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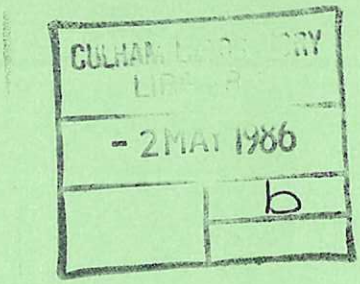
EFFECTS OF LASER IRRADIATION ON β-DECAY

A. M. Lane
Theoretical Physics Division
Building 424.4
Atomic Energy Research Establishment
Harwell, Didcot, Oxon.

Summary

A critical examination is made of proposals that the rate of β-decay can be increased by irradiating the source with an intense laser. The conclusion is that, for optical or UV frequencies and conceivable power levels, the effect is not significant. For hard X-ray frequencies (photon energies 10-100 keV) which strongly excite a suitable nuclear excited state, a large effect is possible, so the proposals should be re-examined if such lasers become available.



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1. INTRODUCTION

In the years 1975-82, H.R. Reiss took out U.S. patents on the proposal to speed-up β decay by exposing the decaying nuclei to an intense laser. Beginning in 1981, this proposal was critically analysed in the accademic literature. Twenty papers have been written by Reiss and by his main critics, W. Becker and M.O. Scully (see References). They have disputed violently about three separate aspects, both sides claiming to show that large increases in decay are possible while claiming that the other side's arguments were wrong. The debate has now ended. Both sides have been wrong but have been reluctant to admit it. The essential result is that, for laser powers up to 10^{18} W/cm^2, no one believes that there are dramatic effects (increases of factors $\geq$ 10 in decay rate). Reiss continues to claim the possibility of "substantial" effects at this power level, but apparently he is thinking of factors of the order of 2, not factors of up to 10^6 that were bandied about at one time.

In the present Report, Section 2 gives the results of reading and analysing the twenty relevant papers, many of which are in conflict with others in the list. The essential physics is described, and the final verdicts on the contentious points are given.

Section 4 summarises how the results would be modified if ever hard X-ray lasers became available. Section 3 refers to the solitary experimental result which purportedly shows a change in β-decay rate due to laser irradiation.

2. BASIC PHYSICS

A laser can affect β decay in three ways:

i) it increases the kinetic energy (and the associated phase-space) of β-particles emerging along the field direction

ii) it can virtually excite nuclear states and thereby convert a forbidden β decay into an allowed one

iii) it can convert a forbidden β decay into an allowed one by changing the angular momentum of the β particle.

Phase-space Effect The intuitive classical idea of the β particle gaining energy from the electric field within the laser period suggests that the condition for appreciable effect (i) is:

$$\frac{eFv}{\omega} \gtrsim \frac{1}{2} mv^2$$

(e, m, v = charge, mass, velocity of β; F = electric field). Since $v \sim c$ for β particles this becomes:

$$\frac{eF}{\omega mc} \gtrsim 1$$

For photons of $\hbar\omega \sim$ leV this means $F \gtrsim 4 \times 10^{10}$ V/cm equivalent to intensity 10^{16} W/cm^2.

This intuitive idea is borne out by detailed calculation. Unfortunately it applies only to β particles emerging in the direction of the electric field force. For those emerging in the opposite direction, the effect is opposite. Upon averaging over directions, there is no net effect on the decay rate.

Becker and Scully originally proposed this affect (Ref. 3) and claimed large increases in decay rate for low-energy decays (eg ^{3}H, ^{18}F) at laser power 10^{18} W/cm^2. They then became aware of earlier Russian work to the contrary, and Reiss (Ref. 4) also disputed their claim. After initially (Ref. 6) rejecting the argument of Ref. 4, they then found their own proof of zero net effect, at least for neutral β-emitters (Ref. 9). Finally, in footnote 14 to Ref. 14, they finally formally withdrew their claim. Meanwhile Akhmedov (Ref. 11) had calculated the effect, and found it to be negligible unless the field is of the order of the critical value for pair production ($F \sim \frac{m^2c^3}{e\hbar} \sim 1.3\times10^{16}$ V/cm) times $(v/c)^3$.

Virtual Nuclear Excitation Effects The probability amplitude for nuclear excitation is of the order $\langle p|e\underline{F}.\underline{r}|p'\rangle/(E_p-E_{p'})$ where p, p' are nuclear particle orbits. To offset one degree of forbiddenness in the β decay, this must be $\gtrsim 0.01$. Taking $\langle p|r|p'\rangle$ as 5×10^{-13}cm, $(E_p-E_{p'})$ as 5 MeV, this gives $F \gtrsim 10^{17}$ V/cm. This estimate means that F must be of the order of the electric field between two protons in the nucleus. There is a rival estimate, based on the alternative form $e\underline{A}.\underline{p}$ of the e/m interaction. Applie naively, this appears to give the more accessible condition, $eF\langle p|r|p'\rangle \gtrsim \hbar\omega$, implying $F \gtrsim 10^{10}$ eV/cm. The question of consistency between the two forms, (which should be equivalent) has been the subject of strong disagreement. The subject was raised in Ref. 14, and argued in Refs. 15-22. Ref. 21 shows how the $e\underline{A}.\underline{p}$ form, when used properly, leads to the condition found for the first form, $eF\langle p|r|p'\rangle \gtrsim 0.01\ (E_p-E_{p'})$. This agrees with intuition since this is the condition appropriate to a static electric field. Since ω << nuclear excitation frequencies, one expects the laser field to be effectively static.

Effect from Angular Momentum Transfer to the β particle This effect appears in the formulae of Ref. 7 where it modifies the form of the nuclear operator coming from the β particle wave-function. For eF> ωmc, the usual $(\frac{mvr}{\hbar})^L$ is replaced by $(\frac{mc\omega}{eF})^{\frac{1}{2}}$ $(\frac{eFr}{\hbar\omega})^L$ where L is forbiddenness. For L=1, the ratio is $(\frac{eF}{\omega mc})^{\frac{1}{2}} \frac{c}{v}$ which is > 1 for eF>ωmc, i.e. for F > 4×10^{10} V/cm (assuming $\hbar\omega \sim$ 1eV). Unfortunately the correction of an error (Refs. 10, 15) modifies the new operator to $(\frac{mc\omega}{eF})^{\frac{1}{2}}$ $(\sqrt{2}\ \frac{mcr}{\hbar})^L$ so the ratio to the usual operator is $(\frac{mc\omega}{eF})^{\frac{1}{2}}$ $(\frac{\sqrt{2}c}{v})^L$. The first factor is < 1 by assumption, so this ratio will be > 1 only for small v/c. Thus the β decay rate is not likely to increase by a dramatic factor, say 10 or more, and may well decrease. Even this effect has been disputed (Ref. 13), then re-asserted (Ref. 18).

3. DATA

Despite the theoretical consensus that there is no significant effect of a laser on β decay at practical intensity levels, it would be nice to have data showing a null-effect. The only data available is revealed in a recent paper (Ref. 23). It is claimed that there is an observed effect for ^{137}Cs, but this is subjective. The laser intensity is not cited. If the effect is real, it would cause great consternation in the theoretical community.

4. X-RAY LASER

This is discussed in Ref. 12. The main effect is assumed to be via nuclear excitation which will be real or virtual depending on whether the laser is tuned to the excitation energy or not. When the laser is tuned to an excited nuclear state whose β decay is less forbidden than that of the ground state, there is a large effect if the excited state is strongly excited. This happens when the Rabi time is less than the spontaneous

decay time. For El excitation this means

$$eF \langle r \rangle \gtrsim e^2 \langle r^2 \rangle \left(\frac{\omega}{c}\right)^3 .$$

For $\langle r \rangle \sim 5 \times 10^{-13}$cm, $\hbar\omega \sim 0.1$ MeV:

$$F \gtrsim 10^{10} \text{ V/cm} .$$

The removal of one degree of forbiddenness means that the decay is speeded-up by a factor ~ 100.

Note that the critical field strength is much less than the value required in the effects listed above ($\sim 10^{16}$ V/cm). This is due essentially to the laser being tuned to the nuclear excitation, thereby maximising the action of the laser.

If a laser is not tuned on a nuclear excited state, its frequency is irrelevant so there is no advantage in using an X-ray laser in place of an optical laser (of the same intensity).

An X-ray laser has an advantage only if it is tuned on a low lying nuclear state, in which case β-decay enhancement can occur. Typically one requires an X-ray laser with the features:

photon energy $\hbar\omega \sim$ 10 KeV to 100 keV:
intensity $\sim 10^{14}$ W/cm^2
resolution $\frac{\Delta\omega}{\omega} \sim 10^{-12}$

Although the required intensity level has been achieved for optical lasers, only modest intensities have been achieved for X-ray lasers with $\hbar\omega <$ 100eV. No laser has yet been made with $\hbar\omega >$ 100 eV. Until lasers with $\hbar\omega \sim$ 10 keV are made, there is no prospect of β-decay enhancement with lasers.

References

1. 1969 Canuto (PR187, 2141) finds decay of neutron is unaffected by magnetic fields up to 10^{12}G. (Highest lab field ~ 10^{6}G).

2. 1977-1982 Reiss takes out 5 patents on the proposal to speed-up β decay by the use of lasers.

3. 1981 Becker et al. (PRL47, 1262) consider phase-space effect due to laser. For 10^{18} W/cm^2, they find that $^{3}H(\beta^-)$ decay is speeded up by $2x10^{4}$, while $^{18}F(\beta^+)$ decay is speeded up by ~ 20.

4. 1982 Reiss (PR48, 652) claims 3. is gross overestimate.

5. 1982 Gersten (PR48, 651) claims 3. is invalidated by neglect of screening.

6. 1982 Becker (PR48, 653) refute 4. and 5.

7. 1983 Reiss (PRC27, 1199) and

8. 1983 Reiss (PRC27, 1229).
This work claims to derive large effects from angular momentum transfer from photons to nucleons and electrons. This converts forbidden into allowed β decays.

9. 1983 Becker (PL94A, 131) argue that the phase space effect of the laser cancel to zero for intensities < 10^{29}W/cm^2. This agrees with quoted Russian work (1964-1979).

10. 1983 Reiss (PRC28, 1402) finds that the main result in 7,8 needs correction. A modest result survives.

11. 1983 Akhmedov (JETP58, 883) disputes 3. and says phase-space effects are negligible (already accepted by 9.).

12. 1983 Goldhaber et al (PL131B, 16) conclude that an X-ray laser could produce significant effects on β decay by changing the initial or final nuclear state.

