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Waste Generated by Fusion Reactors

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Abstract

Although the fuel cycle of a fusion reactor does not involve input of radioactive material (except the short-lived tritium isotope of hydrogen) or generate radioactive waste products, the neutron flux in the core of the reactor will generate parasitic radioactivity. Consequently fusion reactors will generate radioactive waste during operation from routine replacement of irradiated components and following decommissioning. By surveying six D-T conceptual reactor designs (three tokamaks, two reversed-field pinches and a tandem mirror) the volume of this waste has been estimated. In the survey the reactors are broken down into their major components i.e. first wall, blanket, shield, magnets etc, and auxiliary equipment has been added where appropriate. Externals such as reactor buildings and heat exchangers have not been included at this stage. The radioactivity present in the different components of the waste varies by many orders of magnitude and the volume requiring disposal in a nuclear waste repository can be minimised by separating out those low activity components which either qualify as non-active waste or can be recycled.

The estimated volumes of packaged waste for direct disposal from these fusion reactors, resulting from the replacement of highly irradiated components and decommissioning, vary from 5000 to 43000 m^3/GWe. The reasons for this wide variation are discussed.

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Contents

1. Introduction

Fusion reactors will, if successfully developed, generate useful energy from the fusing together of light atoms - typically deuterium and tritium (DT), which are isotopes of hydrogen. By heating these atoms to very high temperatures, where conditions are favourable for fusion, they become ionised - that is the electrons are stripped away from their parent nuclei - and the resulting plasma can be contained using magnetic fields. Several geometries of magnetic confinement have been considered as routes to commercial fusion reactors, although the tokamak is judged most likely to be successful. Although the fusion fuel cycle does not create significant radioactive material some radioactivity will be created - mainly from neutrons emanating from the D-T reaction causing transmutations in the reactor blanket, shield and mechanical structure. Radioactive waste will arise as these materials are replaced during operation and following decommissioning. In this report, the quantities of this radioactive waste have been estimated, as a preliminary stage to a later assessment of the environmental impact of such waste.

The approach has been to examine several conceptual designs for fusion reactors and estimate the total quantity of material reduced to waste over the life of the reactor. From this the normalised waste generated related to the nominal reactor power output, in the form tonne/GWe or m^3/GWe has been calculated for each reactor. This assessment only considers the materials of the reactor core comprising first wall, divertor/limiters, blanket (including multiplier and breeder material), shield, magnet coils and auxiliary heating and fuelling systems. It is possible that other external components, such as heat exchangers, may add to the radioactive waste due to contamination, but lack of sufficient design data prevents compilation of a full inventory at this stage. Where suitable information exists, the waste has been categorised into non-radioactive and radioactive material (or, more strictly, non-repository and repository material), and the latter sub-divided into radioactive categories. A delay of 50 years after shutdown is assumed, unless stated otherwise, before disposal of the waste.

The reactor systems studied are three tokamaks, two reversed field pinches and a tandem mirror. All six are based on magnetically confined D-T plasmas, five being in toroidal geometry and one a linear system. All six represent early stages of fusion power development, one being a demonstration reactor, one a first-of-a-kind power reactor and the remainder commercial reactors. They provide a range of waste estimates so that the conclusions are not confined to one concept.

2. General Description of Reactors

Tokamak reactors require high toroidal magnetic fields (5 to 8 T) to confine the plasma, utilizing superconducting coils at liquid helium temperature. Substantial shielding is required to protect these coils from the neutron flux and to limit the heat deposition at cryogenic temperatures. Since tokamaks operate with low values of the plasma pressure ratio β (5 to

10%) the level of the power loading of the first wall and blanket is usually limited (2 to 5 MW/m^2).

The following tokamak reactors were considered in this study -

STARFIRE This comprehensive study was completed in 1980, and is described in reference [1]. It is a continuously operating commercial tokamak fusion power plant. The main structure is AISI 316 stainless steel which becomes highly activated as does the ZrPb multiplier. The design of the bulk shield was optimised by selecting a construction made from layers of steel (Fe1422), titanium hydride (TiH_2) tungsten (W), boron carbide (B_4C) and a titanium alloy (Ti6Al4V). The optimisation is based on maintaining adequate shielding for the superconducting magnets whilst reducing the activity in the shield itself to facilitate subsequent recycling.

