. The charge machine can be used to lower fuel handling components into this facility via a shielded maintenance hole on the reactor pile cap. The entire fuel handling plant is adequately interlocked to ensure safe operation and to prevent any possibility of inadvertent radiation exposure to operating staff.

3. HEALTH PHYSICS MEASUREMENTS DURING CHARGE MACHINE COMMISSIONING

The Health Physics measurements carried out during commissioning of charge machines were aimed at providing the following information:

(i) Basic integrity of the charge machine shielding for storing a full complement of irradiated fuel.

(ii) The extent of neutron and gamma radiation streaming from the core under on load refuelling conditions.

(iii) Decay rates associated with:
  (a) Delayed neutrons.
  (b) Fission product and capture gammas.

(iv) Dose rates on and around the fuelling machines with special reference to radiations referred to in (iii).

(v) The extent of radioactive contamination associated with refuelling operations.

Measurement positions were chosen as shown in Figs. 1 and 2 to provide a comprehensive survey of the radiation levels that would occur during the various stages of an absorber or fuel handling programme. The majority of measurements were made in the normal working areas around the charge machine and associated make-up shielding. Although access into the area below the pile cap floor is not permitted during charge machine operation, there was considerable interest in making measurements in this area due to the absence of shielding around the reactor standpipe. This made it possible to measure with reasonable accuracy the dose rate from items being removed via the standpipe into the charge machine. Measurements were also made of radiation levels associated with operating equipment and components common to both absorber and fuel handling, such as the charge chute and grabs. These latter measurements provided useful information on the radiological conditions

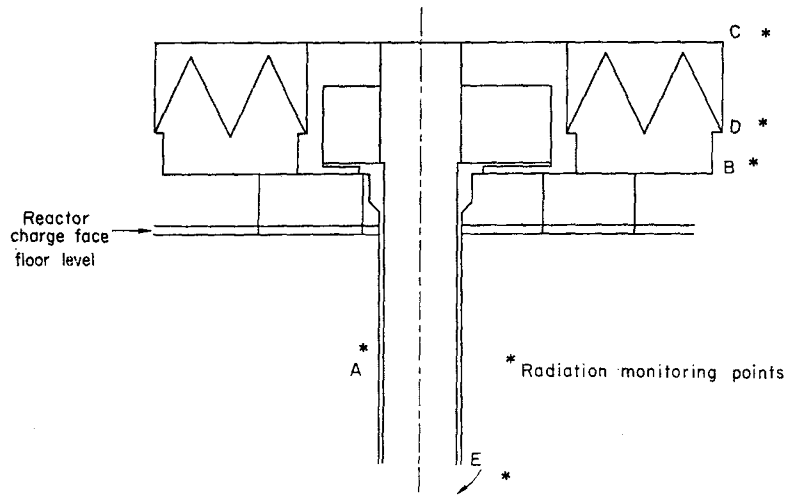

FIG. 2. Make up shielding with control rod sealing sleeve.

associated with the maintenance of charge machine components.

The measurements made and the results obtained are discussed in the following sub-sections.

3.1. *Charge Machine Shielding*

Radiation measurements were made during various on-load absorber changing operations. The transient dose rate from the non-fissile absorber being discharged, measured at the surface of the standpipe, ranged between 350 R/hr and 1800 R/hr for the discharge of No. 1 and No. 5 absorber respectively, whilst the corresponding maximum radiation levels in the pile-cap working area adjacent to the machine make-up shield were 0·5 mR/hr and 1·5 mR/hr respectively. A time lapse of approximately 2 min occurs between the commencement of withdrawing an absorber (or

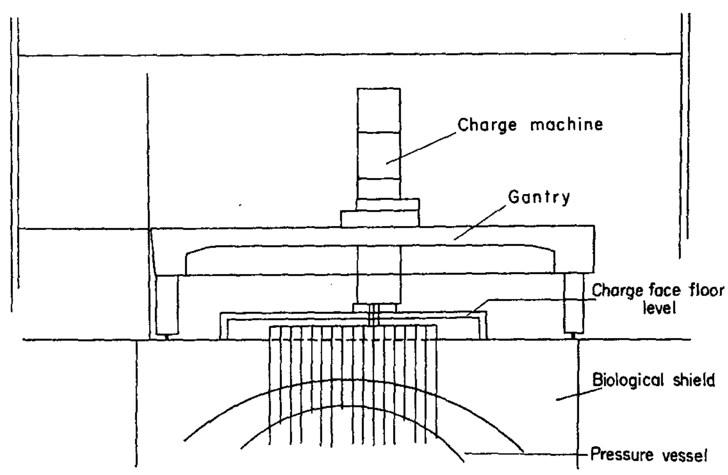

FIG. 3. Charge machine and gantry showing interconnection to reactor pressure vessel.

