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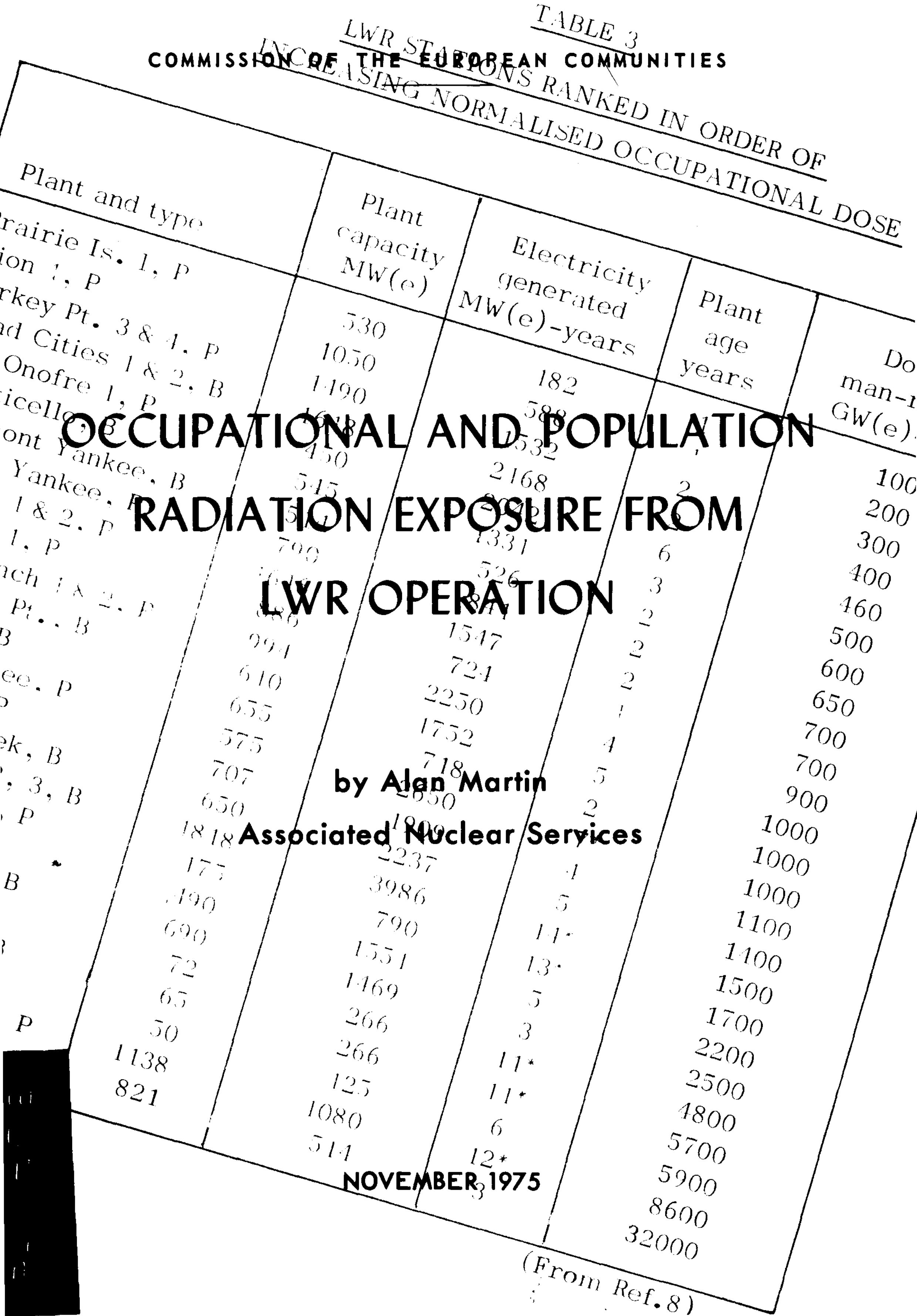

OCCUPATIONAL AND POPULATION RADIATION EXPOSURE FROM LWR OPERATION

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Associated Nuclear Services

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PREFACE

The development of the nuclear power programme has been accompanied by steady improvements in the design and operation of the equipment for processing radioactive effluents prior to discharge, although these improvements have not always been accorded due publicity. Present generation designs lead to very low environmental exposure such that it is often below the limits of direct detection. This evolution is in keeping with the "as low as readily achievable" philosophy, recommended initially by the I. C. R. P. and now widely accepted. It is the purpose of this report to attempt to quantify this evolution, with particular reference to tritium in power stations equipped with light water reactors, as it appears in the open literature.

Collective dose (expressed in man-rem) is one method of quantifying the extent of environmental exposure; such a procedure is not universally accepted; the lack of sufficiently precise criteria may complicate its application and interpretation. Nevertheless, it can on the whole give a reasonable measure of the relative significance of different discharges, and it is widely used in the literature utilised in this study.

Both in terms of length of experience and published information, U. S. reactors form the most convenient field of study in the present context, viz. the quantification of the evolution of environmental exposure and the optimization - as is inherent in the phrase "readily achievable" - of waste management. It is to be hoped that at some later date a comparable study based on European experience and situations will be possible.

Dr. P. RECHT
Director

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1. SUMMARY

This report presents the findings of a study of the relationship between occupational exposure and population exposure resulting from operational practices in light water reactor plants.

A review is made of experience in LWR plants in the US. It is concluded that the level of occupational exposure to date has averaged about 1400 man-rem per GW(e)-year of electricity generated. The main source of occupational exposure is activated corrosion products and between 60% and 80% of exposure occurs during maintenance and refuelling outages. The contribution from the operation and maintenance of waste management systems is of the order of 10% of total exposure. Whilst, in some cases, augmentation of waste systems might marginally increase occupational exposure, in other cases improvements in the system might reduce exposure.

Apart from the global population dose from carbon-14 release, the main contribution to population exposure from early plants arose from noble gas releases from BWR plants. With the introduction of augmented waste systems, these have been greatly reduced and, again excepting C-14, the level of collective population exposure has been reduced far below the collective occupational exposure.

A review is made of the radiological impact of plant water recycle in PWR plants. It is concluded that recycle can result in significant occupational exposure to tritium. In locations where rivers are an important source of drinking water, it seems possible that some small reduction in population exposure could result from recycle.

In optimising the balance between occupational and population exposure, some form of cost-benefit analysis seems to be required. It is pointed out that occupational exposure at the levels experienced in LWR plants represents a direct cost to the utility. Expressed in dollars per man-rem, this cost is one to two orders of magnitude higher than the hypothetical 'health-cost' associated with population exposure.

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2. INTRODUCTION

The operation of nuclear power plants results in the generation of large inventories of radioactivity. These comprise fission products and transuranic elements within the fuel, and smaller amounts of radio-activation products arising from the interaction of neutrons with the coolant and structural materials of the reactor. Apart from relatively small amounts which leak from defective fuel cans, almost all of the fission products and the transuranic elements are retained within the fuel and are eventually transported to a fuel reprocessing plant. That portion of the radioactivity which leaks from fuel appears, along with a proportion of the activation products, in waste streams at the nuclear power plant. The release to the environment of some fraction of the radioactivity in these waste streams cannot be avoided and this leads to the exposure of the general population to ionising radiation. This aspect of nuclear operations has received a great deal of attention in recent years, and the effect has been a tightening up of limits for environmental releases and public exposure.

In general, measures taken to reduce releases of radioactivity to the environment entail more sophisticated waste treatment systems and the accumulation of greater inventories of radioactivity in storage facilities. This is liable to increase both the amount of maintenance required on waste systems and the dose rates encountered. Consequently, any reduction in population exposure could be offset or even out-weighed by an increase in the exposure of plant personnel. In addition to work on the waste treatment system, increased occupational exposure could result from more frequent outages enforced by stringent release limits.

The extent to which occupational exposure has been increased in practice by the augmented waste treatment systems in more recent nuclear power plants is somewhat uncertain. However, it is apparent that the level of occupational dose is a major operational problem at many plants. Thus any measures which might further increase the dose need to be evaluated very carefully.

Another factor is that the view is being expressed in some quarters that occupational dose limits are too high. Many plants would be faced with severe problems if current limits were to be significantly reduced.