The DEMO R254 study [2], completed in 1985, considered the technical feasibility of INTOR-type tokamak reactors and concentrated on the design of the nuclear blanket. Two versions were considered; one in which the breeder is a lithium ceramic, lithium metasilicate, and one with liquid lithium-lead breeder. In both versions the neutron multiplier was beryllium and the main structural material a ferritic/martensitic stainless steel. The coolant, helium at high pressure, was also used as the main tritium carrier. The first version is considered in this report.

PCSR-E (Prototype Commercial-Sized Reactor-E) is a 1250 MWe reactor based on the design of the Next European Torus and a direct extrapolation of present experimental physics results [3]. It was completed in 1986. A water-cooled lithium-lead breeding blanket is included, with the lithium 90% enriched.

Reversed field pinch reactors, in contrast to tokamak reactors, can operate with higher values of the plasma pressure ratio β (20 to 30%) and with relatively low values of the toroidal magnetic field. As a consequence the toroidal field coils may be superconducting or of normal conductors such as copper, and in the latter case substantially less shielding is required. The higher values of β also open the possibility of more compact reactors, with increased levels of the power loading of the first wall and blanket (10 to 20 MW/m^2).

The following RFP reactors were considered in this study -

CRFPR (Compact Reversed Field Pinch Reactors) [4], completed in 1985, are compact reactors with a high power density. For the purposes of this study the highest power density case (CRFPR (20)) was taken which has a LiPb - cooled blanket with pressurised water used as coolant for the first-wall, pumped-limiter and structural shield systems. Copper magnets are used throughout thus reducing the amount of shielding required. This, together with a high wall loading, gives a reactor in which the fusion power core has a volume which is 6% of STARFIRE for a similar electrical output.

TITAN, [5], was completed in 1987 and is a study to determine the technical feasibility of an RFP reactor, at high power density. The TITAN-I option considered here uses an integrated blanket and toroidal field coil concept of liquid lithium breeder and conductor inside a vanadium alloy structure. These are relatively low activation materials. The equilibrium field coils are superconducting and impurity control is by means of a toroidal divertor. Other features (eg high wall loading and overall compactness) are similar to those of CRFPR.

Mirror reactors are linear systems, with enhanced magnetic fields at the ends to reflect the contained particles. They have the advantage of very high values of the plasma pressure ratio β (40% to 80%), and do not depend upon induced currents in the plasma.

The following mirror reactor was considered in this study -

WITAMIR-I, [6], completed in 1980, is a study of a D-T Tandem Mirror to minimise the recirculating power requirements. It is a power plant operating in the steady state with both radio-frequency and neutral beams used to create thermal barriers. The study emphased reliability rather than low environmental impact. Superconducting magnets provide the confining field and the breeder/multiplier/coolant is liquid $Li_{17} Pb_{83}$.

The main characteristics of the above reactors are listed in Table A1 of the Appendix.

3. Waste Categories and Disposal

In most countries, radioactive waste is divided into categories which determine its disposal route. In general higher categories of waste are stored at their site of origin until heat generation levels are low enough for disposal, and are then buried in deep repositories. Less active waste may, in some countries, be disposed of in near-surface repositories. Material with very low levels of activity may be suitable for recycling or for local disposal.

The UK classification [7,8] is based on the total activity per unit mass (A) irrespective of the isotopes present. The categories are:

Non-Active Waste	$A < 0.4$ MBq/te
LLW (Low Level Waste)	0.4 MBq/te $< A < 4$ GBq/te (α activity) 0.4 MBq/te $< A < 12$ GBq/te ($\beta + \gamma$ activity)
ILW (Intermediate Level Waste)	$A > 4$ GBq/te (α activity) $A > 12$ GBq/te ($\beta + \gamma$ activity)
HLW (High Level Waste)	waste requires active cooling.