fuel element) from the reactor core, and its subsequent arrival at pile cap level, prior to entering the charge machine.

^{59}Fe, ^{60}Co and ^{56}Mn are the predominant activities associated with neutron activation of the in-pile fuel handling components such as chutes, grabs, hoist cables and standpipe assemblies. It was estimated that the major dose rate contribution from freshly discharged absorber material would be due to ^{56}Mn. Because of the short half-life (2·58 hr) and relatively high activation cross-section, the manganese will have reached saturation and even a small manganese content, in this case 0·1%, would contribute an estimated gamma dose rate of approximately 1800 R/hr at 1 ft from the surface of the absorber.

Surveys during on load discharge of absorber material and during early off load refuelling exercises showed that there were no major shielding weaknesses in either the charge machine or associated fuel and absorber handling facilities. Particular attention was paid to the storage magazine areas of the machine when loaded with a full complement of irradiated absorbers. A maximum radiation level of 40 mR/hr was measured on top of the machine with the storage magazine empty but with the charging chute stored in the machine. As this area is unoccupied during charge machine operations, these levels do not contribute to a hazard.

3.2. *Core Streaming*

A detailed core streaming survey was carried out on the charge machine between the period of removal of the shield plug and insertion of the fuel chute. The shield plug was held at several chosen positions whilst surveys were carried out with the reactor at power.

The highest levels recorded were of the order of 50 mrem/hr neutron and 50 mR/hr gamma on top of the machine in an area not normally occupied by personnel. These radiation levels existed only with the shield plug removed and with the magazine in a specific orientation.

Radiation levels up to 20 mrem/hr neutron and 1 mR/hr gamma were detected at localized points on the machine at first platform level above the pile cap.

Radiation surveys in working areas adjacent to the machine at pile cap level showed radiation levels only slightly above background.

3.3. *Decay Rates of Delayed Neutrons and Fission Product Plus Capture Gammas*

The decay rates for delayed neutrons and for fission product plus capture gammas were determined by arresting a fuel element in its traverse from the core to the machine and holding it near the base of the machine.

At Bradwell the measurements were made close to the standpipe and were expectedly high. The measurements at Trawsfynydd were made on the pile cap level outside the machine but in a region close to the machine shielding.

The decay curves so determined are shown in Figs. 4 and 5.

Regarding the neutron decay, this shows an initial half-life of approximately ½ min followed by a half-life of almost 1 min after 2 min decay (cf. classically determined delayed neutron half-lives of 22.5 sec and 55.6 sec). The gamma decay curves showed an initial half-life of approximately 1 min.

The major part of the total dose rate (neutron plus gamma) decayed after the first 3 min from withdrawing a fuel element from the core.

3.4. *Dose Rates due to Delayed Neutrons and Fission Products Plus Capture Gammas*

Most of the neutron dose rate measurements made at U.K. civil nuclear power stations are made with the Andersson–Braun rem counter which has a good energy response over the range 0·1 eV to 10 MeV. Using this instrument, comprehensive neutron dose rate surveys were made and transient dose rates determined by extrapolation using the decay curves already determined. These results show:

(a) Negligible transients are experienced at the pile cap operating floor level.
(b) Localized transients as high as 1 rem/hr are experienced above pile cap operating floor level but these are in areas not normally occupied during a refuelling cycle. If access were required for some reason whilst refuelling, then a waiting period of 3–5 min would render radiation levels acceptable. This limitation has not proved embarrassing operationally.