13. 1984 Becker et al (PL101A, 58) give an argument for negligible angular momentum transfer effects.

14. 1984 Becker et al (PRC29, 1124) give yet more arguments for negligible effects, disputing 7, 8 and claiming that nuclear excitation is not actually included in 7, 8.

15. 1984 Reiss (PRC29, 1132) says that 14. is full of fallacies, and that 7,8 (modified by 10) are essentially correct. This paper remarks that field-induced affects are small for allowed transitions (See IV).

16. 1984 Reiss (PRC52, 1061) discuss effect of choice of gauge, and
) claim that 7,8 are correct, while 14 is
17. 1984 Reiss (PRC29, 1925) wrong.

18. 1984 Reiss (PL103A, 312) disputes claims of 13.

19. 1984 Becker (PL52, 2094) disputes 16. and reiterates that "there is no way of enhancing decay by low-frequency radiation".

20. 1984 Reiss (PRL52, 2095) disputes 19.

21. 1985 Cahn and Jackson (PRL54, 1329) dispute 17. and say flatly that it is wrong.

22. 1985 Reiss (PRL54, 1330) disputes 21. and says that 17. is correct.

23. 1985 Reiss (PRC31, 2238) discusses case where a forbidden β decay leads to a forbidden γ decay (isomeric state) and both decays are affected by a laser. For the first time, data is mentioned. (Further details in papers submitted to PRL in Dec. 1984. Non-appearance suggests referee problems).

Table I

Average Annual Radwaste during normal operation

	Mass tonnes	Disposal Classification
STARFIRE		
PCA	71	Exceeds 10CFR61 limits
Zr_5Pb_3	55	C
$LiAlO_2$	100	A
Graphite	27	A
	253	
Low Activation STARFIRE		
Al-6063	3	A
SiC + Li_2O	141	A
Graphite	28	A
	172	

(d) Low Activation STARFIRE

In the low activation STARFIRE design an attempt is made to assess the benefits of low activation materials[18].

Specifically, the limiter, first wall, blanket and shield, and toroidal field magnets are modified to incorporate aluminium alloys and ceramics; this choice of materials is made for illustrative purposes only, and is not necessarily suggested as suitable for the construction of an actual reactor. The radioactivity, after-heat and biological hazard potential for this case are also shown in Figs. 6, 7 and 8. Note that an impurity level of 1 appm iron is assumed and included in all calculations for the low activation design. The low activation design shows a factor three reduction in radioactivity at shutdown. About one day after shutdown it falls six orders of magnitude and is primarily dominated by the iron impurity. This activity level continues for about one year and then drops by another order of magnitude to the level controlled by ^{26}Al at times more than ten years after shutdown. Similarly, the afterheat also shows a dramatic fall (six orders of magnitude) within a few days of shutdown. The biological hazard potential shows an analogous advantage. From Table I we note that the average annual mass of material removed from the reactor is about 170 tonnes; the associated activity is also reduced, as shown by the less stringent disposal requirements. Thus the potential benefits of low activation designs are clear. Whether low activation, or even high activation materials, would also have the appropriate thermal and mechanical properties has yet to be demonstrated. A review of the reasons for low activation fusion has recently been given by Hopkins and Cheng[19].

3. TRITIUM

The tritium inventories envisaged for fusion reactors will almost certainly lead to the release of some radioactivity to the environment. Tritium emits beta particles of very low penetrative power and hence the hazard to man from tritium outside the human body is negligible. In the environment, however, tritium is most commonly found in the form of tritiated water. In this form it may enter the body by ingestion, inhalation and skin absorption. Since tritium is an isotope of hydrogen, it is expected to readily permeate many materials. Thus there is a tritium "containment" problem, the magnitude and environmental impact of which we assess below.

3.1 Radiological Effects and Recommended Standards of Protection

The most likely way which tritium enters the body is through the intake of tritiated water, which rapidly mixes with body fluids; thus the whole body will be uniformly irradiated. Tritiated water leaves the body by excretion and evaporation at a rate determined by the biological half life, which is about 10 days[20]. The derived air concentrations (DAC) as quoted by the International Commission on Radiological Protection, show tritium to be far more readily absorbed in the oxide than in the elemental form. For T_2(and HT) the DAC is $2{\cdot}5 \times 10^4$ times larger than for HTO or T_2O.[13] We note that, unlike some fission products from uranium or plutonium fission, there is no evidence that tritium is concentrated in food chains[3, 21].

In order to assess the necessary degree of tritium containment in a fusion reactor, we consider the radiological standards as recommended in ICRP 26[22]; the application of these recommendations to the U.K. has been discussed by the National Radiological Protection Board[23]. The dose equivalent limits given by ICRP do not take account of natural background radiation or medical procedures. On the basis of risk considerations, the annual dose equivalent limit for individual members of the public is 1 mSv (0·1 rem), provided that the dose is received as a lifetime exposure of the whole body. Previously, the ICRP recommended the higher dose equivalent limit of 5 mSv (0·5 rem) per year as applied to critical groups. However, the Commission states that the latter limit has been adequate and therefore continues to recommend its use. The Commission also states that this annual dose equivalent limit when applied to individual members of the public, is likely to result in an average dose equivalent to the population of less than 0·5 mSv (50 m rem), provided that the practices exposing the public are few and cause little exposure outside the critical group.

If a large population is exposed to whole body irradiation an additional risk can be envisaged, namely genetic injury and its effect on future

generations. In fact, there is no evidence of radiation induced hereditary effects in man at any dose level[24]. However there is no reason to suppose that man is immune from genetic damage due to radiation. In its previous recommendations, the ICRP suggested a genetic dose limit of 5 rem in 30 years, on average 1•7 mSv (170 m rem) per year. In ICRP 26 the Commission does not propose dose limits for populations because these limits could be regarded as suggesting the acceptability of a higher population exposure than is either desirable or necessary. Also, hereditary effects are now considered to be less restricting than somatic effects. The NRPB considers that the maintenance of the annual dose equivalent below 0•5 mSv when averaged over the whole UK population is a reasonable objective; this objective is related to the population from all sources of radiation. The contribution to the annual average dose equivalent resulting from all UK waste management practices is not likely in the Board's view to exceed one-tenth of this dose equivalent, that is, 0•05 mSv (5 m rem) per year.

Recent designs of both actual and conceptual tokamak facilities take 0.05 m Sv per year to be appropriate. Thus, it is stated that JET[5] will not be operated with average doses at the boundary fence in excess of 0•05 mSv per year. Similarly, the same limit is taken into account for the INTOR design studies[6]. In estimating the degree of tritium containment for a fusion reactor (Section 3.3), this is the limit which has generally been adopted.

3.2 Tritium Inventory in a Fusion Reactor

The size of the tritium inventory in conceptual tokamak reactor studies depends on the design details and assumptions, and can vary by a factor of a few. Although the tritium breeding material will be made from some compound of lithium, the ultimate form is as yet unclear. The limited data currently available on the production and extraction processes leads to uncertainties in the steady inventory of the breeding blanket. For purposes of illustration we consider the STARFIRE reference reactor. The

amounts of tritium located in the various systems are given in Table II below.

TABLE II. Tritium inventory in STARFIRE (1.2 GW(e))

System	Inventory (kg)
Fuel Process system	0.154
Vacuum pumps, injectors, etc.	0.117
Blanket	10.0
Blanket process system	0.281
Full storage system	1.125
Coolant, storage pool, atmospheric clean up system	~0.3

Thus the total inventory is of order 12.0 kg, which corresponds to about 100 MCi. The largest part of the tritium, by far, is in the $LiAlO_2$ breeding material. There will be a daily consumption of about 0.5kg, a production and extraction rate from the blanket at least equal to this, and a daily throughput of order 6 kg, say.

All the above systems will involve some release of tritium during routine operation. The tritium release pathways[4] are shown in Fig. 11.

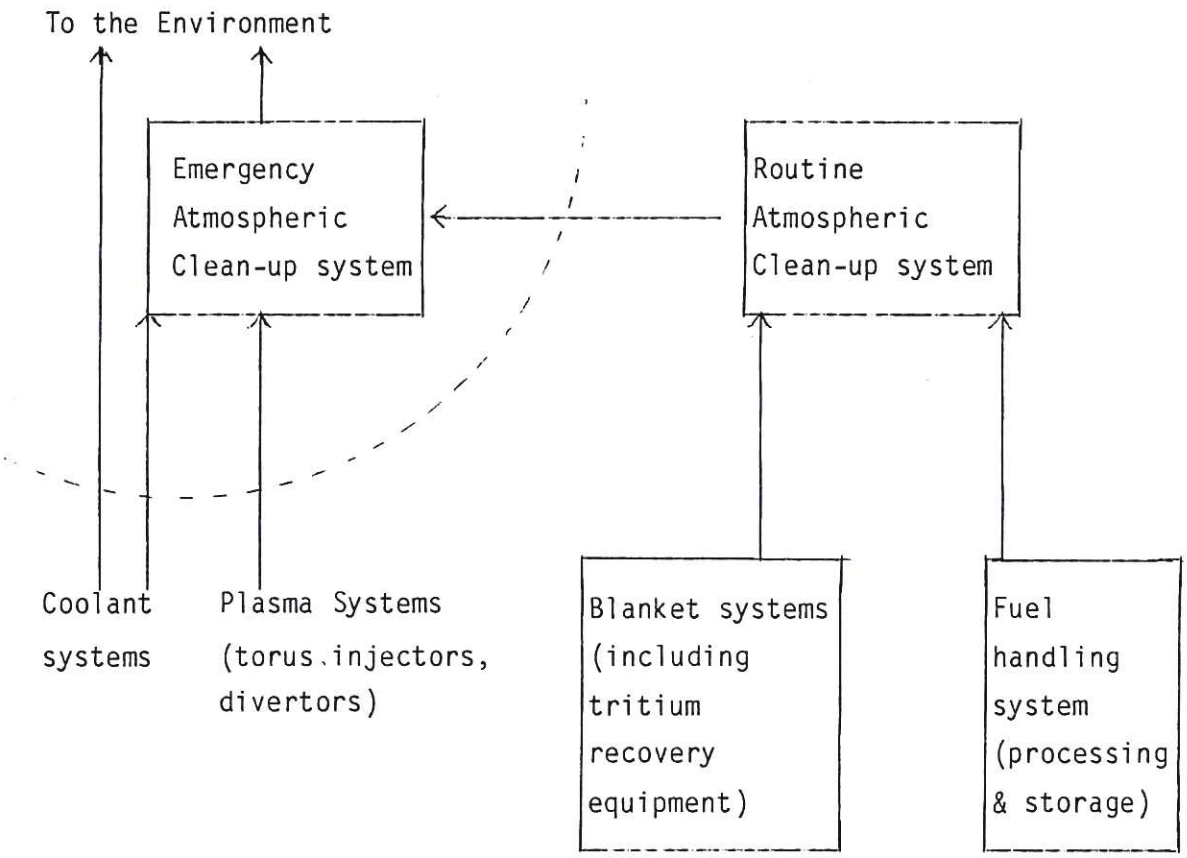

Fig.11 Tritium release pathways (see reference 4).