In this report, attention has been confined to consideration of light-water reactors (LWR) since these are likely to dominate plant investment for the next decade or two. The data presented refer almost entirely to experience in the USA because only in that country is this type of information freely disseminated. It is the author's opinion that both the public and the nuclear industry would benefit from a similar approach in Europe.

It should be noted that the term 'exposure' is used in this report in the general sense of 'exposure to ionising radiation', rather than in its special dosimetric sense. For brevity, 'dose' is used to mean dose-equivalent.

3. RADIOLOGICAL PROTECTION STANDARDS

The authoritative international body concerned with radiological standards is the International Commission on Radiological Protection (ICRP). The current recommendations are essentially those which were introduced in 1959, though various minor modifications have been made. The recommendations were intended to limit somatic effects in individuals and hereditary effects in the population as a whole. Thus, in addition to permissible doses for individuals, limits were recommended for the average dose to whole populations. For individuals exposed in the course of their occupation, the annual dose limits are 5 rem to the gonads and bone marrow, 30 rem to skin, bone and thyroid, 75 rem to extremities and 15 rem to other organs. The limits for individual members of the public are one-tenth of the occupational exposure values, except in the case of the thyroid, for which the limit is one-twentieth, that is 1.5 rem/y. A provisional limit of 5 rem per generation (i.e. per 30 years) was recommended for the genetically significant dose in a whole population. The apportionment of this limit between various possible sources of exposure is left to national authorities. In all cases mentioned above, the permissible dose is in addition to that resulting from natural background radiation and medical procedures.

These dose limits, which the Commission has repeatedly emphasised are maximum values, were re-affirmed in 1973. The Commission, recognising that any exposure may involve some degree of risk, recommended that unnecessary exposure be avoided and that all doses be kept as low as practicable (Ref. 1). This was later re-worded in ICRP Publication 9 (Ref. 2) to read "..... and that all doses be kept as low as readily achievable, economic and social considerations being taken into account". The interpretation of these terms gave rise to some controversy and the Commission subsequently clarified its intentions in Publication 22 (Ref. 3). In this, the adverb 'readily' is replaced by 'reasonably', and some guidance is given as to how to take account of social and economic considerations by the application of a methodology of differential cost-benefit analysis.

The ICRP recommendations form the basis of national and international regulations such as the Euratom Directives relating to radiological protection standards. In addition, various types of derived standards are in use. These provide interpretation of the basic standards for specific applications and take account of special local considerations.

An important example of such derived standards is the issue by the US Nuclear Regulatory Commission of Docket No. RM-50-2, "The opinion of the Commission on the Matter of Rulemaking Hearing - numerical guides for design objectives and limiting conditions for operation to meet the criterion 'as low as practicable' for radioactive material in light-water-cooled nuclear power reactor effluents". The design objectives, which are specified on a 'per reactor' rather than a 'per site' basis, require limitation of releases of radioactivity so that the annual dose to any individual in unrestricted areas will not exceed 3 mrem to the total body or 10 mrem to any organ in the case of liquid effluents, and 5 mrem to the total body and 15 mrem to the skin due to gaseous effluents. For particulates and radio-iodines the individual organ limit is 15 mrem per year. There is also a requirement to install such further equipment as would be justified by a cost-benefit analysis. The benefit is defined as the reduction in the collective dose out to a distance of 50 miles, valued provisionally at $1000 per man-rem and per man-thyroid rem. The Nuclear Regulatory Commission has replaced the phrase 'as low as practicable' by the version used in ICRP 22, that is 'as low as reasonably achievable'.

Another example of derived standards is the issue by the Central Electricity Generating Board in the UK of 'Radiological Design Criteria' (Ref. 4). These criteria are intended to alleviate possible difficulties which could arise should existing dose limits be reduced in the future. Broadly, the criteria allow for a five-fold reduction in whole body dose limits, and for special consideration of dose from ingestion pathways or long-lived deposited activity.

4. OCCUPATIONAL EXPOSURE AT LWR PLANTS

The level of occupational radiation exposure to plant maintenance staff and contractors' personnel is posing an increasing and, in some cases, severe problem in LWR nuclear power plants. In order to obtain statistics to enable trends to be observed and major sources to be identified, more attention is being given in the US to the recording and reporting of exposure data. These data are reported in limited detail in plant semi-annual operating reports to which some reference has been made (e.g. Ref. 5). Reports have also been published in the US summarising exposure statistics and analysing the results in relation to such factors as type, age and size of plant, as well as attempting to identify the job activities contributing most to exposure (Ref. 6, 7 and 8).

It is clear that the main contribution to in-plant exposure arises during maintenance of primary systems and comes from activated corrosion products, of which Co-60 is usually the most significant.

An area of particular interest to the present study is the extent to which radioactive waste treatment (radwaste) systems contribute to occupational exposure. Data are reported relating to this topic (Ref. 9 and 10).

4.1 Factors influencing occupational exposure

A summary of occupational radiation exposure in LWR plants in the US from 1969 to 1974 is shown in Table 1 (Ref. 8). The increase in the average annual man-rem dose per unit over the period can be partly explained by the increasing size of the later plants. This is illustrated by Table 2, in which the dose is normalised to electrical power generation. After normalising in this way, any trend is masked by the variability of the data. However, this apparent relationship between exposure and power generation is probably due to a combination of factors, some tending to increase, and others to decrease, the dose.

An analysis of the factors which affect the level of occupational exposure in LWR plants was made in Ref. 7, which was prepared under the National Environmental Studies Project (NESP). The major findings, supplemented in some cases from other sources, are reviewed briefly below.

Type of plant: The dependence of the level of occupational exposure on the type of plant is illustrated by the weighted average values in the bottom lines of Tables 1 and 2. The average dose per plant is about 40% higher in PWRs than in BWRs but, when normalised to power generation, the difference between the two types of plants is not significant, the average dose being about 1400 man-rem/GW(e)-year. It should, however, be noted that one PWR plant, Palisades, has been excluded from the analysis because of exceptionally unfavourable experience, see 4.3 below.

Age of plant: The importance of activated corrosion products as a contributor to occupational dose has already been mentioned. These arise as a result of the deposition in the core region of corrosion material which is then activated by the neutron flux. Coolant exchange and transport processes transfer some of the activity to the out-of-core regions of the primary system. In the early years of operation of a plant, the inventory of corrosion products in the system and on core surfaces is increasing and so the rates of production and release of activated material also increase. On this basis the dose rates around the plant due to activated corrosion products would be expected to be a supra-linear, perhaps a quadratic, function of time. By the time the equilibrium refuelling cycle is achieved, the rate of release from the core is constant, neglecting short-term transient behaviour. The buildup of the radiation levels then follows the form $[1-\exp(-\lambda t)]$, where λ is the decay constant of the principal contributor, which is usually cobalt-60 (half-life 5.3 years). Thus there is a period of linear increase followed by a levelling off at 10 to 15 years.

Although this is a somewhat simplified view, which does not take full account of crud bursting, crud flushing and plant cleanup mechanisms, the general form of behaviour is borne out by plant experience. This is illustrated in Fig. 1 in which the buildup of the dose rate in the vicinity of the primary system of a BWR plant is shown (Ref. 11). The increase of occupational exposure with time is not necessarily of the same form, since the maintenance requirements may change. However, the analysis performed in Ref. 7 suggested that, for LWRs as a whole, the occupational dose increased by a factor of two to three per year for the first few years and thereafter at about 12% per year. This seemed to be true both of the early plants and of the larger second-generation plants.

The importance of plant age is also illustrated by Table 3 (Ref. 8) in which twenty-five stations are ranked in order of cumulative occupational dose per MW(e)-year. This shows a clear correlation with plant age and it can be seen that at only one plant aged 5 years or more is the normalised dose less than 1000 man-rem per GW(e)-year.

Plant size: The influence of plant size on the level of occupational dose is partly concealed by the shorter operating times of the larger plants. The analysis in Ref. 7 showed that for BWRs, the annual exposure is considerably higher in the large plants than in the early small plants. When exposure is normalised to power generation the reverse is true. Very roughly, it appears that for a given age of plant, the annual occupational dose is proportional to the square root of plant size.

The difference in size between the first and second generation PWR plants is much smaller than in the case of BWRs. The differences in dose are consequently smaller but are not inconsistent with a similar form of relationship to that mentioned above.