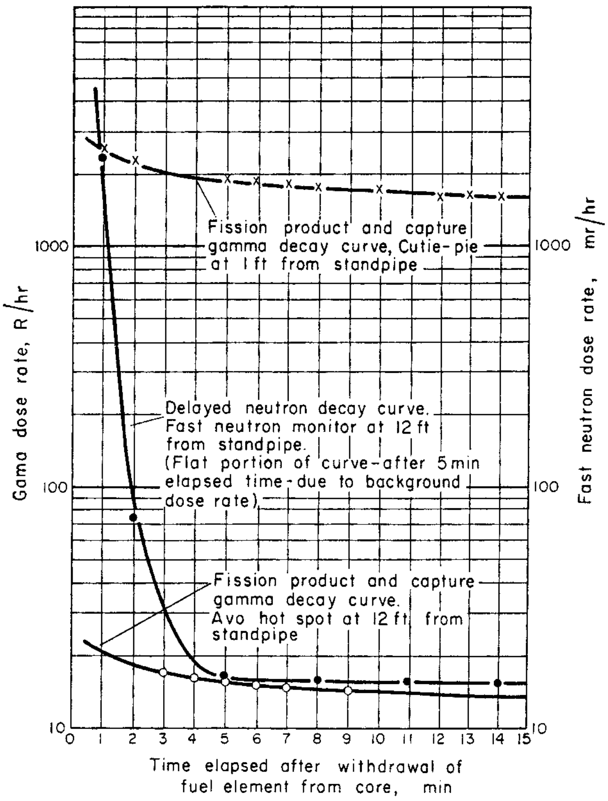

FIG. 4. Capture gamma and delayed neutron decay curves for fuel element withdrawn from reactor whilst at power. Measurements made on Bradwell Reactor No. 1. Reactor power—300 Mw (Thermal). Element Irradiation 1000 MwD/Tonne over 400 days.

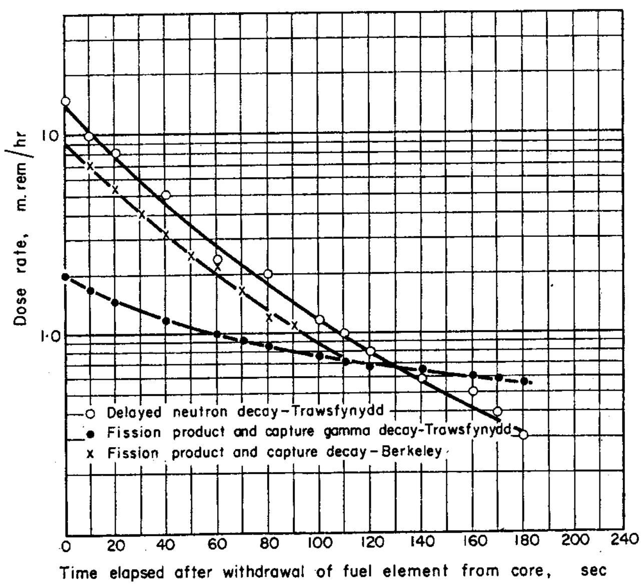

FIG. 5. Capture gamma and delayed neutron decay curves for fuel element withdrawn from reactor whilst at power. Measurements made at Trawsfynydd and Berkeley.

To obtain an estimate of the degree of shielding provided by the charge machine, measurements were made in the vicinity of the unshielded standpipe between the primary and secondary reactor floors whilst fuel was being discharged from the reactor. The dose rate under these conditions, when discharging fuel from the flattened zone at full power, gavc levels up to 50,000 rem/hr (compared with 1800 rem/hr when discharging absorbers). An R.C. 7 ionizing chamber sensitive to both gamma and neutron radiation was used for this measurement, and as there was a considerable fast and thermal neutron contribution in the vicinity of the standpipe (of the order of 5×10^4 neutrons/cm^2) this could lead to an overestimation of the gamma contribution by up to a factor of 10 with this type of detector. Calculations using the Way-Wigner equations show that the surface dose rate from a freshly discharged 3500 MWD/Tonne fuel element is of the order of 3×10^6 rem/hr. This does not include the very short-term fission product decay, and for on-load discharge it is most certainly an underestimate. Thus, the value of 50,000 rem/hr measured at a position approximately 1 ft from the fuel element and shielded by approximately 1 in of steel (with a 90 sec decay after withdrawal from a neutron flux of approximately 10^{13} n/cm^2/sec) would appear to be a realistic value.