All fuel processing and storage equipment is likely to be located in a special tritium facility. Components with potentially high leak rates or high-tritium-hazard will be housed within secondary containment, with essentially continuous processing (detritiation) of the containment atmosphere. This is called the "routine atmospheric cleanup system".

Under normal operating conditions tritium may escape by (1) discharge from the clean-up systems, (2) leakage of containment coolant, and (3) routine discharge of ventilation atmosphere. Questions arise as to the degree of tritium containment necessary and whether it can be achieved in practice.

3.3 Required Limits on Tritium Containment

The tritium in a reactor will be present principally as the element or as lithium tritide within the lithium blanket; however, other forms such as T_2O may well occur. As we remarked earlier the oxides show a far greater accessibility to the human body than do the gases HT or T_2. Thus to accurately assess the exposure hazard due to a tritium release, it is important to know the ratio of element to oxide in the release, and the subsequent oxidation rate prior to assimilation by individual members of the population or any critical group. Unfortunately our present knowledge of oxidation, both in the laboratory and the environment, is very sparse. Under laboratory conditions there is evidence[25] that the oxidation rate is very slow, but can be enhanced by the presence of hot metal surfaces, radiation fluence, elastomers etc.[26] With respect to the environment the position is difficult to interpret. In 1974, an accidental tritium gas release of nearly 0.5 MCi in the elemental form occurred at the Savannah River Plant. The atmospheric oxidation rate was determined to be under 1 percent per day, and measured concentrations were well under calculated levels [21]. On the other hand, accelerated conversion mechanisms in the environment have been observed and reported both in France[26] and at Savannah River[27]. These are not understood and their consequences unknown. Apparently the key ingredients are certain green plants and bacterially active soils.

In order to assess the degree of control over tritium required, it is necessary to make the conservative assumption that all the escaping tritium is released as the oxide. The extent to which this assumption over-estimates the hazard is unknown. Turning to the 'local' hazard, we now briefly review the bases of three typical calculations:

(a) Hafele et al.[3] assumes that the predominant release pathway will be in the water discharged from the condensers. Assuming that an individual takes all his or her drinking water from the discharge, Hafele et al. calculate the upper limit to the activity due to the HTO which is compatible with a dose rate of 0.05 mSv per year. They

find this to be or order 10^5 Ci/yr. This is equivalent to controlling the leak rate to about one part in 10^6 per day.

(b) An investigation[28] of a release from a 30 m stack into the atmosphere leads to a similar result. Emission of 10^5 Ci of HTO per year produces a dose of 5 mrem/yr at a distance of 600 m. Here the degree of control is again of order 1 part in 10^6 per day.

(c) As a further example Watson and Mitchell[29] have calculated the effect of a total loss to the environment per day, of 1 ppm of a total site inventory of 3kg of tritium. Again assuming the tritium to be completely oxidised, they find the maximum annual dose to the population to be of order 6 mrem for a critical group, at 500 m from the site.

Carruthers et al.[1] have given a somewhat different calculation. They assume the total world energy demand to be met by fusion and again assume the tritium leakage (as oxide) to be 1 ppm per day. The corresponding steady state activity in the environment is of order 1.2×10^9 Ci. Assuming this activity to be mixed with atmospheric water and the upper 75 metres of the sea, the specific activity is of order 2×10^{-6} Ci/m^3. This leads to a dose rate at least an order of magnitude below 5 mrem/yr. On the global scale then, noting that the distribution of activity is assumed homogeneous, this calculation suggests the environmental impact of tritium to be insignificant.

Thus the above estimates indicate a control of tritium loss to 1 ppm/day: just how conservative or otherwise this value might prove, is unknown. Given the present lack of knowledge concerning the oxidation process, it is essential to err towards caution. Thus it would appear that conceptual fusion reactors should be designed with anticipated losses of 10^5 Ci/yr in mind. In fact, for the STARFIRE and SS-Li-He[4] reference reactors it is claimed that tritium loss rates of order or less than 10^4 Ci/yr are feasible. The release rates for the various components of the two conceptual reactor studies are shown in Table III.

TABLE III SUMMARY OF CALCULATED TRITIUM RELEASES TO THE ENVIRONMENT FROM COMPONENTS OF FUSION REACTORS (reference (4))

	Tritium release rates (Ci/yr)		
	STARFIRE		SS-Li-Hl reactor
	Atmospheric	Aquatic	Atmospheric
Plasma vacuum & fuel systems[a]	3340-4040		6000-8950
Coolant systems	2740[b]	910	75
Solid waste handling systems	365		365
Total release rate	6445-7145	910	6440-9790

(a) releases occur primarily during infrequent maintenance periods of a few days duration.

(b) assume 10 ℓ/d at 1 Ci/ℓ.

Radiological dose estimates [4] based on the above table, suggest that for the population living within 80 km of the plant, the dose would be a small fraction of that from natural background sources.

The results given in the Table III should be treated with care. Some have been obtained by taking over comparable experience from HWR's (CANDU), and others from direct estimates of tritium permeation through the appropriate materials. The latter is of particular concern as the diffusion of hydrogen isotopes is extremely complex and depends on temperature, pressure, level of irradiation and the amount of gas entrapped in the material of interest. A brief survey [30] of the current state of this area is given in APPENDIX I. At the present time it would not appear to be possible to give a reliable theoretical estimate of the

degree of tritium containment which could be achieved under fusion conditions.

3.4 Practical Problem of Tritium Control

The question arises as to the degree of tritium control which could be achieved in practice. A limited amount of information is available from present tritium handling facilities. Since the CANDU heavy water reactors have tritium inventories in the range 10^7 - 10^8 Ci, these facilities can provide relevant experience[31],[32]. Pickering A, for example, has a tritium inventory of order 5.0 kg, and this corresponds to an estimated leak rate of 5 x 10^4 Ci/yr (3 ppm of inventory per day). Based on emision data, measured concentrations and meteorological data, an assessment of dose to the public[31] as a result of the operation of Ontario Hydro's nuclear generating stations, has been made. The studies are broad and take account of seven different biological pathways. These include inhalation near the site boundary, skin absorption of airborne tritium near the boundary, ingestion of locally grown fruit and vegetables, etc. The total effective dose equivalent to an average critical group is stated to be smaller than 0.03 mSv. This is equivalent to the statement that the radiation doses outside the fence, but within the vicinity of the site, are a few per cent of that arising from natural radioactivity, and less than 0.05 mSv as adopted in paragraph 3.1. It is claimed that the more recent designs of CANDU have shown a reduction in tritium loss, and this is principally due to simplification and reduction in the number of components.

Tritium inventories in light water reactors are comparatively low, but the losses at reprocessing plants have been given as 72 Ci/day per GW(e)[14]. The French tritium production plant at Marcoule[33] apparently loses 4% of its output per day, and this despite double jacketing of all components. It should be stressed that although such an escape may be highly undesirable it does not necessarily put the surrounding population at significant risk. Thus there is evidence[21], for example, that dilution of tritium gas to non-hazardous levels occurs

rapidly in the atmosphere. Experience in accidental releases of the gas indicates that even large escapes may not result in serious consequences. As already mentioned, an accidental tritium gas release of 0.5 M Ci in elemental form occurred at the Savannah River Plant [21] in 1974. The atmospheric oxidation rate was determined to be less than 1% per day. Deposition in surface water was below the MPC (maximum permissible concentration) and the levels in vegetation etc, did not represent a significant health hazard[21].

Thus given the present situation, it is impossible to predict what the degree of tritium control in a fusion reactor might actually be. Further information on the practical degree of control should come from the Tritium Systems Test Assembly[34] at Los Alamos, which will soon come into full operation with an inventory of 200g of tritium and a requirement not to exceed a loss of 200 Ci/yr to the environment under normal operation.

4. WASTE MANAGEMENT

During routine operation of fusion reactors the structure will become activated to widely varying levels of activity. The literature on the treatment and disposal of this material is discussed in the present section.

4.1 Primary Waste

Most of the radioactivity will come from activation products generated in the first wall and blanket; in STARFIRE, for example, this amounts to 98% of all the activity. The remaining 2% arises from ancillary equipment and components external to the blanket. Estimates of the radiation damage to the first wall and blanket suggest that they be replaced periodically. The replacement frequency and options for handling, storage, and disposal, are very uncertain, but are expected to be design dependent. The most thorough analysis of this problem has been made for the water cooled STARFIRE reactor. Major maintenance operations

for this design include annual replacement of four toroidal blanket sectors and replacement of 24 vacuum pumps and associated isolation valves. The four blanket sectors removed weigh 260 metric tons in total, and contain the materials shown in Table IV. The Table shows both the quantities of waste involved for the different materials, and their levels of activity at times subsequent to removal from the reactor. Wastes arising from the disassembly of the torus are referred to as primary wastes.[4]

4.2 Secondary Waste

These are the wastes arising from routine maintenance, clean-up of reactor coolant and other circulatory streams. These will be of low-activity, and for STARFIRE, the types of waste and their volumes and activities are as summarised in Table V.

With the wastes placed in 0.2 m^3 drums, 4500 drums would be filled annually. Waste volumes derived from treatment of the storage pool water are also listed in Table V. The latter would lead to a further 100 - 1000 drums.