4.2 Main sources of exposure

A detailed analysis was made in the NESP report of data from LWR plants in order to estimate the contributions to exposure from various tasks. Considerable variation was found from plant to plant. Only 47% and 65% respectively of total exposure in BWR and PWR plants was accounted for in detail; most of the remainder was attributed to routine operations and maintenance. An extremely detailed breakdown of exposure at the 1974 refuelling at the H.B. Robinson PWR plant was recently reported (Ref. 12). Combining this with information from the semi-annual operating report from the same plant (Ref. 5), an overall breakdown of exposure similar to that in the NESP report can be obtained. It is worth noting that, at this plant in 1974, 512 man-rem out of the annual total of 680 man-rem dose arose during the refuelling and maintenance outage.

Table 4 summarises the breakdown of exposure by task for BWR and PWR from the NESP report and for the 1974 data from the H.B. Robinson PWR plant as discussed above. It appears that, for both types of plant, about 60% to 80% of total exposure occurs during refuelling and maintenance outages. In PWRs, steam generator work is the main contributor, averaging about 25% of total exposure. Next in importance are pressure vessel head removal and installation work, in-service inspection and waste system operation and maintenance. At BWR plants, no single source of exposure is dominant. General primary system maintenance, waste system operation and maintenance and in-service inspection are the main sources of occupational exposure.

An analysis of the recorded annual total-body exposures for 1974 at the H.B. Robinson PWR plant is shown in Table 5 (derived from Ref. 5). This gives a breakdown of individual doses and indicates the main duties of those individuals receiving the bulk of the exposure.

4.3 Major equipment failures

The very high cumulative exposures at some of the plants listed in Table 3 are atypical in that they are attributable to major equipment failures. Several plants have experienced such failures and the occupational exposures involved in the repair operations have ranged from a few hundred to a few thousand man-rem. The health physics aspects of one of these operations, the repair of defective welds in boiler downcomers at Indian Point 1 PWR, have been reported in detail (Ref. 13). The repair, which utilised about 1500 people, resulted in a total exposure of 3500 man-rem.

At another PWR plant, Palisades, severe problems with steam generators have resulted in very high occupational exposure. The dose statistics from this plant have been excluded from the analyses of Tables 1 and 2 because they distort the overall picture to an unreasonable extent. However, it must be noted that the occasional major failure can involve exposure equivalent to several years of normal plant operation. The later in plant life that the failure occurs, the more severe will be the problem, because of the increasing radiation levels around the reactor systems.

4.4 Internal exposure

Little has been published about occupational exposure resulting from the intake of radioactivity by plant personnel. The main possibilities for this mode of exposure would seem to be;

a) inhalation of activation products during system maintenance with a resulting lung dose,

b) inhalation of radio-iodines during containment access or, in BWRs, turbine-house work, and

c) exposure to tritiated water vapour in PWRs.

The occasional incident involving inhalation of activation products has been reported. For example, at the H.B. Robinson plant in 1974, a worker experienced an intake of Co-58 and Co-60 corresponding to a lung dose commitment of 16 rem (Ref. 5).

Thyroid dose from inhalation of radio-iodines seems to have been very low but the trend towards recirculatory ventilation systems in BWR turbine-houses might result in some increase.

Exposure to tritium does not appear to have been measured routinely. However, estimates have been made of the likely exposure, under various assumptions as to the extent of plant water recycling (Ref. 14 and 15). This aspect is discussed in greater detail in Section 7.

5. OCCUPATIONAL EXPOSURE FROM RADWASTE TREATMENT

The exposure resulting from the operation and maintenance of radwaste systems is of particular interest in the context of this report. The information presented in Section 4.2 suggests that waste management is a significant but not a dominant source of exposure, ranging from about 6% to 13% of total exposure. The data from the H.B. Robinson plant include only the contribution from work during the refuelling outage and it is thought that exposure during routine operation might increase the total contribution from waste management from 6.4% to about 10%.

The extent to which occupational dose is affected by the imposition of more stringent release limits is somewhat uncertain. Apart from any direct effects arising from the operation and maintenance of the augmented waste systems, there is also the possibility of indirectly influencing exposure because of constraints on the operation of the plant as a whole. Examples of both these situations are discussed below.

5.1 Direct exposure

Increased waste treatment involves the processing and accumulation of larger inventories of radioactivity within the plant and could lead to higher occupational doses. However, in practice, this is true only to a very limited extent. For example, arisings of radioactivity in solid wastes, comprising mainly evaporator bottoms and spent resins, are typically of the order of 500 Ci per year in an LWR plant. Discharges of radioactivity, other than tritium, in liquid effluents from first-generation plants have seldom exceeded 20 to 30 Ci per year. Thus, the introduction of augmented waste systems, which have reduced these discharges to about 1 Ci or less per year, can only have increased the accumulated activity by a few per cent. Whilst the volume of waste and the costs of its treatment, drumming and disposal have increased substantially, the associated dose is unlikely to have increased appreciably.

Another example is the increased tritium activity within PWR plants when primary water is recycled. This is considered in greater detail in Section 7, in which it is concluded that significant exposure can arise as the concentration of tritium builds up.

In the case of gaseous wastes, there is a significant increase in the holdup of radioactivity when augmented systems are used. However, with properly designed systems, there is no reason why this should increase occupational exposure.

The possibility must also be considered that more stringent release limits may reduce rather than increase occupational exposure. This could result from improvements in plant design in relation to such factors as leakage collection, plant layout and, more generally, increased reliability.

An appreciable proportion of the dose received from radwaste systems seems to be attributable to under-design or unreliability. For example, drumming stations are amenable to a greater degree of remote handling and automation than has been the practice in most plants. Evaporators, on the other hand, have often caused unnecessary exposure because of unreliability.

5.2 Indirect exposure

In principle, the imposition of more stringent limits for the release of radioactivity in effluents can place additional constraints on the operation of a plant. This can lead to increased outage time and consequently to additional exposure. For example, there is a range of defective fuel incidence which is clearly tolerable, at least until a scheduled outage. Similarly there is a much higher range which is intolerable even for short periods and which necessitates immediate shutdown. Between these lies a grey area in which the decision as to whether or not operation can continue will depend on a number of factors including the actual and the permissible rates of effluent release. A parallel situation exists with coolant leakage, particularly steam generator tube leakage in PWRs, for which the tolerable level may be dictated by release limits.

It has been argued (e.g. Ref. 10) that stricter limits must result in more frequent shutdowns and, consequently, in increased occupational exposure from head removal, fuel inspection and fuel handling, or from tube repairs. However, this seems to be an over-simplified argument, since the magnitudes of fuel defects or coolant leakages which are tolerable depend on, among other factors, the design margins of the radwaste system. In practice, the designs of augmented systems in recently commissioned plants in the US seem to have about the same margin on the current release criteria as did the designs of older plants on the original criteria. It might alternatively be argued that augmentation of waste systems provides greater operational flexibility. This might still lead to increased occupational exposure, since prolonged operation with severe fuel defects or high coolant leakage rates will inevitably worsen the work environment.

Another factor which leads to reduced flexibility of operation is the unduly restrictive interpretation of limits. An example of this is the expression of limits intended as annual averages in terms of instantaneous dose rates or radionuclide concentrations. In some circumstances this can increase not only occupational dose but also public dose since it encourages a policy of deliberate discharge.

6. POPULATION EXPOSURE FROM LWR OPERATION

Apart from releases of radioactivity, exposure of the public due to nuclear power plants can arise from direct gamma radiation from the plant or from radioactive materials in transit from the plant. In this report, attention is confined to the contribution of releases of radioactivity to population exposure, since in the other cases there is, in general, no conflict between the interests of occupational and population exposure.

The doses received by individuals within the population, as a result of releases of radioactivity from nuclear installations, vary widely depending on their location, age, dietary and other habits, and other statistical factors. More generally this can be expressed in the terms that a given release of radioactivity results in a dose distribution in the population. The most highly exposed individuals, at the upper extremity of the dose distribution, form the so-called critical exposure group. The area under the distribution curve represents the total population dose which, under the linear dose-risk assumption, is a measure of the total detriment resulting from the release.