The maximum transient radiation level of 13 mrem/hr was measured around the make-up shield during the discharge of elements from the reactor at full power, and the maximum steady radiation level obtained during a comprehensive survey of the machine containing 24 freshly discharged elements was 2·6 mrem/hr opposite the element deflector mechanism, these results being a factor of 10 down on the predicted design radiation levels. Tables 1 and 2 give a summary of the radiation levels obtained when withdrawing irradiated fuel.

Table 1. *Radiation levels measured at various positions beneath the pile-cap floor during on-load fuel discharge from Reactor No.* 1, *Bradwell Nuclear Power Station*

Charge machine operation	Position of measurement (see Fig. 1)	Peak radiation level in rem/hr		
		Fast neutrons at 12 ft. from Point A	γ at 1 ft. from Point A*	Contact at Point A†
Reactor thermal power—300 *MW*				
Discharge—No. 8 Element	A	2	1500	5000
7 ,,	A	2·4	2500	6000
6 ,,	A	3	3200	6000
5 ,,	A	3	3300	8000
4 ,,	A	2·6	2800	7000
3 ,,	A	2·3	2000	6000
2 ,,	A	1·6	1100	5000
1 ,,	A	0·75	500	3000
Reactor thermal power—400 *MW*				
Discharge—No. 8 Element	A	1·5		20,000
7 ,,	A	1·9		30,000
6 ,,	A	1·9		40,000
5 ,,	A	2·3	Instrument	50,000
4 ,,	A	2·4	OFF	50,000
3 ,,	A	2·0	Scale	50,000
2 ,,	A	1·7		20,000
1 ,,	A	0·9		10,000

* Measured with Cutie Pie. † Measured with RC7 ion chamber.

An alternative method of measuring high gamma dose rates adjacent to irradiated fuel has been used at Trawsfynydd. This involved the use of the radiation-induced conductivity of cadmium sulphide crystals to measure dose rates over the range $1\text{–}10^6$ R/hr. The ease of placing the detector (no longer than half a cigarette) in awkward positions (e.g. under the charge machine) and the facility to read remotely proved extremely useful. The use of cadmium sulphide crystals for this particular application has been reported by W. H. R. Hudd. [1]

3.5. *Radioactive Contamination Associated with Refuelling Operations*

Contamination of internal surfaces of the fuel handling equipment is bound to occur during fuelling. The charge machine is fitted

Table 2. Integrated gamma doses measured at floor level at various working positions around the Bradwell charge machine during discharge of four fuel channels (32 Elements)

Reactor power—400 MW (Thermal)
Total dose integrating time—8·5 hr.

Distance of measurement from edge of charge machine make-up shield	Integrated dose (mr)
2 metres	22
4 metres	2·5
6 metres	4·5

Gamma dose rates due to streaming, measured on the outside face of the charge machine and make-up shielding

No. 4 fuel element held stationary between secondary floor and make-up shield.
Reactor power—300 MW (Thermal)

Position of measurement (see Fig. 1)	Dose rate—mr/hr			
	West face	North face	East face	South face
B	3·0	0·5	10	H 60*
C	5·0	4·5	5	H 12
D	H 20	13	4	0·9

* Decayed to 18 mr/hr after 40 sec.

H, Make-up shield not sitting completely level on pile cap floor. When level, maximum transient dose rate reduced to 13 mr/hr.

Four channels of irradiated fuel stored within charge machine
Reactor power—300 MW (Thermal)

Position of measurement (see Fig. 2)	Dose rate—mr/hr
W	0·35
X	0·12
Y	0·35
Z	2·6

with a gas tight valve which prevents gross contamination reaching the external environment. Nevertheless, the make-up shielding around the base of the machine and the discharge shielding associated with transfer of irradiated fuel elements from the charge machine into the cooling ponds, acquire considerable surface contamination. The maximum level during commissioning was found to be 3×10^{-2} μCi/cm^2 beta/gamma with no evidence of alpha contamination. No abnormal airborne contamination was detected in the vicinity of the charge machine and pile cap during the refuelling operation.