4.3 On-Site Storage

On-site storage requirements will vary significantly for the various types of radioactive materials arising. Low activity wastes may be shipped for disposal immediately after packaging, but some surge storage may be provided to permit operations to be independent of transport availability. However, radioactive waste with a sufficiently high decay-heat generation may require several years of on-site storage to allow for cooling to levels low enough for economic transportation and disposal. For the U.S.A the radiological impact associated with transportation of wastes from reference STARFIRE is expected to be comparable to that for the shipment of radioactive material to and from LWRs.[4]

TABLE IV. CONTENTS OF THE FOUR BLANKET SECTORS OF THE REFERENCE REACTOR

	Weight (metric tons)	Volume (m^3)	Activity at time indicated (MCi/m^3)			
			0	1 year	10 years	30 years
(a) PCA (First-wall)	4.8	0.61	3.82×10^{2}	1.39×10^{2}	1.24×10^{1}	3.73×10^{-1}
PCA (Second wall)	3.4	0.43	1.43×10^{2}	5.0×10^{1}	4.46	1.36×10^{-1}
PCA	66.8	8.5	1.37×10^{1}	4.51	0.40	1.4×10^{-2}
$LiA\ell O_2$	101.0	29.7	6.6	1.46×10^{-6}	1.46×10^{-6}	1.46×10^{-6}
Zr_5Pb_3	54.7	6.1	6.28×10^{1}	3.54×10^{-1}	1.18×10^{-3}	6.33×10^{-6}
Graphite	27.39	17.0	2.32×10^{-8}	2.32×10^{-8}	2.31×10^{-8}	2.31×10^{-8}
Limiter(b)						
Ta-5W	5.23	0.32	3.63×10^{2}	4.06×10^{1}	2.85×10^{-7}	
V-20Ti	1.8	0.32	5.97×10^{1}	1.44	1.30×10^{-3}	2.81×10^{-10}

(a) Prime Candidate Alloy, an austenitic stainless steel.

(b) Only one of the options, Ta or V, would be used in actual practice.

4.4 Recycling

As we have seen the volumes and masses of highly active waste (at shut-down) using conventional materials, such as stainless steel, are comparable to those arising from fission systems. A comparison of the activities of the wastes, however, shows a somewhat different picture. Thus, according to Hafele et al.[3], at shut-down the activity in an LMFBR is about five times higher than that in a stainless steel fusion reactor. The activity of SS316 fusion reactors will be about the same as from the fuel, coolant and cladding wastes in an LMFBR after reprocessing. This equality persists for about 10 years, after which the activity from the long-lived fission products and actinide wastes keep

TABLE V. ESTIMATE OF THE QUANTITIES AND ACTIVITIES OF LOW-ACTIVITY WASTES GENERATED FROM ROUTINE MAINTENANCE AND PROCESS STREAM PURIFICATION AT FUSION PLANTS

			Category 1, solidified concentrates		Category 2, HEPA filters and other dry wastes		Category 3, filters, filter sludges, resins, etc.	
	Total volume* (m^3)	Total amount (Ci)	Volume (m^3)	Activity (Ci/m^3)	Volume (m^3)	Activity (Ci/m^3)	Volume (m^3)	Activity (Ci/m^3)
STARFIRE reactor systems	900	1.2×10^3	205	1.0	540	0.2	155	5.6
Storage pool	20-200	9.0×10^{-2}	10-120	1.0×10^{-4}			5-60	1.0×10^{-6}

*All volumes are those generated per year for a 1000-MWe plant.

the LMFBR wastes approximately one or two orders of magnitude higher than the fusion reactor waste. This comparison shows - apart from the period 1 to 10 years - that fusion systems made from stainless steel can have a distinct advantage over fission systems as regards activity. This bears, of course, on the eventual strategy for recycling and waste disposal. The options are many, and at the present time it is only possible to discuss the more obvious ones, and then, only in the broadest terms.

In the UK, fissile waste from fission systems is reprocessed; this involves transportation and, obviously, entails added costs to ensure adequate safety for the public. The wastes from conventional materials in fusion reactors, on the other hand, do not necessarily have to be recycled and could be stored in an on-site repository. Since the after-heat from fusion wastes is about 0.1 or less of that from fissile wastes, the cooling problem is less severe. Given a fusion economy, however, such a strategy would lead to the disappearance of a valuable reserve of materials. If this is to be avoided, recycling of at least some of the reactor components will be required. Given sufficient development of remote handling techniques perhaps all reactor components could be recycled, with virtually no high level waste at all. If it should prove necessary, however, to consider at least some hands-on maintenance, the question arises as to whether this could be accomplished after an economically viable period of time. Baker et al.[15] have made a detailed investigation of this for the baseline STARFIRE design; they have selected materials appropriate to the role of particular components. On the basis of ICRP recommended standards they take the following criteria for possible material recycling: (i) a dose limit of 0.05 mrem/hr for recycling to the general public, and (ii) a dose limit of 2.5 mrem/hr for recycling for a limited use such as recycling to fusion reactor construction.

Application of these criteria show that limiter components and vacuum pumping shields can be recycled in 30 years at most after they are removed from the reactor. Duct shields associated with the vacuum system require much shorter times. On the other hand, components made of PCA will take 1000 years or more before they decay to 2.5 mrem/hr.

The above discussion is concerned with the use of conventional materials for the first-wall and blanket. As we have seen from the low-activation STARFIRE results, however, there are potentially very significant advantages to be had by carefully choosing wall and blanket materials. It is conceivable that an alloy could be devised which would not require the first-wall to be replaced throughout its lifetime.

4.5 Disposal

The methods of waste disposal depend on the levels of activation under consideration and whether reclamation for reprocessing at some future date will be necessary. The present rules governing waste disposal are different for the U.S.A. and the U.K., being somewhat more stringent for the latter. We shall very briefly review some aspects of waste disposal, as recently investigated in the U.S.A. and the U.K.

(a) U.S.A.

The classification of nuclear waste, as outlined in the guidelines of 10 CFR Part 61, are summarised in Table VI. This guidance is for health and public safety[35]. Waste classifications A, B and C are based on the specific activity of various radionuclides in the waste. The allowable activity limits and restrictions on disposal increase from Class A to Class C. All these classes are regarded as low-level wastes, and 10 CFR61 is not intended to cover the higher levels of activated structures that will be produced, for example, in alloys containing niobium and molybdenum in service in a fusion reactor. The guidelines are appropriate to the evaluation of materials developed for reasons of their low-level or rapid decay of radioactivity and can be used to judge the potential of materials for simple, inexpensive, near-surface disposal.

We now consider the reference STARFIRE wastes in the light of the above classification. According to TABLE I, where the materials comprising the primary waste are listed, only PCA is unsuitable for near-surface burial. The radionuclides of greatest concern are ^{14}C, ^{59}Ni and ^{63}Ni, with half-lives of 5730, 80,000 and 100 years, respectively. The PCA

would have to be disposed of by very deep burial and engineered barriers, or possibly, by deposition in a deep-mine geologic repository. No such wastes are currently disposed of in this manner in the commercial sector. The secondary wastes would, of course, be disposable by near-surface burial.

TABLE VI NUCLEAR WASTE CLASSIFICATION AND STORAGE UNDER 10 CFR 61 RULES

Waste Class	Definition	Disposal
Class A Segregated Waste	Decays to acceptable levels during site occupancy.*	Segregated, Minimum requirements
Class B Stable Waste	Stabilized and decays to levels which do not pose a danger to public health and safety in 100 y.*	Covered to reduce surface radiation to a few percent of natural background.
Class C Intruder Waste	Does not decay to safe levels in 100 y. Decays to acceptably safe level in 500 y.*	≥ 5 meters below surface with natural or engineered barriers.
Waste which does not meet Class C Intruder Waste definition	Does not qualify for near surface disposal - proposed disposal methods are considered on case-by-case basis.	

*10CFR61 defines "acceptable level" to mean that inadvertent entry into the waste, with continuous occupancy would result in less than 500 mrem yearly dose. Natural background radiation gives an average yearly dose of about 100 mrem.

(b) U.K.

Present UK guidelines[36] identify the radioactive material as containing α, β, γ emitters and specify the limits in Ci/g (or Ci/ml) as follows:

(i) Highly active waste is dangerous and must be kep in strong purpose-built stores capable of dissipating the decay heat from the waste. This form of storage is expensive and supervision is essential.

(ii) Once the decay heat has been dissipated, the waste can be packaged and transferred to a well-engineered repository placed deep underground which, when full, can be sealed off. This can be termed geologic disposal, from which retrieval is not intended. Such disposal facilities do not yet exist. The level of activity for which this form of disposal is relevant would be determined by the dose-rate received by the workers operating the plant.

(iii) Solid waste of up to 100μ Ci/g of β,γ emitters with half-lives in excess of 0.5 years can be disposed of at sea.* Tritium and β, γ emitters with half-lives less than 0.5 years are assigned the higher limit of 1 Ci/g.

(iv) The Department of the Environment may authorise the disposal of unpackaged radioactive waste by burial on sites where access is controlled. Each case is judged on its merits and any burial of radioactive waste that has taken place in the past cannot be taken as a precedent.

(v) The disposal of moderately active solid waste by burial is generally permitted at the Drigg Site, owned and operated by BNFL. The relevant limits for β, γ waste are 60 nCi/ml and an unshielded surface dose-rate from the waste of less than 750mr/h.

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*This has now been halted.

(vi) Wastes of specific activity less than 10^{-2} nCi/g* can be made exempt from the regulations and are therefore suitable for unsupervised disposal ('dust-bin' approach).

Some radioactive components removed from a fusion reactor may have to be kept in a special store until the decay heat generation has fallen to an acceptable level[10]. This form of storage might be required for a number of years. Once the forced cooling is no longer necessary the next step in the disposal procedure may be considered. It is probable that all that will happen is that the forced cooling will be stopped and the waste material will be left in the repository until the radioactivity has decayed to such a low level that the final disposal of the material can take place[10].

In order to assess the length of time for which the waste will have to be stored Jarvis[10] has calculated the times needed for all potential "first-wall" elements (primary waste) to decay to each of the four levels; 100μCi/g, 60 nCi/g (not ml, here), 2 nCi/g* and 10^{-2} nCi/g. He concluded that the majority of elements require millions of years to decay to levels compatible with "dust-bin" disposal. The percentage values for each element needed to meet the four activity limits after a 300 year cooling time have also been calculated. Even minor impurities will pose a severe problem for "dust-bin" disposal.

The principal concern is the disposal of structural materials (primary waste). The materials used for neutron multiplication, tritium generation and as coolants may suffer from a degree of radiation damage and from transmutation effects but their mechanical integrity is not a requirement and there need be no disposal other than at the expiry of the lifetime of the reactor, if then.

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*A substance is defined to be radioactive if its activity is in excess of 2nCi/g.