Waste management operations have, in the past, been controlled mainly on the basis of critical group considerations. This provides a convenient and practicable method of setting discharge limits to ensure that no individual exceeds the dose limit. It does not take account of the possibly greater risk represented by the exposure of the population as a whole to relatively low doses. Nor does it permit the overall risk from radioactive releases to be minimised, since steps taken to reduce critical group dose could actually increase the total population dose.

More recently, the collective dose to whole populations has been recognised as a possible criterion against which to judge the significance of releases. As with occupational exposure, it is often useful to normalise population dose to unit operation, such as 1 GW(e)-year of generation.

In using the concept of population dose, the question as to how far the integration of dose should be extended is a controversial matter. In principle, it is possible to estimate the dose commitment from a given release integrated over the whole global population and over the whole period of existence of the radioactivity in the biosphere. A criticism which is often made of this method is that much of the dose arises from the exposure of large populations in regions remote from the point of release to dose rates which may be orders of magnitude below natural background. The alternative is to place finite limits on the integration and these may take the form of a threshold dose rate, a limiting distance from the point of release, or a limit in time after the release. However, in any of these cases, the selection of a limit involves an arbitrary and probably controversial decision. In addition, major inconsistencies arise when multiple or increasing sources of release are considered. For example, if a threshold dose rate is applied, the contribution at a given point from a particular source of release may be below the threshold, but when added to the contribution from another source may exceed the threshold. Strictly, in applying a threshold, the contribution of every nuclide from every contributory source would need to be considered.

6.1 Assessment of population dose

The magnitude and composition of routine releases of radioactivity from nuclear power plants depend on the plant performance, the modes of operation and the methods of treatment. The radiological impact of a given release depends on the mode of release, the site characteristics and the prevailing conditions. Usually, the levels of release of radioactivity from nuclear installations are so low that the dose to an individual cannot be obtained or inferred by environmental monitoring, except in the case of a member of the critical group. Reliance must instead be placed on idealised mathematical models. In the last few years a great deal of effort has been expended on the development of methods of estimating the exposure of local, regional and global populations.

In the US, detailed computer programs have been developed to enable the assessment of radiation doses to individuals and to populations arising from the release of radioactivity from nuclear installations. Both the Directorate of Regulatory Standards of the Atomic Energy Commission (now the Nuclear Regulatory Commission) and the Environmental Protection Agency (EPA) have used detailed models to compare the radiological impact of LWR plants with a range of waste treatment options (Ref. 16 and 17).

For the purpose of this study, estimates of the population dose resulting from releases of radioactivity in liquid and gaseous effluents from LWR plants have been derived from the EPA work. Results are presented for two alternative waste treatment systems for each type of plant. These are a basic system, typical of systems in plants commissioned in the late 1960s, and an augmented system, representing plants commissioned in the mid-1970s. In the former case, the EPA estimates have been adjusted where necessary to correspond more closely to actual release rates as reported in Ref. 18. However it is emphasised that considerable variation occurs from plant to plant.

Three generalised categories of site have been used in the assessments in the US, seacoast, river and lake. The differences between the three types of site from a dose assessment point of view arise from different population distributions, variations in meteorological characteristics and major differences in pathways of exposure to liquid releases. The integration of population dose has, in most cases, been confined to the population within 80 km of the plant. In some cases, particularly of releases to atmosphere, the total dose could have been underestimated.

6.2 Population dose from PWR operation

In the first generation PWR plants gas treatment is usually confined to holdup of primary system off-gases. For the basic system, 15 days holdup was selected. No special features to limit I-131 release to atmosphere were assumed. In the augmented treatment case, the gas holdup was increased to 30 days and the features provided t- release were a re-circulatory containment cleanup system, au building charcoal absorber and the venting of the blowdown to the main condenser.

Typical liquid effluent treatment in early PWR plants consists of evaporation of high- and low-purity wastes, filtration of laundry waste and the untreated discharge of steam generator blowdown. In the augmented system all waste streams are evaporated and demineralised and additional tank capacity enables a high recycle capacility.

A summary of the estimates of population total-body and thyroid doses is shown in Table 6. It will be seen that, in all cases, the radiological impact is greatest for the generalised river site. The major contribution to total-body dose in the basic treatment case is from releases in liquid effluents. Significant contributions to thyroid dose arise from both liquid and atmospheric releases. The population dose from a PWR with augmented treatment systems is very low.

It is noticeable that, in spite of the relatively high rates of tritium release in liquid effluents, the contribution of this nuclide to population dose is negligible. The EPA estimates of population dose from tritium release to rivers correspond to 6×10^{-4} man-rem per curie. It seems likely that the population dose from tritium release to most European rivers could be higher than for typical US rivers. Results reported in the UK suggest a figure of the order of 0.1 man-rem per curie for releases of tritium into the Thames (Ref. 19). This is an extreme case since the Thames supplies a large proportion of the drinking water for the population of Greater London. The Rhine is probably more typical and it is estimated that the population dose commitment is of the order of 10^{-2} man-rem per curie of tritium released.

Releases of tritium to atmosphere have attracted little attention in the literature. However, a cursory review of semi-annual operating reports from PWR plants showed that release to atmosphere varied from a fraction of a curie per year up to about 900 Ci/y. The latter release, for the Oconee plant in 1974 (Ref. 20), compared with a release in liquid effluents of about 350 Ci for the year.

Overall, it seems likely that tritium releases to atmosphere could be 10 to 20% of releases in liquid effluents, and this would have an additional radiological impact. The first-pass dose to the population within 10^3 km of a UK site has been estimated to be 10^{-2} man-rem per curie of tritium released (Ref. 21). Assuming a release to atmosphere of about 100 Ci per GW(e)-year, the contribution to population dose from this mode of release could exceed that from liquid release, but the total would still only be of the order of 1 man-rem per GW(e)-year.

6.3 Population dose from BWR operation

The system assumed for the basic BWR gaseous effluent case is a 30 minute delay of the air-ejector off-gas. No special provision is made for the control of radio-iodine. In the augmented case, allowance is made for 10 day xenon holdup using a recombiner/charcoal bed, and a clean steam supply to the turbine gland seals and to valves of diameter exceeding 2.5 inches.

In the basic liquid system, clean liquors are filtered and demineralised, allowing 90% recycle. Chemical wastes are evaporated prior to discharge, whilst dirty, including laundry, wastes are filtered. In the augmented system, additional tanks are provided so that essentially complete recycling of clean wastes is achieved. The chemical and dirty wastes are evaporated, the former being partially recycled.

A summary of the estimates of population total-body and thyroid doses, normalised to 1 GW(e)-year of operation, is shown in Table 7. This shows that almost all of the dose from BWR operation is due to noble gas release and that substantial reduction is achieved by the introduction of charcoal bed systems. The thyroid dose from radio-iodine release is also reduced significantly by augmentation of the waste system. The total-body dose from liquid releases is quite small even for the basic treatment system.

6.4 Globally dispersed activity

In addition to the so-called first-pass doses discussed above, further contributions can arise from the widespread dispersal of some long-lived environmentally persistent nuclides. The significance of H-3 and Kr-85 in this respect has long been appreciated but only recently has the much greater importance of C-14 been recognised (Ref. 21, 22 and 23). The radiological impact of releases of these nuclides is summarised in Table 8. Estimates of the population dose factors, in man-rem per curie, vary considerably in the literature, being dependent on the assumed modes of release, the realism of dispersion models and dosimetric models, and the assumed future growth of population. The release rates assumed for H-3 and Kr-85 are those for the basic treatment cases considered in 6.2 and 6.3. In the case of tritium, it has again been assumed that 10% of the release is to atmosphere.

Estimates of the total production of C-14 in LWRs have ranged from about 15 to 50 Ci/GW(e)-year and, with the assumption of a release of about one third from the reactor, a figure of 10 Ci/GW(e)-year seems reasonable for the present purpose.

Table 8 illustrates the importance of carbon-14 as a source of population exposure.

7. PLANT WATER RECYCLE IN PWR PLANTS

The recycle of primary water of suitable quality in LWR plants is clearly a sensible and economic mode of operation. Where there is deliberate drainage of systems, the collection of the water in such a way as to maintain its purity and avoid aeration enables its re-use, thus reducing the load on the waste processing system. To this extent, recycle has been practised in all LWR plants.