A fairly high proportion of contamination in the main cooling circuit of the reactors has been identified as low energy beta contamination (mainly ^{35}S). Some of the contamination experienced on fuel handling systems is from this source. This low energy activity is not detected by conventional geiger contamination monitors and other types of instruments and measuring systems have been used. The details of this work are outside the scope of this paper.

3.6. *Summary of Measurements*

Results of the on-load fuel handling survey programme proved that despite the high dose rates associated with fuel elements withdrawn on load, the radiation levels in the working area of the pile cap remained sufficiently low to allow unrestricted access during refuelling operations. Localized points of high dose rate occur but access is not required to these areas during normal operations. As would be expected, the radiation levels measured during the commissioning period have remained sensibly constant during the past 3½ years of on-load fuel handling operations.

4. MAINTENANCE OF FUEL HANDLING EQUIPMENT

Ease of maintenance is essential if the on-load fuel handling equipment is to achieve a high load factor. There are two quite distinct health physics aspects. Firstly, maintenance of large plant items such as the internal areas of the charge machine where the major problem is associated with contamination. This will also include work on make-up shielding, test, storage and disposal facilities. Secondly, maintenance on ancillary fuel handling equipment such as grabs, cables, chutes and standpipe assemblies where the items may have received considerable neutron irradiation and for which dose rate will be the controlling factor. Dose rate can also be the controlling factor when dealing with major plant items in the first category if irradiated components or fuel remain stored within these facilities whilst maintenance is being carried out. It is normal procedure to discharge the irradiated fuel from the charge machine before attempting to carry out maintenance work, although certain break-down maintenance may preclude this.

Experience to date has shown that the limiting health physics factor in the maintenance of fuel handling equipment is due to the dose rate from activated components. This in turn depends upon the dwell time within the reactor flux, activation cross-section of the material, and its half-life. Steel is a major component in all inpile equipment and the initial controlling dose rate for handling freshly irradiated components is due mainly to the activation of ^{56}Mn. Small amounts of brazing often associated with electrical contacts can give rise to high localized dose rates due to the use of high activation cross-section brazing materials. The major long term activity build-up in steel components is due to ^{60}Co and it is this isotope which ultimately limits the amount of maintenance which can be carried out on irradiated components.

4.1. *Repair on Fuel Element Grabs*

The fuel element grab is a complex mechanism, and must provide a reliable performance if a satisfactory fuel handling programme is to be achieved. Much of the maintenance on fuel handling equipment has been associated with grab repairs, and considerable emphasis has been placed on the development of repair techniques and measurement of the accrued radiation dose to the repairer. Figure 6 shows a typical decay curve for a fuel handling grab for various irradiation and decay periods. The initial contact dose rate varies between 2000 R/hr for a grab irradiated for 1 hr in a thermal neutron flux of 10^{13} n/cm^2/sec to almost 10,000 R/hr for any irradiation period in excess of 10 hr. The decay curve follows the ^{56}Mn, 2·58 hr half-life, for approximately 2 days, after which the decay period has a half-life of approximately 30 days. This in turn gives way to a long-term residual activity determined primarily by the decay of ^{60}Co, a typical residual activity value being 7 R/hr per 24 hr period of irradiation.

Due to the mechanical complexity of the fuel handling grab, it is impracticable for any but the simplest repairs to be made using remote handling equipment. However, it has been found advantageous to set up a simple grab handling facility comprising a shielded work bench with a small single arm power manipulator. The more elementary dismantling and assembly operations can be carried out, thus limiting the amount of physical contact time involved. Much of the work, however, involves manual contact with the grab and consequent finger tip radiation exposure. Pre-planning of the repair sequence is an essential feature. Where practical, practice runs are made on a non-irradiated grab to familiarize the operator with the work to be carried out. Working times are established, which in conjunction with measured dose rates taken on the grab to be repaired allow an accurate assessment of the potential exposure.