Of the primary structural elements only V and Cr are candidates for dust-bin disposal. Si meets the "non-radioactive" criterion. Relaxation of the requirement to the trench-burial standard permits C to be additionally considered. Only when sea disposal is considered does the choice of structural elements become widened but even here Ni, Nb and Mo must be limited to not more than 2% of the material constitution; of the secondary structural elements, only Cu gives cause for concern. The element acceptability ratings for sea disposal after 300 years are summarised in Table VII.

TABLE VII. SUMMARY OF ELEMENT ACCEPTABILITY IN STRUCTURAL MATERIALS FROM CONSIDERATION OF SEA DISPOSAL AFTER 300 YEARS.

A. Structural elements (intended constituents)	
Primary constituents:	C, Mg, Al, Si, Ti, V, Cr, Mn, Fe, Co, Zr, Ta, W
Minor constituents (1-10%):	Ni, Nb, Mo
Trace constituents (0.1-1%):	Cu
B. Impurity elements	
Unlimited (> 10%):	Li, Be, B, O, F, Na, P, S, Cl, Sc, Y, Cd, In, Tl, Pb
Minor impurities (1-10%):	Ca, Zn, Sn
Trace impurities (0.1-1%):	N
Acceptable impurities (10-1000 ppm):	K, Ag.

In summary, unless a structural material can be devised from Mg, V and Cr, the dust-bin disposal method is inappropriate even after a 300 year cooling period. Sea-disposal levels are more readily attainable, for a wider selection of elements. However, it is most probable that sea-disposal will be strongly discouraged so that purpose-built repositories may be necessary for permanent disposal, in which case the cooling time needed before transhipment is permitted from one repository to another is determined by the decay heat problem alone.

4.6. Conclusions

Construction of fusion reactors from conventional materials will lead to primary waste, a part of which would probably require disposal in a special repository[10), 15)]. This may be a purpose-built structure or a carefully selected deep mine (geologic disposal) with engineered barriers[4)]. In the long term, it may be necessary to reclaim structural materials from fusion reactors; in establishing a repository this might also have to be taken into account[4)]. The lower activation wastes could be disposed of by shallow burial. In this respect, the U.K. rules are more stringent than those pertaining to the U.S.A. Whereas the U.K.regulations are non-specific, those of the U.S. set concentration limits on individual radioisotopes for various classes of waste. As far as we are aware , no government agency in either the U.S.A. or U.K., has put forward any recommendations pertaining specifically to the disposal of material discharged from a fusion reactor.

4.7. Future Developments

We have seen the potential advantage of developing low-activation wall materials. As well as element selection, however, there is an additional route, namely, isotopic selection. For example, extensive efforts are now being made in the field of laser-isotope separation, where the goal is to obtain enriched uranium with a throughput of hundreds of tonnes[37)]. Redirected towards the isotopic separation of the chemical elements in structural materials [38)], there arises the prospect of removing some of those isotopes which are mainly responsible

for the production of long-lived radionuclides so that a reduction of several orders-of-magnitude in the long-lived radioactivity could be effected even when the conventional prescriptions for certain materials (e.g. 316 stainless steel) are retained. This is clearly illustrated in some conceptual reactor studies for UWMAK I[4]; the radioactivity after shutdown for type 316-SS first-walls isotopically tailored to different selectivity levels, is shown in Fig. 12. Thus, isotopic tailoring could beneficially influence the present views of recycling and waste disposal. To obtain the full benefit of this approach, a very stringent policy of excluding impurity elements would be essential. Whether this procedure is technologically feasible on an industrial scale is unknown.

5. DECOMMISSIONING

After the useful life of a reactor, it will be necessary to dismantle its components in preparation for possible recycling or disposal. All the highly active first wall and blanket could be removed in the manner used for routine maintenance. There remains the neutron shielding and magnets, ancillary equipment and buildings. The single largest component in a fusion reactor is the shielding (mainly iron), which in the STARFIRE design is 13,000 tonnes. All but 300 tonnes of this is of sufficiently low activity (1.2×10^{-3} Ci/m^3 after 30 ys) that it could eventually be recycled[4]. The superconducting coils contain 2200 tonnes of copper which also decays to a low level of activity (6×10^{-3} Ci/m^3 after 1 yr). Experience in cutting and dismantling tokamak components will be gained with the eventual decommissioning of JET. Decommissioning should not lead to any basically new recycling/disposal problems.

6. BIOMAGNETIC EFFECTS

Magnetic fields produced by a tokamak reactor should have an insignificant effect on the public since the fields fall to the strength of the earth's field (about 0.5 G) at a distance of order 500 m. Fields inside the reactor areas where personnel may be present are expected to range up to 500 G. However, there is no evidence to suggest that exposure to such fields produces any adverse effects. Indeed, NMR, which

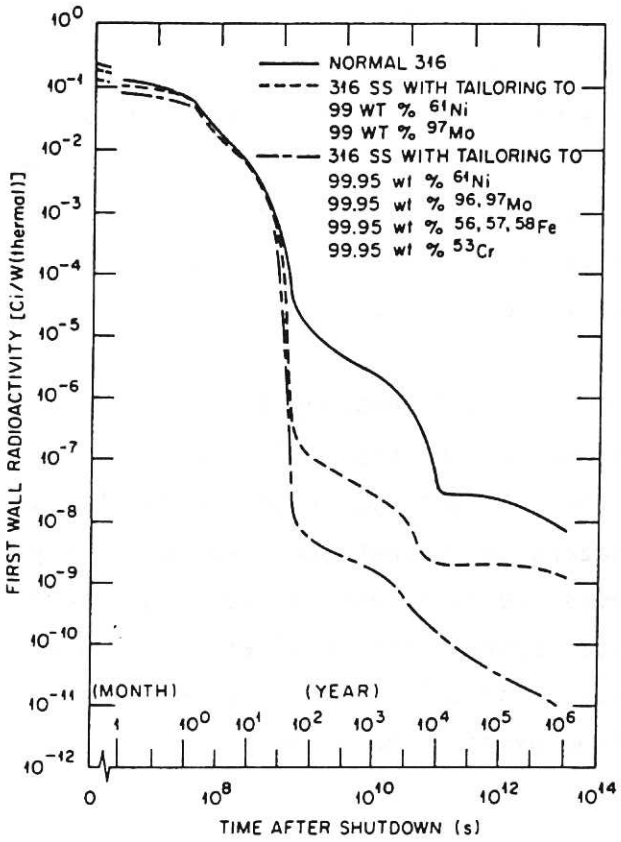

Fig. 12. Radioactivity after shutdown for type 316-SS first-walls isotopically tailored at different selectivity levels. Reactor was UWMAK I, operated for 2 years at 1.25 MW/m^2 before shutdown.

uses much higher magnetic fields, is a diagnostic which is now widely used, without any apparent ill effects. Nonetheless, the NRPB have recently published advice as to the conditions to be fulfilled during the operation of NMR clinical imaging equipment[39]. The principal recommendations are: (1) for the patient the magnetic field should not exceed 2.5 tesla to the whole or part of the body, (2) staff operating the equipment should not be exposed for prolonged periods to more than 0.02 tesla to the whole body or 0.2 tesla to the arm or hands. Unofficial laboratory guidelines vary widely but in general they limit long term whole body exposure to fields of a few hundred gauss; they limit exposure time during a working day to less than one hour for fields of a few kilogauss. The INTOR[6] design study, for example, makes such recommendations. Fusion reactors should be able to meet such guidelines if they prove necessary.

Radio-frequency methods for both heating and driving currents in plasmas are now receiving increased attention. The STARFIRE design, for example, includes facilities for 1.67 GHz operation. Radio-frequency sources present a possible hazard to occupational personnel only[40]. Although most biological effects are attributed to heating, there is evidence that below 30 kHz biological systems can be altered by low intensity electromagnetic fields. The only form of human morbidity seriously suggested to occur from exposure to microwaves is cataractogenesis[41].

7. NUCLEAR SAFEGUARDS

It would be a relatively straightforward matter to produce ^{233}U or ^{239}Pu by bombarding natural uranium with the energetic neutrons from a fusion reactor. It has been stated[3] that this could only be done with the knowledge and full co-operation of the reactor operators.

8. ACCIDENTS (with Ian Cook)

Of particular concern is the possibility of an accidental release of radioactivity to the environment. This can happen only as a result of the joint occurrence of

(a) a release of part of the radioactive inventory to the atmosphere of the containment building, and

(b) a breach or bypass of the containment building.

Thus it is necessary to identify the sources of radioactivity and the mechanisms by which they could be released. Note that the identification of a potential sequence of accidents says nothing about their likelihood. Present conceptual fusion reactors do contain some safety features. These designs, however, do not have sufficient detail to allow anything but a cursory assessment of the accident hazards involved. To our knowledge, no probabilistic risk assessments equivalent to those for fission reactors, have ever been carried out.

8.1. Sources of Tritium

The major components of fusion reactor systems that handle significant quantities of tritium can be grouped as follows:

1. The recycled fuel processing system,
2. Vacuum pumps and fuel injectors located near the torus,
3. The blanket system,
4. The blanket processing system,
5. The main fuel (surge) storage system, and
6. The coolant, storage pool, and atmospheric cleanup system.

We shall briefly discuss each of the above groups in turn, taking the STARFIRE design by way of illustration[4]. The STARFIRE study divides the tritium inventory into two parts: a part "vulnerable" to release and a part "non-vulnerable" to release. This is useful as an initial evaluation of accident hazard but is not based on analyses of specific

accident events or sequences. It should be noted that vulnerability usually refers to the likelihood for release from the primary containment. This does not mean that the material would reach the environment there being secondary and possibly further containments. As far as we are aware, the STARFIRE study does not assign values for probabilities of release to the terms "vulnerable" and "non-vulnerable".

(a) Fuel Processing System

All of the 154-g tritium inventory in the fuel processing system of STARFIRE is considered to be nonvulnerable. The process equipment is installed in a separate building with multiple levels of containment. Accidents or component failures should not contaminate the reactor hall. The maximum credible release from any nonvulnerable component of STARFIRE was assumed to be 50 g of tritium. The atmospheric processing system in the fuel processing building is designed to reduce (i) the exposure of personnel to acceptable levels and (ii) the release to outside the building to a few Ci/day.