In more recent years, considerable attention has been given to the design of PWRs and to operational procedures to permit more complete recycling of plant water. The main reason for this is reported to be the control of tritium release (Ref. 24). Thus, as tritium cannot easily be removed from water, it was apparently reasoned that, by recycling plant water, the tritium would accumulate in the plant rather than be discharged. In practice, some loss of plant water is unavoidable and since, with recycle, the specific activity of the water increases over the plant life, it is of interest to know by how much the tritium release is reduced in the long-term. Another factor is that, with a high degree of recycle, the increased specific activity of tritium can result in significant levels of tritium in air in and around the plant, so contributing to the radiation exposure of plant personnel.

In this section the effects of primary water recycle in PWR plants on occupational and population exposures are examined.

7.1 Tritium production in PWR plants

The modes of tritium production in PWR plants include fission and neutron capture reactions on deuterium, boron and lithium. The total production and the expected releases to coolant from the various sources are summarised in Table 9 (Ref. 25). The 1% release of fission product tritium is based on experience with Zircaloy clad fuel. The total tritium release to coolant is 690 Ci/y except in the initial (first-year) cycle, in which there is an additional contribution of 520 Ci from burnable poison rods. Both of these values have been normalised to 1 GW(e).

7.2 The accumulation and release of tritium

During the operating cycle of a PWR, there is an increase in the level of tritium in the coolant, which is taken to mean the contents of the reactor system, the recycle holdup tank and the reactor makeup water storage tank. During refuelling, the coolant is mixed with the contents of the refuelling water storage tanks (RWST) and, to some extent, with the spent fuel pit (SFP) water.

In Ref. 14 it was estimated that complete retention of plant water would cause the tritium level to rise to about 4.5 μCi/ml after 32 years of operation of a 1300 MW(t) PWR. In practice, some loss of water is unavoidable and an analysis has been made of the effects on accumulations and releases of tritium of different assumptions as to the leakage rates. The model assumed and the principal parameters are shown in Fig. 2.

If I_n is the tritium inventory in the coolant at the beginning of the nth cycle, the inventory I_n' at the end of the operating cycle will be

$$I_n' = \frac{r}{(\lambda + \mu)}\left[1 - e^{-(\lambda + \mu)}\right] + I_n e^{-(\lambda + \mu)}$$

it being assumed that the cycle period is one year.

The tritium leakage from the coolant system during this period $L_{i,n}$ is

$$L_{i,n} = \frac{r\mu}{(\lambda + \mu)}\left[1 + \frac{1}{r}\left(I_n - \frac{r}{(\lambda + \mu)}\right)\left(1 - e^{-(\lambda + \mu)}\right)\right]$$

At the end of the operating cycle the inventory I_n' is mixed with J_n', the joint inventory of RWST and SFP, and a fractional leakage m of the total inventory occurs during the refuelling operation. The total tritium release for year n is therefore

$$L_n = L_{i,n} + m(I_n' + J_n')$$

At the end of refuelling, the refuelling water is returned to storage and the initial inventories for year $(n + 1)$ are

$$I_{(n+1)} = (I_n' + J_n')(1 - m) K$$

$$J_{(n+1)} = (I_n' + J_n')(1 - m)(1 - K)$$

$$\text{where } K = \frac{\text{volume of coolant}}{\text{total volume}}$$

Four cases have been evaluated, one representing the idealised case of Ref. 14, that is complete recycle, and three representing a range of more practical situations. In the latter three cases, the fractional loss, m, during refuelling has been assumed to be 0.02. In Ref. 25 the design estimate of non-recycleable leakage was 20 gallons per day. Operational experience suggests that this is rather low and a value of about 48 gallons per day (corresponding to $\mu = 0.137\ y^{-1}$) has been used to illustrate the realistic maximum-recycle case. The other cases evaluated are $\mu = 1$, corresponding to 350 gallons/d, and $\mu = 3$. The latter case would be representative of a situation in which recycle was being deliberately minimised.

7.3 Assessment of occupational dose

No direct measurements of tritium dose to PWR plant personnel have been reported, but attempts have been made to estimate the relationship between the tritium concentration in coolant and in the containment atmosphere. A series of measurements at the H.B. Robinson plant over a five-month period in 1974 (Ref. 15) gave average concentrations of tritium in reactor coolant of 0.222 μCi/ml and, in the containment atmosphere, 3.84 pCi/ml, though the results were rather variable. Accepting this as the only estimate available, a coolant concentration of tritium of 1 μCi/ml will give 17 pCi/ml in the containment atmosphere. Assuming a breathing rate of 1.25 m^3/h, and a dose commitment of 10^{-4} rem per curie intake of tritium, it can be deduced that 1 μCi/ml in coolant results in 2×10^{-3} rem per hour of containment access.

During the refuelling, with the containment ventilated, the relationship between the water and air concentrations may be different. The average temperature, and therefore the average water vapour pressure, in the containment will be lower during refuelling. However this factor must be set against the effects of exposing a large area of water, the surface layer of which is relatively hot. It has, therefore, been assumed that the same factors can be applied to containment access during refuelling, as during periods of operation.

The amount of containment access required is liable to variation but, in Ref. 15, it was estimated that the annual requirements were 750 man-hours during plant operation and 10,000 man-hours during the annual shut-down. These values have been used, in conjunction with the dosimetric data discussed above, to estimate the impact of plant recycle on occupational exposure.

7.4 Assessment of population dose

The impact of plant water recycle on population dose depends on the mode of release and the site location. It has been assumed that all tritium released during the operating phase is in liquid effluents but that during refuelling outages half of the release is to atmosphere, via the containment ventilation system. The resulting population dose has been calculated for a site location on a European river, using a value of 1×10^{-2} man-rem per curie of tritium released either to the river or to atmosphere, as discussed in section 6.2.

7.5 The radiological impact of plant water recycle

The effects of plant water recycle in PWR plants are summarised in Table 10. The results show that, as compared to a low-recycle case, the idealised complete recycle of plant water would increase the average tritium concentration in plant water by an order of magnitude. In the case of a more realistic maximum-recycle case, the factor of increase is five. The occupational dose resulting from recycle, although subject to considerable uncertainty, does appear to be significant.

Under the particular assumptions used in this study, the population dose from tritium release is comparable to the in-plant dose from the nuclide for a plant operated in a low-recycle mode. Increasing the degree of recycle does result in some reduction of population dose but this is more than off-set by the increase in occupational dose. For site locations other than rivers, the population dose from tritium release is lower and so the benefit accrued from increased recycle would be even less. In some cases, such as a coastal site, for which the population dose from the release of tritium in liquid effluents is very low, recycle could actually increase the dose since a higher proportion of the release would be to atmosphere.

8. THE OPTIMISATION OF RADIOLOGICAL PROTECTION

8.1 Occupational exposure

The review in section 4 of this report showed that occupational exposure is posing an increasingly severe problem in the operation and maintenance of LWR plants. Occupational exposure to date, in the US, has averaged about 1400 man-rem per GW(e)-year of electricity generated. This is liable to increase substantially as plants age. Possible developments to alleviate the occupational exposure problem include measures to control corrosion product behaviour, chemical decontamination of reactor systems and the general optimisation of plant layout and design. The first of these methods represents the ideal solution, since reduced activation of corrosion products would reduce both occupational exposure and the burden on waste management systems. In practice, the scope for corrosion product control in normal plant operation by manipulation of coolant chemistry is very limited, particularly in the PWR, because of the use of chemical shim. However, a technique involving the chemical and thermal cycling of the coolant during plant shutdown has been used successfully in Canada (Ref. 26). Reactor system decontamination by the addition of proprietary chemical reagents, basically ammonia citrate/oxalate, has also been performed successfully in the UK (Ref. 27). The disadvantage of chemical cleaning is that the large volumes of chemical wastes may pose a disposal problem.

The imposition of more stringent limits on releases of radioactivity does not appear, necessarily, to increase occupational dose. In many cases, improved system design would be effective in reducing both categories of exposure. In a few cases, most notably that of tritium, it does appear that attempts to reduce releases might significantly increase occupational exposure.