A typical system for controlling radiation dose to personnel carrying out grab maintenance involves the use of dosimeters as follows:

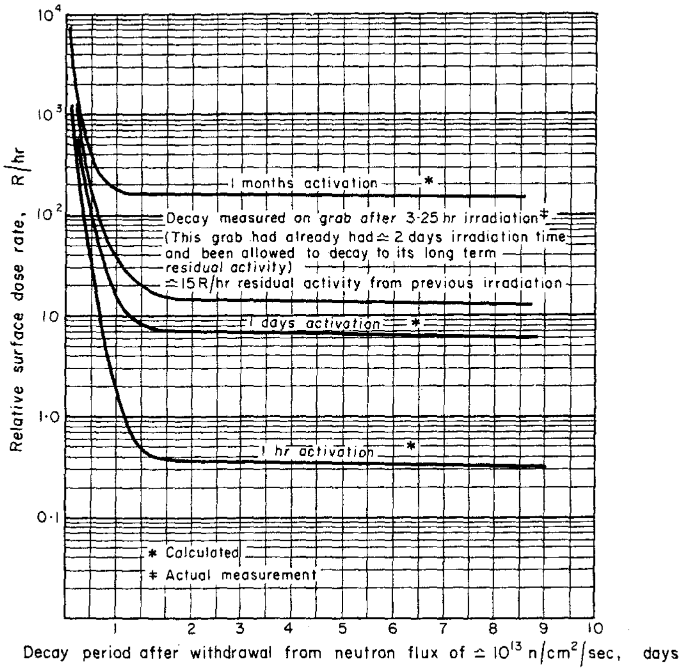

FIG. 6. Comparison of calculated and measured gamma surface dose-rate for Bradwell fuel element grab following various irradiation times.

(a) All personnel involved are issued with a body Q.F.E. (Quartz Fibre Electroscope–Pocket Dosimeter) which together with their monthly film badge measures and controls their whole body dose.

(b) A film badge is attached to each wrist together with a Q.F.E. which can be read by the operator without removal from the wrist.

(c) A lithium fluoride thermoluminescent dosimeter is worn on a finger tip on each hand. These dosimeters consist of flat sachets of approximately 1 cm^2 in area having a plastic cover of 12 mg/cm^2 (comparable with the dead outer layer of skin on the fingers). The dosimeter is wrapped around the finger and does not interfere with finger sensitivity. This type of dosimeter has a good response to both beta and gamma radiation over a wide energy range and is particularly useful for measuring total dose to finger tips when handling components of this type where the finger dose can be much higher than the wrist or body dose.

Experience has shown that the ratio between the finger tip dose and wrist Q.F.E. dose can vary between 5 and 15 to 1 depending on the item handled. For operational control a ratio of 10 is used and the work is controlled on the basis of dose as shown by the wrist and body Q.F.E.s. A typical permissible figure for a job would be 50 mrem on each Q.F.E. (implying a finger tip dose of not greater than 500 mrem). Using a shielded facility to limit head and body dose, finger-tip dose becomes the controlling factor.

Using the facilities and procedures described above and sharing the work load between operatives, it has been possible to dismantle, repair and reassemble grabs with surface gamma

FIG. 7. Wrist and finger dosimeters used when handling active components. Left-hand picture shows lithium fluoride sachet normally worn under protective gloves (see right-hand picture).

FIG. 8. Use of wrist and finger dosimeters (shown in Fig. 7) when repairing an irradiated fuel element grab.

dose rates in excess of 20 R/hr without accruing a whole body dose greater than 100 mrem or a finger dose greater than 500 mrem.

During the past $2\frac{1}{2}$ years of fuel handling operations, the major dose contribution has been associated with maintaining fuel element grabs. The feed back of this information to the designers has done much to eliminate the use of high activation cross-section, long half-life materials wherever practicable. The simplification of mechanical design has allowed shorter maintenance working times. Where maintenance must be carried out on irradiated components it is essential that the designer and the Health Physicist should be aware of the composition of the materials being used.

4.2. *Non-routine Operation and Maintenance*

There will always be a percentage of non-routine operations due to faults occurring within the normal fuel handling programme. Certain of these will lead to additional Health Physics requirements, either as part of the operation or during any subsequent maintenance work. The following case provides a useful example:

Removal of Fractured Fuel Element from a Charge Machine

During the normal procedure of discharging fuel from the reactor into the charge machine, a fuel element became dislodged from the grab and fell across the rotating section of the charge machine magazine instead of into a magazine fuel storage tube. With subsequent rotation of the charge machine magazine, the body of the fuel element was severed into two pieces held together by longitudinal splitter vanes which are used to centre the element within the fuel channel. As there is no means of viewing the inside of the charge machine, the degree of damage to the fuel element could not be determined. However, the assumption was made that a fracture had occurred and the appropriate Health Physics precautions were instituted.