(b) Vacuum Pumps and Fuel Injectors

Although the vacuum pumps and fuel injectors are operationally part of the fuel processing system, they are considered separately because they will be located in the reactor hall. The STARFIRE design uses gas puffing to refuel the reactor, so very little inventory is associated with fueling. However, the vacuum pumps can contribute significantly to the vulnerable tritium inventory of order 100 g. The degree of vulnerability of tritium in the vacuum pumps has not been established with certainty. Further efforts will be necessary to reduce the probability of either air (or any other gas) filling the torus (and thus the pump chambers) or loss of liquid - helium cooling to the pumps. Either of these events would release tritium from the pumping surfaces. However, if the integrity of the vacuum seals is maintained, tritium should not be released to the reactor hall.

(c) Blanket System

The principal source of tritium lies in the blanket modules. The inventory in these modules will depend upon the effectiveness of the tritium recovery system. Because the effectiveness of blanket process techniques are unknown, estimates of the tritium in the modules are uncertain. For STARFIRE (solid breeder) the inventory is of order 10 kg - approximtely 90% of the total inventory. All of this is classed as nonvulnerable. This should not be taken to mean that no tritium can be released from the blanket; it should only mean that the probability of significant release is low, and the fraction released even under a violent accident would be small. The danger of losing both coolant and tritium recovery capabilities may need to be considered. The loss of coolant would result in a temperature rise in the blanket that could cause an increase in tritium release.

The tritium inventory in a liquid-lithium blanket would have to be considered as vulnerable because it would be possible for any liquid to drain, at least partially, from the reactor after some accidents. The amount of lithium that could drain, the behaviour of lithium under accident conditions, and the consequence of such events are the subject of recent and current studies.[42] However, the potential hazard of such events is the principal reason for the selection of $LiA\ell O_2$ in the STARFIRE design and in the strong current interest in solid breeding materials.

(d) Blanket Processing System

The equipment that recovers tritium from the blanket material can be housed in a separate building with special multilevel containment facilities. Such an arrangement significantly lowers both the potential and consequence of accidents and could result in the nonvulnerable classification for all tritium in this equipment. The equipment for recovering tritium from liquid lithium is likely to be larger, more complex and with higher inventory than that used with $LiA\ell O_2$ (~300 g for STARFIRE).

(e) Main Fuel Storage System

A central storage will be required to buffer plant opertions from short-time disruptions in full recycle or blanket process operations. The STARFIRE design calls for storage of approximately 1 kg of tritium in several containers of UT_x. This amounts to just over one day's feed and is contained in several separate containers in a well-protected cell with multiple levels of containment. Thus, this inventory of tritium is considered nonvulnerable.

(f) Coolant, storage pool, and atmospheric cleanup systems

The coolant, storage pool, and atmospheric cleanup systems are considered as a unit because of their similar safety and containment problems. All contain potentially large quantities of water with low concentrations of tritium. Tritium enters the STARFIRE water coolant by permeation and leakage. Tritium enters the storage pools via contaminated equipment stored in the pools. Tritium removed in routine emergency atmospheric cleanup systems also ultimately exits those systems in tritium-contaminated water. The concentration of tritium in the coolant is expected to be maintained at less than 1 Ci/l. If the total coolant inventory is of order 10^6Ci, this would correspond to 0.1 kg of tritium. Comparable quantities could be present in the atmospheric cleanup system, and, perhaps in the storage pool if components are not fully decontaminated before storage.

8.2 Activation Products

While tritium is the only radioactive material intrinsic to a D-T fuelled fusion reactor, other materials can be activated wherever the reaction-product neutrons reach. The most vulnerable components of the plant radioactive inventory are fluids that have only one level of containment. This includes the reactor cover gas (which may contain relatively small quantities of radioactivity) and the coolant, which can circulate outside the reactor building. A much lower level of vulnerability can be

assigned to the structural components of the reactor as releases of the radioactivity contained in these materials requires melting and aerosol or vapour transport to the external environment. Release of this material which contains 98% of the activation product inventory (in the STARFIRE design), also requires a breach of the reactor containment and of the reactor building. Some control of the radioactive inventory can be achieved by choice of materials as indicated in section 2.3a.

8.3 Potential Accidents

A number of potential accidents may be postulated to cause a release of a part of the radioactive inventory of a fusion reactor. These could lead to the conversion of stored energy into thermal or mechanical energy as a result of system or component failures.

(a) Magnetic Disturbance

Superconducting coils are regarded as essential in the design of an economic fusion reactor. The magnetic fields are typically of order 10 tesla and this corresponds to a total magnetic energy of about 40 GJ. This could serve to initiate an accident sequence.

The principal abnormal condition which might arise is the sudden quenching of the superconductivity[43]. Such a situation could be triggered by a cryogenic instability resulting from the loss of adequate cooling or sudden localised heating. This could lead to excessive heating in parts of the coils and unacceptable thermal stresses. This in turn could initiate other accidents such as loss of coolant or lithium spills, together with the possible release of activity within the containment building. In the present STARFIRE work it is claimed that the system has been designed so that quenching cannot occur.

The question arises as to whether magnet failure could actually lead to a breach of containment. In discussing the various accidents which might occur, Arendt and Komarek[43] consider the simultaneous rupturing of the winding at two locations to be the most severe event. They calculate

that missile generation could occur, the kinetic energy involved being less than values assumed for airplane crashes into the containment structure of fission power plants. Since fusion plants will also have to be designed to withstand aircraft crashes, the magnetic generation of missiles should not lead to a breach of containment. It is recognised that much further investigation into this area is required. The present large Coil Program at Oak Ridge[44] should clarify many aspects of safety and reliability pertaining to large superconducting magnets.

(b) Plasma Disturbances

There are two principal types of plasma disturbance. In the first, an MHD instability can lead to a gross movement of the plasma towards the reactor wall or limiters. A form of this phenomenon occurs in tokamaks and is generally referred to as a major disruption. The cause or causes of major disruptions are not understood and are the subject of much research. The outcome of a disruption is to deposit a proportion of the plasma energy on the first wall. The accident potential of this phenomenon has been considered in the INTOR studies.[6] These show that the energy required to ablate a significant amount of wall material exceeds the design reference plasma disruption energy (approximately $3MJ/m^2$) by two orders of magnitude. Thus plasma disruptions have no severe accident potential although they may present an operational problem. Furthermore, devices which do not depend on a net inductive current in the plasma to provide confinement (e.g. mirror machines and currentless stellarators) are not expected to show this phenomenon. This has been confirmed, for example, by the results obtained from the "currentless" stellarator at Garching (Wendelstein VII A)[45].

The second possible type of plasma disturbance, an overpower transient (nuclear runaway), could occur if the fusion reaction cross-section increased sufficiently rapidly with energy compared to available loss mechanisms[4]. Since a steady-state ignited plasma has yet to be achieved, there is considerable doubt whether this phenomenon could actually arise. Even if it could occur it should not be serious; since for economic reasons a reactor would operate close to the maximum

overpower transient will cause the plasma to disrupt, leading to a dumping of thermal energy as discussed above.

An important concern is to provide a means for an emergency plasma shutdown. If a component in the reactor failed, so that heat removal and plasma control became impossible, then it would be necessary to halt the fusion process. It has been suggested that the plasma could be quenched by the injection of gaseous impurities[4].

(c) Coolant System Failure

The primary coolant system of a fusion reactor removes heat energy from the first-wall and blanket, the latter containing lithium in some form to breed tritium. A large number of coolant-lithium-structure combinations have been proposed for fusion reactors. Loss-of-coolant or loss-of-coolant flow accidents will require prompt shutdown of the plasma heat generating process. A reactor generating 1 GW(e) will have a thermal load in the first-wall/blanket of order 3 GW. Large temperature excursions leading to first-wall/blanket structural failure and possible radioactive material release can occur if the heat source is not shut down sufficiently rapidly. Chan[46] investigated this phenomenon for a stainless steel blanket with two different coolant-breeder combinations: helium with a liquid lithium breeder and water with a lithium aluminate breeder. He found that if the plasma is not shut down, structural melting will occur in 8.0 secs. in the helium cooled design and in 27.0 secs in the water-cooled design. Emergency shutdown could be achieved by the injection of impurities in order to increase the line radiative and Bremsstrahlung losses; the injection of iodine has been proposed for STARFIRE[4].

Following shutdown, the decay heat from activated structural material in fusion reactors is about an order of magnitude less severe than fission reactor core decay heat. Nevertheless, the level is not insignificant, as indicated by calculations of first-wall structure temperatures after reactor shutdown for a loss-of-coolant flow accident[47]. The results

show possible mechanical failure of the wall/blanket materials considered. This could conceivably lead to some other accident such as a liquid lithium spill. It is possible that some release of activation products could occur without gross melting, via the formation of volatile oxides[48]. However, it seems unlikely that afterheat alone could breach the containment and thus could not pose a severe accident threat. In any case, the operation of auxiliary cooling, or possible alternative designs, could effectively remove the afterheat problem. There is no afterheat generation in the plasma.

(d) Hydrogen Explosions

Fusion reactors will contain significant quantities of the hydrogen isotopes deuterium and tritium. Hydrogen can combine explosively with oxygen under certain conditions. Detonation can occur with hydrogen concentrations from the spontaneous ignition limit of about 19% to 59% in air. The consequences of such explosions are a strong function of the total amount of hydrogen available for reaction, building geometry and volume, and the gas chosen for the building atmosphere. Preliminary investigations suggest that fusion reactors could be designed to accommodate such accidents[4].

(e) Lithium Fires

Some early designs of fusion power stations required large quantities of liquid lithium for use both as a breeding material and a coolant. However, all recent reactor designs have adopted liquid or solid lithium compounds as the blanket materials. Liquid lithium reacts strongly and exothermically with oxygen, nitrogen, water and concrete. Both analysis and experiments[48, 49] indicate that lithium spills could have the capacity to melt structural components and volatilise activation products. A localised lithium spill would not threaten containment integrity, but a large-scale loss of lithium accident might threaten the containment and must therefore be designed against. Possible steps to eliminate or adequately mitigate a lithium fire accident include subdivision of the blanket and coolant loops, use of a dump tank below

likely spill areas, steel liners for concrete, and the exclusion of the use of water-coolants with liquid metal blankets. The consequences of a hypothetical large-scale lithium fire might be expected to be comparable, in broad terms, to a sodium fire in a fast reactor. Jeppson[50] has demonstrated that inert gas flows or Graphex powder are very effective at extinguishing fires in small lithium spills at least - up to 15 kg or so. Fusion reactor designs using solid breeding materials (lithium oxide, lithium aluminate etc.) with helium coolant eliminate the lithium fire risk entirely.

Very recently, Piet, Kazimi and Lidsky[51] have attempted to assess the influence of choice of materials on the consequences of fusion reactor accidents.