8.2 Population exposure

The review in section 6 showed that there have been two main contributions to population exposure from the operation of LWR plants. One of these, the population dose from gaseous releases from early BWR plants operating on the least favourable sites, has been reduced from about 3000 to less than 100 man-rem per GW(e)-year. The other major contribution is from the release of carbon-14. This is a problem which has only recently become apparent and its absolute significance is still a subject of discussion. It seems unlikely that any measures which might prove necessary for the control of carbon-14 releases would pose significant problems in the control of occupational exposure.

Apart from this question of C-14, it is clear that the level of collective population exposure from plants with augmented waste systems is far, typically two orders of magnitude, below the level of collective occupational exposure.

8.3 Maintaining exposure 'as low as reasonably achievable'

The need to maintain radiation dose 'as low as reasonably achievable' applies to both occupational and population exposure. However, it is only relatively recently that any attempt has been made to apply the principle formally to occupational exposure, (e.g. Ref. 28 to 30). The stage does not seem to have been reached at which the numerical techniques of cost-benefit analysis can be usefully applied. A great deal can be achieved without any obvious cost penalty by placing more emphasis on the occupational dose aspect during the design stages. An important aspect of this is the feedback of information from operating plants to assist in the analysis of access requirements to different parts of the plant, to enable unsatisfactory or unreliable equipment to be identified, and to enable the design of plant to be generally optimised.

8.4 Cost-benefit analysis

Attempts have been made to apply the method of differential cost-benefit analysis to maintaining population dose as low as reasonably achievable (Ref. 16 and 17). Even in these cases, the technique appears to have been used mainly retrospectively in attempts to justify decisions already taken.

Some form of cost-benefit analysis seems to be the only possible approach in optimising the balance between occupational and population exposure. Thus in evaluating the cost of features aimed at reducing one category of exposure, account should be taken of any possible increase in the other category. However, in judging the relative worths of the two types of exposure, three points must be borne in mind:

i) the levels of occupational exposure are real and measurable whereas, in many cases, population exposure is hypothetical and is often based on grossly pessimistic models,

ii) levels of occupational exposure are, for a significant fraction of plant personnel, quite close to the individual dose limits, whilst those to individual members of the population are orders of magnitude below the limits, and

iii) occupational exposure imposes a direct cost on the operation of the plant since, for some tasks, it dictates the level of manning.

Estimates of the monetary equivalent of population exposure, at levels to individuals of a few per cent of natural background or less, have ranged from about \$10 to \$1000 per man-rem. Among the factors which have contributed to the breadth of this range are the use of discounting procedures by some workers but not by others, and the inclusion or not of a contribution from the genetic effects of radiation. In general, the more sophisticated estimates have tended to be at the lower end of the range, and a value exceeding $100 per man-rem seems difficult to justify.

ICRP 22 suggested that doses close to the limit could be discouraged by arbitrarily increasing the monetary equivalent of one man-rem by a factor of ten or so. Applying this factor of ten would yield a value of $1000 per man-rem for occupational exposure. However, the real cost to a utility, of reduced productivity and increased manning could be substantially higher than this. Estimates of $5000 to $10000 per man-rem of occupational exposures have been reported on this basis. Thus, in balancing occupational and population exposure, the application of a weighting factor of between ten and one hundred to the monetary equivalent of occupational exposure can be justified on economic grounds.

9. CONCLUSIONS

The level of occupational dose in LWR plants is posing an increasing problem. A review of plant operating experience in the US indicates that occupational exposure averages about 1400 man-rem per GW(e)-year of electricity generated. About 10% of this exposure arises from the operation and maintenance of waste management systems. The augmentation of waste systems to meet more stringent release limits will, in some cases, marginally increase exposure whilst in other cases improvements in the system could reduce exposure.

Estimates have been made of the population exposure due to releases of radioactivity from LWR plants equipped with basic and augmented waste systems. Apart from the global population dose from carbon-14 release, the main contribution to population exposure from early plant arose from noble gas releases from BWR plants. With the introduction of augmented gas treatment systems, these have been greatly reduced and, again excepting C-14, the level of collective population exposure has been reduced far below the collective occupational exposure. However, it should be noted that the estimates of population dose have been based mainly on US data and apply to typical US sites. For typical European sites the population dose from a given release could be rather higher, reflecting the increased population density and the greater utilisation of river water.

An assessment has been made of the likely occupational and population exposure to tritium and their dependence on the extent to which plant water is recycled. It is concluded that recycle of plant water results in significantly increased occupational exposure whilst only marginally reducing population exposure.

The evaluation of the cost of features aimed at reducing either occupational or population exposure must take account of any possible increase in the other. This will involve some form of cost-benefit analysis which will need to be undertaken on a case-by-case basis, taking account of the design of a particular plant and its site characteristics. It is suggested that the direct monetary cost to a utility of unit occupational exposure is one to two orders of magnitude higher than the hypothetical cost often associated with population exposure.

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11. REFERENCES

1. ICRP. Recommendations of the International Commission on Radiological Protection, Report of Committee II on Permissible Dose for Internal Radiation, Publication 2, 1959.

2. ICRP. Recommendations of the International Commission on Radiological Protection, Publication 9, 1965.

3. ICRP. Implications of Commission recommendations that doses be kept as low as readily achievable, Publication 22, 1973.

4. G.C.Dale. CEGB Nuclear power stations, radiological design criteria, CEGB Report NHS/73/R110, 1974.

5. H.B. Robinson Plant, Unit 2, Semi-annual operating report no.9, July - December, 1974, Carolina Power and Light Co., Docket 50261 - 479.

6. T.D.Murphy. A compilation of occupational radiation exposure from light water cooled nuclear power plants, 1969-1973 WASH-1311, May 1974.

7. National Environmental Studies Project, Compilation and analysis of data on occupational radiation exposure experienced at operating nuclear power plants, prepared for Atomic Industrial Forum Inc. by SAI Services, September 1974.

8. Summary information from NUREG 75/032, presented to Advisory Committee on Reactor Safeguards Environmental Sub-committee meeting, August 7, 1975.

9. C.A.Willis. A study of the occupational radiation exposure due to radwaste treatment systems at nuclear power plants, Docket RM-50-2, rule making hearings, effluents from light water cooled nuclear power reactors, November 1973.

10. M.I.Goldman. Occupational exposure, additional testimony on behalf of the Consolidated Utility Group, Docket RM-50-2, November 1973.

11. D.G.Bridenbaugh et al. BWR maintenance and in-service inspection, IAEA Symposium on experience from operating and fuelling of nuclear power plants, October 1973.

12. B.H.Webster. Occupational exposure data for H.B. Robinson PWR plant, presented to ACRS Environmental Sub-committee meeting August 1975.

13. A.S.Cheifetz. Health physics aspects of a nuclear boiler downcomer repair operation, paper presented at American Nuclear Society Annual Meeting, 1974.

14. J.Locante, D.D. Malinowski. Tritium in pressurised water reactors, paper presented at tritium symposium Las Vegas 1971.

15. B.H.Webster, J.A.Padgett. An evaluation to determine the feasibility of re-use of reactor coolant in a pressurised water nuclear power plant. Paper presented at Health Physics Society meeting, 1974.

16. USAEC. Final environmental statement concerning proposed rule making action: Numerical guides for design objectives and limiting conditions for operation to meet the criterion "as low as practicable" for radioactive material in light-water-cooled nuclear power reactor effluents, volume 1, the statement, volume 2, analytical models and calculations, volume 3, comments on the draft environmental statement and staff responses thereto WASH-1258, July 1973.

17. US Environmental Protection Agency. Environmental analysis of the uranium fuel cycle Part II - Nuclear power reactors, EPA-520/9-73-003-C, November 1973.

18. A.P.Hull. Average effluent releases from U.S. nuclear power reactors, compared with those from fossil-fuelled plants, in terms of currently applicable environmental standards, Report BNL 18823, Brookhaven National Laboratory 1973.

19. N.T.Mitchell. Radioactivity in surface and coastal waters of the British Isles 1971, Report FRL 9, Ministry of Agriculture, Fisheries and Food 1973.

20. Oconee nuclear station, units 1, 2 and 3, semi-annual operating report and non-radiological environmental specifications, June - December 1974, Docket 50269-337.

21. A.Martin H.ApSimon. Population exposure and the interpretation of its significance, Paper IAEA-SM-184/9, IAEA Symposium, Portoroz May 1974.