Charcoal loaded filter papers were fitted into the charge machine blowdown lines prior to discharging the CO_2 to atmosphere. The machine was blown down to atmosphere in controlled stages, the filter papers being examined at each stage prior to proceeding to further blowdown. The blowdown took place about 3 hr after removing the fuel element from the reactor core, and examination of the filter papers showed no indication of gaseous fission products. With the charge machine at atmospheric pressure it was positioned over the shielded emergency discharge route between pile cap level and the underground fuel element pond. In an attempt to discharge the distorted element into the cooling pond, it became wedged in the upper part of the discharge well shielding. It was still impossible to view the extent of the damage to the fuel element although limited introscope inspection at this stage indicated that it was bent through an angle approximately 30°. The effectiveness of shielding around the discharge well can be appreciated from the following radiation dose rates measured during this operation:

Directly over emergency discharge well 5,000 R/hr
At edge of gap in telescopic shielding 1·5 R/hr
3 ft from gap in shielding (position of operator) 100 mR/hr

The damaged element was eventually freed and lowered down the emergency discharge well, but was prevented from passing into the pond by handling equipment fitted at the bottom of the discharge well tubing. The element was returned into the charge machine whilst entry was made into the discharge area to remove the emergency cropping gear causing the discharge tube restriction. During this period, a number of small particles were found which gave dose rates between 0·1 and 300 R/hr measured at a few inches. A spectrum analysis identified these particles as small flakes of irradiated uranium.

With the discharge tube restriction removed, the element was lowered into the cooling pond. Only at this stage could the degree of damage to the element be assessed. Severance of the element had exposed several square centimetres of uranium metal, and an attempt was made to prevent contamination of the cooling pond water by lowering the element into a dustbin at the bottom of the pond and covering it with a lid. After 4 weeks the water within the dustbin was sampled and the fuel element then returned with other irradiated fuel to the reprocessing plant. The sample of water taken from the dustbin indicated that there had been no significant leaching of fission products from the fractured area of the fuel element during its 4 week storage period. Because of the small pieces of irradiated uranium found in the vicinity of the fuel handling machine, a thorough de-contamination of the internal sections of the machine was carried out. This was important from two aspects. Firstly, due to the potentially high dose rates that could be associated with small particles of this nature, a contamination/radiation hazard could occur whenever the machine was used. Secondly, if these particles of bare uranium were introduced back into the reactor they could give rise to spurious signals on the burst cartridge detector system and could result in considerable embarrassment to subsequent reactor operations. This entire operation was carried out without any of the operators receiving a radiation dose exceeding 100 mrem but did much to emphasize the need for remote viewing equipment to ascertain conditions inside shielded areas.

5. CONCLUSIONS

The results and experiences presented in this paper have summarized some of the more important Health Physics aspects associated with on-load fuel handling in the United Kingdom. Although the initial radiation levels associated with handling freshly irradiated fuel are expectedly high, they have not resulted in a significant increase in exposure to personnel working within the fuel handling areas. Although there have been several instances where fuel handling components or fuel elements have had to be recovered from the reactor using non-standard routines, there has been no undue radiation exposure to those taking part in these operations. The major dose contribution for personnel has come from maintenance of irradiated fuel handling components and especially from the repair of fuel handling grabs. Care and forethought at the design stage, such as the selection of low activation cross-section materials, and designs, which shorten the time spent on the dismantling and assembly of active components, can do much toward minimizing the radiation exposure of maintenance personnel. When installing expensive and complex fuel handling plant the provision of adequate maintenance facilities, designed to provide the maximum of radiological protection consistent with economic working, should be adopted as a general principle. Finally, as testimony to the success of on-load fuel handling, at Bradwell Power Station over 26,000 fuel elements have been discharged during the past three years without any person receiving a dose in excess of the I.C.R.P. recommendations for occupationally exposed workers.

REFERENCE

1. W. H. R. Hudd. Central Electricity Generating Board, Berkeley Nuclear Laboratories Report RD/B/N567 (1966).