8.4 External Accidents

In a typical fission power station risk study only about half of the calculated core melt frequency is contributed by internal initiators. The balance stems from external hazards such as earthquakes or aircraft impact. Accidents of this type might breach the containment of a fusion reactor with a probability comparable to that in the fission reactor case. However, the low volatility of the activated materials and the lower BHP associated with them, are an advantage for fusion. The consequences following a release of tritium are discussed in the following section.

8.5 Tritium Release

In the event of an accident, mechanical damage of the blanket, coolant circuits, or other components, could release tritium into the secondary containment of a fusion reactor. Present estimates of the tritium likely to be released under different accident conditions, are based on uncertain assumptions. Thus, for example, the form and composition of the blanket will significantly affect the consequences of any accident. A fire or explosion could, presumably, release a considerable fraction of the tritium from the blanket and tritium processing plant, and

possibly volatilise active material leading to the production of toxic gases. A full and proper study of such accidents has yet to be made. A satisfactory safety assessment must include the consequences of an accidental radiation dose to the public. There are certain generic differences between fission and fusion, however, which suggest the environmental impact of a non-routine release to be smaller in the case of fusion:

i. No actinides or fission products are involved.

ii. The BHP's of tritium and the activation products in a fusion reactor are less than for actinides and fission products,

iii. Tritium does not concentrate in the body, unlike actinides,

iv. The biological half life of tritium in the body is of order ten days,[20] and can be further reduced by an intake of water.

v. Having entered the environment, tritium disperses much more rapidly than most fission products and actinides [see, for example, references (52) and (53)].

Hafele[3] has estimated the number of early deaths in the surrounding population which result from the escape of 10kg of tritium in the form of tritiated water. This study considers a population density of 300 people/km^2. The results suggest that for a relatively short range plume containing tritium (possible worst case), the number of early deaths would be of order a few tens. For fission reactors, however, a large uncontrolled release of the volatile fission products could, under the same conditions, lead to much worse consequences. Given identical meteorological conditions and population density, rough estimates[3] suggest that the number of early deaths from a fission accident would be of order 40 times greater than for a fusion accident. This comparison could well be misleading, however, since it is the frequency of such hazards which determine the risk. Current practice in the U.K. for

fission reactors requires the frequency of such an event to be 10^{-7}/y or less. It remains to be demonstrated whether or not fusion reactors can achieve a similarly low frequency. More detailed studies in this area have been begun by Edlund[54] for NET. Not surprisingly, this work contains many assumptions, the validity of which are uncertain. Edlund states that his results should not be taken too seriously at this stage. Clearly, much more thorough investigations of tritium release from a fusion reactor and its consequences, are called for. As pointed out in 3.4, however, there is evidence from the Savannah River accident that a release of 0.5 M Ci in the elemental form did not lead to a significant health hazard[21]. Thus even large escapes of tritium may not result in serious consequences.

As an independent investigation, we have directly compared the BHP of the tritium inventory in a fusion reactor (20 kg) with that of the volatile fission products released following an accident in a fission reactor. This has been carried out for both a PWR and a fast reactor. The type of accident and other details are given in APPENDIX II. For the PWR we find the BHP to be two to three orders of magnitude larger than for the tritium inventory in a fusion reactor; a similar result pertains to the fast reactor.

8.6 Overview of Accidents

It should be clear from the preceding paragraphs that accident studies for fusion reactors are in an early stage of development. There is, for example, nothing to compare with the Sizewell B degraded core analysis for a PWR reactor[55]. The potential safety advantages of fusion power is perhaps one of the most important reasons for investigating fusion. The essential safety factors of the D-T reactor would appear to be

i. the low potential for an overpower situation;

ii. the fact that radioactivity can, in principle, be controlled by the reactor design;

iii. its low afterheat power density; and

iv. the relatively low BHP of radionuclides in the tritium and activated structures following release.

The advantage associated with the type of radioactivity contained within a fusion reactor as compared to fission, is worth stressing. The essential difference is that the radioactivity in a fusion reactor will comprise fixed activation products, some active corrosion products, and tritium. A fission reactor, on the other hand, contains activated corrosion products, actinides, fission products and tritium. Some of the fission products are volatile and toxic and lost more readily under accident conditions than the structurally contained activation products. Thus, there are important safety differences between the two types of reactor. Another significant difference is that fission power systems rely on the reliable operation (on demand) of active safeguard systems. Although such systems may be required to prevent economic harm in a fusion reactor, there is no evidence at the moment that these will be necessary for safety and environmental reasons.

9. GENERAL CONCLUSIONS

The work of Carruthers et al.[1] drew attention to certain essential features associated with the operation of D-T fusion reactors. In particular, they estimated the necessary degree of tritium containment, and further noted that an inventory of radioactive isotopes would be produced by the fusion neutrons, and that lithium presented a potential source of accident. These, and many other aspects, were much more fully examined by Hafele et al.[3], the latter also making a comparison between the environmental impacts of fusion and fast breeder reactors. Broadly speaking, the topics discussed in these earlier reports still remain the questions of most importance. Moreover, many quantitative results which were quoted in these earlier works have not been significantly revised. The intervening period, however, has seen a developing interest in conceptual fusion reactor designs. The most sophisticated of these is STARFIRE, and in discussing particular features of environmental impact and safety we have made frequent reference to this design. Where possible we have made comparison with the environmental impact of fission reactors. Given the uncertainties of the basis upon which these comparisons are made, the latter should only be regarded as illustrative. Nonetheless, a fusion reactor does have the intrinsic advantage over a fission reactor in that it will not produce actinides and fission products.

We now give a brief survey of the salient features which have emerged during this review.

1. Fusion reactors will contain a substantial inventory of radioactive isotopes after one or two years of operation. This will be of order 1 GCi/GW(th), principally concentrated in the first-wall and blanket materials. It will present no hazard to the public under normal operation, and the level of activity at shutdown will be similar to that arising from a fission reactor of the same capacity. In terms of biological hazard potential, however, the radioactive inventory

of a fusion reactor shows a significant advantage over that of the fast breeder reactor[3]; this is clearly illustrated in Figs. 9 and 10.

2. The STARFIRE studies indicate a primary waste of order 250 tonnes per annum (volume of 62 m^3), which arises from the replacement of first-wall and blanket. The disposal classification according to U.S. regulations would allow near-surface burial for all materials except PCA. The latter constitutes approximately 70 tonnes (9.5 m^3), its activity being principally due to ^{14}C, ^{59}Ni and ^{63}Ni. The disposal of PCA would require the provision of special repositories; these could, of course, require facilities for eventual reclamation of components for recycling. We note that a typical high-level vitrified reprocessed radwaste from a 1 GW(e) liquid fast breeder reactor is estimated to be of order 12 m^3 per annum. The waste disposal problem could be significantly reduced by using less conventional materials such as vanadium and titanium or low activity alloys. Whether these, or any, materials can be developed with the appropriate physical and mechanical properties, remains to be seen.

3. Calculations by different authors indicate that the tritium inventory of a fusion reactor would need to be contained to 1 ppm/day (equivalent to a loss of 10 - 100 ci/day). There is evidence that the present CANDU heavy water reactors are achieving currently acceptable containment. Whether the above limit could be achieved in a fusion reactor, however, is unknown. Present knowledge of tritium permeation is not good enough to make a prediction of the ultimate degree of containment.

4. Decommissioning of fusion reactors is not expected to lead to any fundamentally new problems.

5. There is no evidence to suggest that exposure to strong steady magnetic fields produces any adverse effects. Nonetheless, the NRPB have recently published advice as to the conditions to be fulfilled during the operation of NMR equipment. Most laboratories set unofficial guidelines, which fusion reactors should be able to meet if they prove necessary. There is some evidence, however, that cataractogenesis can be caused by exposure to low intensity microwaves.

6. Accident studies for fusion reactors are in an early stage of development. Fusion reactors, however, do have a number of potential safety advantages and these have been discussed. The sources of radioactivity and types of accident have been listed. Preliminary estimates and observations suggest that an accidental release of tritium will not necessarily lead to a significant hazard for the population. In an independent investigation (see Appendix II) we have directly compared the BHP of the tritium inventory in a fusion reactor with that of the volatile fission products released following an accident in a fission reactor. We find that the BHP for the fission product release from a PWR or fast breeder reactor to be two to three orders of magnitude greater than that for the tritium inventory in a fusion reactor. Unlike fission systems, however, no fault-tree analyses or probalistic risk assessments have been carried out. These would require detailed fusion reactor designs which are not available at the present time.

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APPENDIX I

Tritium Permeation and Materials

To achieve adequate containment requires choosing processes and materials and designing components to minimise the inventory and leakage rate. To estimate these quantities we need quantitative data on tritium solubility constants (S), diffusion coefficients (D) and permeation constants (P) for all materials likely to be employed in a reactor system. In particular, it is important to know the dependence of these coefficients on temperature (up to 800^{0}C say), and on pressure (down to 10^{-10} to 10^{-12} torr). Also temperature gradients may quite markedly increase or decrease rates of transport, depending on the size and magnitude of Q, the "heat of transport". Thus Q is another quantity of interest, at least for materials likely to be in regions of steep temperature gradient.

A summary of our present understanding of experiment and theory with regard to the permeation of tritium through different materials has recently been made by Le Claire[30]. We briefly review some of the more important features. There is a considerable body of data on the solubility, diffusion and permeation rates of hydrogen in many materials, mostly metallic, and this is being thoroughly assessed at the present time. Similar data for tritium itself is unfortunately very sparse indeed. However, there is little error made in estimating tritium behaviour at elevated temperatures from hydrogen data, especially bearing in mind other uncertainties of temperature, impurities and radiation effects that will prevail under the conditions for which the estimates are being made. At temperatures well below room temperature on the other hand, isotope effects can be very large indeed, due to the onset of quantum effects made manifest by the low masses. For example, at -125^{0}C, the ratio of the diffusion coefficients for hydrogen and deuterium, D_H/D_D, is 6 for V, 20 for Nb, and maybe even larger for Ta. These are the only three metals studied to such low temperatures so it is not known how widespread such giant effects might be. They suggest though the interesting possibility of using membranes of such materials for

separation of hydrogen isotopes, especially as their diffusion coefficients are high and their solubilities are large and increase with decreasing temperature.