22. P.J.Magno et al. A consideration of the significance of carbon-14 discharges from the nuclear industry, Paper presented at 13th AEC air-cleaning conference 1974.

23. R.O.Pohl. Nuclear Energy: Health impact of carbon-14, (In draft), Laboratory of Atomic and Solid State Physics, Cornell University 1975.

24. G.Mara. Westinghouse Electric SA, Personal communication, October 1975.

25. FURNAS CENTRAIS ELECTRICAS S.A. Preliminary facility description and safety analysis report, Usina Nuclear de Angra, Unidade No.1.

26. B.Montford. Decontamination by cycling techniques at Douglas Point nuclear generating station, Proc. American Power Conf.1973.

27. D.Smith et al. Operational characteristics of the steam generating heavy water reactor, Paper A/CONF. 49/P/471, September 1971.

28. C.W.Edwards, E.E.Gutwein. An architect-engineers approach to as-low-as-practicable in operations, American Nuclear Society Conference, Albuquerque 1975.

29. G.C.Biro. Shielding design for better plant availability, ANS annual meeting, 1975.

30. L.Clemons. Minimising radiation exposures in PWR plants, ANS annual meeting 1975.

TABLE 1

SUMMARY OF OCCUPATIONAL EXPOSURE IN LWR PLANTS IN US 1969-1974

Calender Year	Average collective occupational dose Man-rem per unit year			
	BWR	PWR	All LWR	Cumulative all LWR
1969	195 (3)	165 (4)	178 (7)	178 (7)
1970	130 (5)	599 (5)	365 (10)	288 (17)
1971	255 (7)	340 (6)	294 (13)	291 (30)
1972	286 (10)	463 (8)	364 (18)	318 (48)
1973	330 (14)	772 (12)	534 (26)	400 (74)
1974	507 (14)	364 (18)	427 (32)	404 (106)
Weighted average	332 (53)	475 (53)	404(106)	-

(From Ref. 8)

Note: Numbers in brackets are the numbers of unit years over which the value is averaged

TABLE 2

OCCUPATIONAL EXPOSURE IN LWR PLANTS, NORMALISED TO ELECTRICAL POWER GENERATION

Calender Year	Average collective occupational dose Man-rem per GW(e)-year			
	BWR	PWR	All LWR	Cumulative all LWR
1969	3300	700	1100	1100
1970	600	2400	1600	1400
1971	1400	1100	1200	1300
1972	800	1400	1100	1200
1973	1000	2100	1600	1400
1974	1800	1000	1300	1400
Weighted average	1400	1400	1400	-

(From Ref. 8)

Note: Palisades PWR plant excluded from PWR data.

TABLE 3

LWR STATIONS RANKED IN ORDER OF
INCREASING NORMALISED OCCUPATIONAL DOSE

Plant and type	Plant capacity MW(e)	Electricity generated MW(e)-years	Plant age years	Dose man-rem per GW(e)-year
Prairie Is. 1, P	530	182	1	100
Zion 1, P	1050	588	1	200
Turkey Pt. 3 & 4, P	1490	1532	2	300
Quad Cities 1 & 2, B	1618	2168	2	400
San Onofre 1, P	450	2042	6	460
Monticello, B	545	1331	3	500
Vermont Yankee, B	514	526	2	600
Maine Yankee, P	790	841	2	650
Surrey 1 & 2, P	1646	1547	2	700
Oconee 1, P	886	724	1	700
Point Beach 1 & 2, P	994	2250	4	900
Nine Ml. Pt., B	610	1752	5	1000
Pilgrim, B	655	718	2	1000
Conn Yankee, P	575	2650	7*	1000
Robinson, P	707	1909	4	1100
Oyster Creek, B	650	2237	5	1400
Dresden 1, 2, 3, B	1818	3986	14*	1500
Yankee Rowe, P	175	790	13*	1700
Ginna, P	490	1551	5	2200
Millstone Pt., B	690	1469	3	2500
Big Rock Pt., B	72	266	11*	4800
Humboldt Bay, B	65	266	11*	5700
Lacrosse, B	50	125	6	5900
Indian Point 1 & 2, P	1138	1080	12*	8600
Palisades, P	821	514	3	32000

* Last 6 years used (From Ref.8)

TABLE 4

BREAKDOWN OF EXPOSURE BY WORK CATEGORY

Work category	Average % of exposure in work category		
	From NESP report (1)		From Refs. 5 & 12
	BWR	PWR	PWR (2)
Liquid waste treatment	5.6	4.1	0.5
Solid waste handling	3.3	2.5	2.9
Gaseous waste systems	2.7	0.4	3.0
Head removal and installation	1.4	6.5	17
Fuel handling	5.5	3.6	3.0
Fuel pool	0.5	0.3	0.2
In-service inspection	4.9	5.6	2.0
Instrumentation	3.0	1.3	1.1
Control rod drive work	3.2	low	0.5
Recirc. pumps, cleanup system	7.8	-	-
Turbine and aux. equipment	2.7	-	-
Condensate demineraliser	1.2	-	-
Steam generator work	-	27	23
Reactor coolant pumps	-	2.8	5.9
Main coolant loops	-	5.1	3.2
Charging pumps	-	1.4	0.2
Valves	5.2	4.1	0.8
Health physics and cleanup (3)	-	-	9.3
Routine operations and maintenance	33	19	25
Miscellaneous (4)	20	16.3	2.4

Notes:
1. Figures from NESP report are averages of up to 14 plant years
2. These figures are for one year at H.B. Robinson PWR plant
3. This item not included in NESP report. Probably comes partly under specific categories and partly in routine work
4. Includes unaccounted-for exposure

TABLE 5

RECORDED ANNUAL TOTAL-BODY EXPOSURES FOR 1974 AT H.B. ROBINSON PWR PLANT

(Derived from semi-annual operating report No. 9 July-Dec. 1974 Ref. 5)

1. Radiation exposure by increments

Annual dose range, mrem	No. of individuals in each range
No measurable exposure	586
Measurable exposure below 0.1	354
0.1 - 0.25	82
0.25 - 0.5	57
0.5 - 0.75	49
0.75 - 1.0	46
1.0 - 2.0	151
2.0 - 3.0	89
3.0 - 4.0	18
4.0 - 5.0	5
> 5.0	3

2. Numbers of individuals exceeding 0.5 rem in year, by duty function

Duty function	No. of individuals > 0.5 rem
Routine plant surveillance	92
Routine plant maintenance	82
Special maintenance: a) Steam generator	83
b) Other	25
Routine refuelling operations	12
Special refuelling operations	0
Decontamination and cleaning	38
Health physics	28

3. Numbers of individuals exceeding 2.5 rem in year, by duty function

Duty function	No. of individuals > 2.5 rem
Refuelling maintenance and routine duties	29
Steam generator maintenance	6
Fuelling operations and inspection	4
Health physics coverage	8
Janitorial services	9
Waste drumming and shipping	1

4. Internal exposure

One person experienced an intake of Co-58 and Co-60 corresponding to a dose commitment to lung of 16 rem.

5. Estimated total man-rem exposure

The total exposure in 1974 is estimated from the data in item 1 to be 680 man-rem, of which 512 man-rem arose during the refuelling outage.

	Release Ci per GW(e)-y	Population dose, man-rem/GW(e)-year					
		Total-body			Thyroid, from I-131		
		Seacoast	River	Lake	Seacoast	River	Lake
<u>PWR - basic</u>							
Gaseous							
Noble gas	3 10^4	1	10	0.8	-	-	-
I-131	1.0	-	-	-	30	200	20
Liquid							
H-3	750	0.005	0.5	0.08	-	-	-
Non-tritium *	50	6	110	3	-	-	-
I-131	16	-	-	-	60	400	80
Total		7	120	4	90	600	100
<u>PWR - augmented</u>							
Gaseous							
Noble gas	8 10^3	0.25	2.4	0.2	-	-	-
I-131	0.11	-	-	-	4	22	2.3
Liquid							
H-3	500	0.003	0.3	0.05	-	-	-
Non-tritium *	0.40	0.0013	0.013	0.0012	-	-	-
I-131	0.07	-	-	-	0.26	1.6	0.4
Total		0.25	2.7	0.25	4.3	24	2.7

*Includes radioiodines (Derived from Ref. 17 & 18)

<u>TABLE 6</u>

<u>POPULATION TOTAL-BODY AND THYROID DOSES PER GW(e)-YEAR FROM BASIC AND AUGMENTED PWR ON TYPICAL US SITES</u>

	Release Ci per GW(e)-y	Population dose, man-rem/GW(e)-year					
		Total-body			Thyroid, from I-131		
		Seacoast	River	Lake	Seacoast	River	Lake
BWR - basic							
Gaseous							
Noble gas	3 10^6	640	3000	330	-	-	-
I-131	2	-	-	-	70	400	40
Liquid							
H-3	120	0.001	0.08	0.013	-	-	-
Other *	80	3	25	0.8	-	-	-
I-131	4	-	-	-	14	84	20
Total		640	3000	330	84	480	60
BWR - augmented							
Gaseous							
Noble gas	10^5	12	80	8	-	-	-
I-131	0.044	-	-	-	0.03	6	0.6
Liquid							
H-3	80	0.0006	0.05	0.008	-	-	-
Other *	0.22	0.01	0.03	0.003	-	-	-
I-131	0.16	-	-	-	0.6	3.8	0.88
Total		12	80	8	0.63	9.8	1.5

* Includes radioiodines

(Derived from Ref. 17 & 18)

TABLE 7

POPULATION TOTAL-BODY AND THYROID DOSES PER GW(e)-YEAR FROM BASIC AND AUGMENTED BWR ON TYPICAL US SITES

Quantity	Units	H-3	C-14	Kr-85
Half-life	years	12.3	5730	10.6
Population dose factor	man-rem/Ci	$5 \; 10^{-4}$	300	10^{-4}
BWR				
Typical release rate	Ci/GW(e)-year	120	~ 10	500
Population dose commitment	man-rem/GW(e)-year	$6 \; 10^{-2}$	3000	$5 \; 10^{-2}$
PWR				
Typical release rate	Ci/GW(e)-year	750	~ 10	1000
Population dose commitment	man-rem/GW(e)-year	0.4	3000	0.1

Note These estimates do not include the first-pass dose (Ref. 21)

TABLE 8

POPULATION DOSE FROM THE GLOBAL DISPERSION OF RELEASES OF H-3, C-14 AND Kr-85 FROM LWR PLANTS

TABLE 9

TRITIUM PRODUCTION AND RELEASE TO COOLANT IN PWR

Tritium source	Total production Ci per GW(e)-year	Expected release to coolant Ci per GW(e)-year
Fission	11500	115 (1)
Burnable poison rods (initial cycle only)	660	520
Soluble boron	550	550
Lithium-6	15	15
Lithium-7	10	10
Deuterium	2	2
Total: Initial cycle	12700	1210
Total: Equilibrium cycle	12100	690

(Derived from Ref. 25)

Note: 1. Assuming zircaloy clad fuel

Case	Average concentration of H-3 in water μCi/ml		Average release rate Ci/GW(e)-year		Radiological impact man-rem/GW(e)-year	
	Operation	Refuelling	Operation	Refuelling	Occupational	Population
Low recycle case $\mu = 3.0$, $m = 0.02$	0.38	0.29	610	19	6	6
Typical recycle case $\mu = 1.0$, $m = 0.02$	0.92	0.69	460	44	15	5
Maximum recycle case $\mu = 0.137$, $m = 0.02$	2.0	1.5	140	97	30	2
Idealised recycle case $\mu = 0$, $m = 0$	3.0	2.5	-	-	50	-

Notes: 1. Averages are over a 30-year reactor life.
2. Population dose estimates are for a location on the Rhine and are based on 10^{-2} man-rem per Ci of tritium, both for river and atmospheric releases (see section 6.2).

TABLE 10

SUMMARY OF THE EFFECT OF DIFFERENT DEGREES OF WATER RECYCLE IN PWR PLANTS

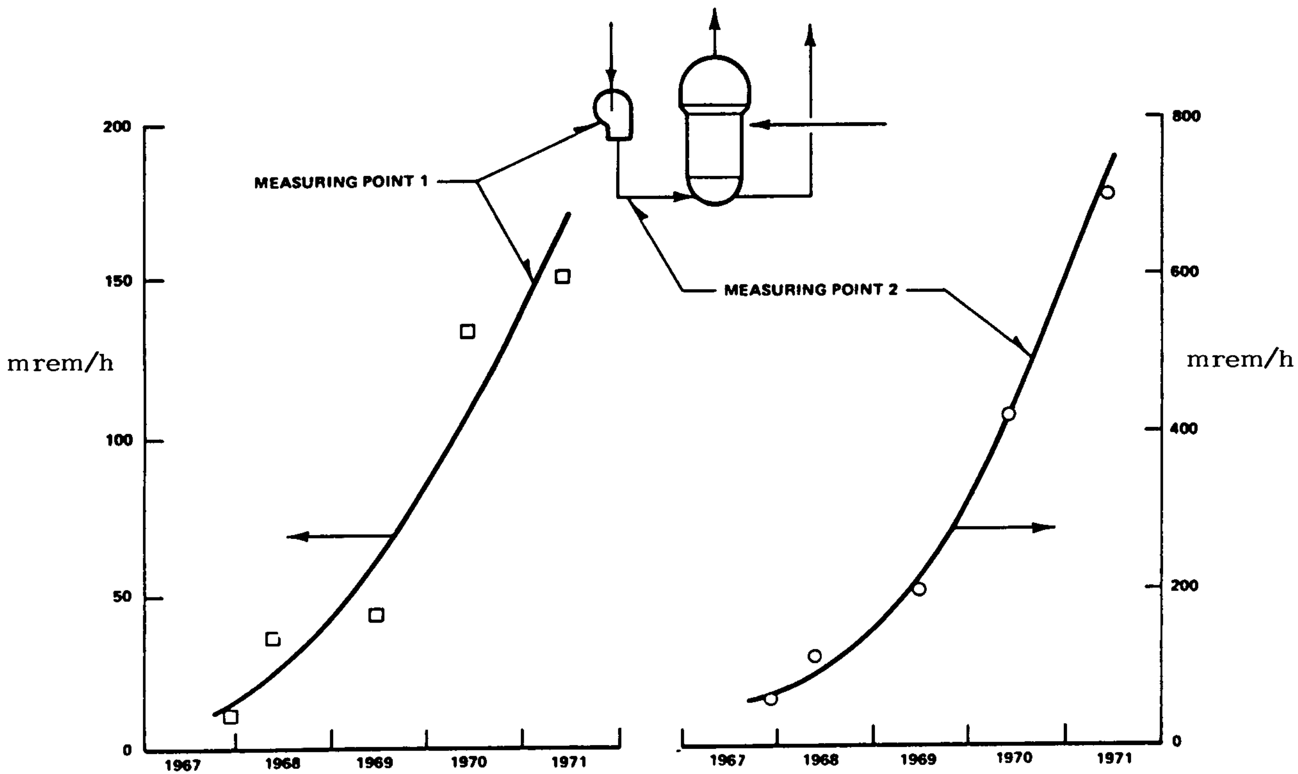

FIGURE 1 BUILD UP OF DOSE RATE IN VICINITY OF PRIMARY SYSTEM OF THE KRB PLANT (Ref. 1·1)

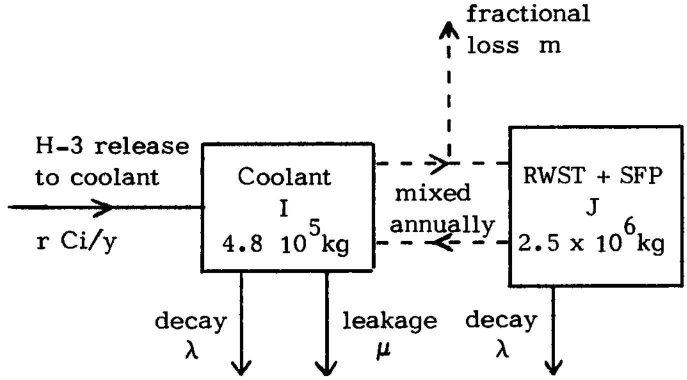

FIGURE 2 SIMPLIFIED MODEL TO ILLUSTRATE EFFECTS OF PLANT WATER RECYCLE IN PWR