While much of the existing data on solubility and transport of hydrogen is good, information is lacking in many important respects. For some of the materials of interest the data are unreliable. The permeation behaviour of Nb and V is an example: reported measurements are in very poor agreement and incompatible with reliable solubility and diffusion data. It has been suggested that surface effects are a major factor in their permeation to hydrogen, but the nature of these is not understood. Also, the hydrogen diffusion constants for W and Mo are very inadequately established. Even for Fe and steels, on which so much work has been done, reported measurements of the diffusion coefficients of H show, at temperatures below 150°C, widely disparate results for reasons that are not yet properly understood. For all these materials further measurements are necessary. Then again, there are some materials for which data are very incomplete, such as ceramics and many of the proposed breeder materials. The LiH and LiD systems have been well studied recently, but if interest grows in any of the Li alloys or salts that have been suggested, measurements of their solubility and transport properties will be needed. Data on heats of transport are also very sparse, being limited to a few transition metals and even fewer alloys.

Even where good data exist there is comparatively little quantitative information on the influence of radiation damage and impurities on transport rates and solubilities, although these are known to exercise a significant effect in some cases. Radiation damage generates microstructural changes like dislocations and vacancies. These can serve to trap tritium amd markedly alter effective solubilities and transport rates. There is very little understanding of such effects, especially following the very high radiation levels that will arise in a fusion reactor. A particular aspect of this is that radiation damage may impair the effectiveness of some of the surface coatings of relatively impermeable materials that have been suggested to reduce overall permeation rates through some components.

Finally, it is to be remembered that nearly all permeation measurements have been made at pressures in experimentally convenient ranges - 10^{-1} to 1000 Torr, although some workers have recently ventured down to 10^{-6} Torr. At the higher pressure the permeation rate is proportional to the square root of the pressure. To use the high pressure data to extrapolate to the very low pressures of fusion interest - 10^{-10} torr, can lead to very serious error. There is evidence that at very low pressure the permeation rate is directly proportional to the pressure. If permeation rates where pressures are low are in practice much less than estimated from high pressure data, as they mostly have been, much larger tritium pressures might in fact be tolerable for a given level of containment. A beneficial consequence of this would be a reduction in the amount of breeder material needed to be processed for a given production rate. Again, with much smaller permeation rates other routes for leakage (glands, joints etc.) that might previously have been thought of as secondary, might become of primary importance. Measurements of permeation rates down to very low pressures on relevant materials are clearly needed to establish the pressures at which these reduced permeation rates set in.

Work at Oak Ridge National Laboratory indicates that oxide layers formed in heat exchangers may significantly reduce the permeation rate[56]. To further reduce permeation, blankets may need to operate at low tritium concentrations and employ barriers such as double walls and surface coatings. A glass surface coating formed by surface treatment with phosphoric acid is an example of an innovative approach that appears to be very effective in reducing tritium permeation[57]. As more information on the technical and economic feasibility of such techniques becomes available, better estimates of the operational release to be expected can be made.

APPENDIX II

Comparison of Radioactivity of Tritium in Fusion Reactors and Volatile Fission Products in Fission Reactors

(with M.R. Hayns (SRD))

In the section on tritium release we discussed Hafele et al.'s work on comparing the radiological consequences following accidents in fission and fusion reactors. This appendix addresses the same question but from an alternative point of view. In core melt sequences in fission reactors it is only the volatile fission products which have the potential for release to the atmosphere; Table VIII shows the various categories of fission products. Here we compare the BHP of the whole tritium inventory (20kg) in a fusion reactor with the release of volatile fission products associated with the worst accidents currently thought possible. This comparison is made for both a PWR and fast reactor. Since the DAC for elemental tritium is $2{\cdot}5 \times 10^4$ larger than for HTO, we suppose that the entire release of tritium is in the form of tritiated water. It is considered that a 100% release of volatile fission products is not possible, and even the fractional releases considered here, are very unlikely. Despite its limitations, this form of comparison is the only one currently worthwhile, since a full probalistic risk assessment for a fusion reactor has yet to be done.

(a) PWR

We consider the PWR specified by CEGB with a thermal power of 3411 MW fuelled with enriched uranium (3.2% uranium-235). The inventory of radionuclides depends on the operating history of the reactor. We take the inventory specified by CEGB and quoted by Kelly and Clarke[58]. This was evaluated for an equilibrium core, assuming continuous operation at full power, and is the inventory at the end of a refuelling cycle. The radionuclides listed are limited to those of potential significance in determining the radiological consequences[58]. The complete inventory - if required - can be obtained from the FISPIN code (version 5.2)[59]. In assessing the consequences of a degraded core accident, twelve release categories (UK1....UK12) have been investigated. For the present purpose we simplify the study by choosing the UK1 release category as

TABLE VIII Fission Product Categories

Group	Category	Family	Principal Form	Element
I	Noble Gases	Noble Gas	Elemental	Xe, Kr
II	Halogens	Halogen	Elemental	I, Br
III	Volatile Solids	Alkali Metal	Metal	Cs, Rb
		Transition	Metal	Ag, Cd
			Metal	As, Se, In, Sn, Sb, Te
IV	Low Volatility (or "Nonvolatile") Solids	Transition	Metal	Tc
		Noble Metal	Metal	Ru, Rh, Pd
		Alkali Earth	Oxide	Sr, Ba
		Transition	Oxide	Mo, Y, Zr, Nb
		Rare Earth	Oxide	La, Ce, Pr, Nd, Pm, Sm, Eu, Gd
		(Lanthanides)		

illustrative of large releases. This choice is made for the following reasons: (a) it represents one of the largest releases of the very volatile species I and Cs[58], and (b) it is dominated by a single readily identifiable accident sequence. It does not, however, describe the largest release possible of less volatile species such as Te, Ru or La, which have a significant BHPs. From Table 4.2 of reference (60) release categories UK2 to UK5 are different in detail, but similar in impact, to UK1 and the total BHP is only insignificantly affected. Like the work of Hafele et al., the present study does not consider the probabilities of such releases.

TABLE IX RELEASE OF ELEMENTS DURING FOUR HOURS FOLLOWING UK1 ACCIDENT IN PWR

Element	Half-life	Fraction of Core Inventory Released	Activity Bq	BHP(air) Based on ICRP 30 m^3
^{88}Kr	2.8h	0.9	2.1×10^{18}	3.0×10^{13}
^{87}Kr	76 min	0.9	1.5×10^{18}	7.5×10^{12}
^{131}I	8 days	0.7	2.4×10^{18}	3.4×10^{15}
^{132}Te	78h	0.3	1.6×10^{18}	4.0×10^{14}
^{132}I	2.3h	0.7	3.5×10^{18}	3.5×10^{13}
^{133}Xe	5.2d	0.9	6.2×10^{18}	1.5×10^{12}
^{133}I	21.0h	0.7	4.8×10^{18}	1.2×10^{15}
^{134}I	52 min	0.7	5.4×10^{18}	7.7×10^{12}
^{134}Cs	2y	0.5	1.9×10^{17}	9.6×10^{13}
^{135}Xe	9.1h	0.9	1.5×10^{18}	3.0×10^{12}
^{135}I	6.6h	0.7	4.5×10^{18}	2.2×10^{14}
^{136}Cs	13.2d	0.5	6.6×10^{16}	6.6×10^{12}
^{140}Ba	12.7d	0.06	3.6×10^{17}	1.8×10^{13}
			Total :	5.4×10^{15}

$$\frac{\text{20kg of T(HTO)}}{\text{Tritium Inventory in Fusion Reactor}} \qquad 1.0 \times 10^{13}$$

The fractions of core inventory (for each radionuclide) released over a four hour period following a UK1 accident, are shown in Table IX. Also shown are the corresponding activities and biological hazard potentials (in air) evaluated at four hours. The largest BHPs are seen to arise from ^{131}I, ^{133}I and ^{132}Te. The rare gases are not absorbed in the body, and represent only a small part of the potential hazard to an individual. We observe that the total BHP for the volatile fission products is two to three orders of magnitude larger than that due to the total tritium inventory in the form of HTO.

(b) Fast Reactor

In the case of the fast reactor, the spectrum of fission yields is slightly different from the thermal case. For fast reactors we do not have as detailed a study of accidental releases as is available for the PWR. In order to ascertain consequences, we have taken the data for the core inventory and release fractions as published by Kelly et al.[61] in their study of a fast breeder reactor with an installed capacity of 1300 MW(e). This work includes the core inventory of the volatile species and recommends a notional release fraction of 10%. No estimates are available of the probability of such a release. Table X shows the activities of the various volatile fission products and their associated BHPs at shut down. Again we observe the total BHP for the volatile fission products is two to three orders of magnitude larger than that due to the total tritium in a fusion reactor, assumed released in the form of tritiated water.

(c) Conclusion

The above considerations show that in the absence of any discussion of frequency, the notional large releases from either a PWR or a fast reactor lead to BHPs at least two orders of magnitude greater than the BHP associated with the total release of tritium (20kg) in the form of tritiated water.

TABLE X. RELEASE OF ELEMENTS FOLLOWING 10% VAPORISATION OF CORE IN A FAST BREEDER REACTOR

Element	Half-life	Fraction of Core Inventory Released	Activity Bq	BHP(air) Based on ICRP 30 m^3
^{88}Kr	2.8h	0.9	1.1×10^{18}	1.6×10^{13}
^{88}Rb	18m	0.8	1.0×10^{18}	5.0×10^{11}
^{131m}Te	30h	0.2	8.0×10^{16}	1.3×10^{13}
^{131}Te	25m	0.2	5.4×10^{17}	6.8×10^{12}
^{131}Sb	26m	0.2	5.2×10^{17}	1.3×10^{12}
^{131}I	8d	0.9	2.7×10^{18}	4.0×10^{15}
^{132}I	2.3h	0.9	4.0×10^{18}	4.0×10^{13}
^{132}Te	78h	0.2	8.1×10^{17}	2.7×10^{14}
^{133}Te	2m	0.2	6.5×10^{17}	8.0×10^{12}
^{133}I	21h	0.9	5.0×10^{18}	1.3×10^{15}
^{133}Xe	5.3d	0.9	5.0×10^{18}	1.3×10^{12}
^{134}I	52m	0.9	5.3×10^{18}	7.6×10^{12}
^{135}I	6.7h	0.9	5.0×10^{18}	2.5×10^{14}
^{135}Xe	9.1h	0.9	5.3×10^{18}	1.1×10^{13}
^{136}Cs	13d	0.8	1.5×10^{17}	1.5×10^{13}
^{137}Cs	30y	0.8	8.9×10^{16}	4.5×10^{13}
			Total :	5.9×10^{15}