Non-active waste, because of its low activity, can be treated for disposal purposes as non-nuclear waste. The other categories require special repositories for disposal and in future repository waste will, according to current government policy, require deep geological burial although the options for shallow land burial and deep ocean disposal are being retained for bulky decommissioning waste [9].

In the USA, waste disposal is governed [10] by the code 10CFR61, which defines the concentration limits for individual isotopes for several classes of disposal. These classes are:

Class A Segregated waste, which decays to acceptable levels during site occupancy.

Class B Stable waste, which decays to levels that do not pose a danger to public health and safety in 100 years.

Class C Intruder waste, which decays to an acceptably safe level in 500 years.

The above classes of waste may be disposed of by near surface burial; in the case of Class C, at a depth of more than 5m and with natural and engineered barriers. Waste with higher levels of activity than are allowed in Class C must be disposed of in a geological repository.

In the studies considered in this report, the STARFIRE, CRFPR and TITAN wastes are classified according to the US regulations. However, the STARFIRE waste has been recalculated on the basis of UK regulations.

In this study the waste masses and volumes for each reactor have been calculated in the following categories:-

(a) Non-active waste. In the U.K. this can be disposed of without special precautions. However, non-active waste is still defined as a radioactive substance [7] and cannot be re-used without restrictions.

(b) Repository Waste for Direct Disposal (net). This is simply the total waste less (a) above.

(c) Repository Waste for Direct Disposal (packaged). The increase in repository waste volume due to packaging has been included.

4. Reactor Components and Waste Generated

Fusion reactors will yield operational waste by the routine replacement of first and second walls, divertor/limiters, blanket structure, reflector and in some cases parts of the shield. The total volume of this operational waste due to component relacement is estimated on the basis of the component lives quoted in, or deduced from, the individual studies. The breeder, multiplier and reflector materials may be reprocessed to give effectively longer replacement periods. Thus in STARFIRE it is deduced from Reference 1 that these materials effectively last for 20 years. For the other studies it is assumed that these materials last for the reactor lifetime. Since little information is available on any secondary wastes resulting from reprocessing, this is excluded from the study. The operational waste, together with the decommissioning waste, will be stored on site after the reactor shutdown and before disposal, and therefore no distinction is made in this report between the two.

The waste has been normalised to masses or volumes per GWe, calculated on the basis of the nominal full power rating of the reactor, and the volumes are summarized in Table 1. As some of the waste quantities relate to replacement operations during the life of the reactor, there will be some sensitivity to the assumptions on load factor. However, more than 60% of the repository waste arises from components which last for the whole life of the reactor and therefore normalisation to the nominal power rating is more representative of each reactor than normalisation to the energy generated, which depends on arbitary assumptions about the reactor life and availability.

	NON-ACTIVE WASTE	REPOSITORY WASTE	
REACTOR SYSTEM	m^3/GWe (net)	m^3/GWe (net)	m^3/GWe (packaged)
STARFIRE	1720	1940	10400
DEMO R254	2840	1990	10700
PCSR-E	1940	8050	43400
CRFPR	0	906	4870
TITAN	0	1120	6030
WITAMIR	480	3900	21000

Table 1 SUMMARY OF WASTE VOLUMES

Details of the breakdown of the fusion power core of each reactor system together with the appropriate masses, volumes and replacement times based on the published data are given in Tables A2 to A7 of the Appendix. The waste categories of each component after a suitable period for the decay of activity are also included where data is available. Since there is no activation data for the DEMO study the division between non-active and repository waste was scaled from the STARFIRE figure. For a large part of each reactor, the components should not need replacement and the lifetimes quoted are the design lives of the reactors. In the case of the remaining components, the times are quoted in calendar years and are calculated assuming that the reactors operate at the limits of their availability - for lower load factors the lifetimes of these components would increase.

Most of the reactor studies give the assumed first-wall lifetimes, and these are quoted in Table A1 in terms of the neutron fluence to which the first walls are exposed before replacement; in the case of the DEMO R254 reactor the figure quoted is not known to better than a factor of two. In all cases the lifetime of the blanket structure was assumed to be the same as for the first wall.

An important characteristic of RFP reactors for present purposes is the relative thickness of the magnet shield. In tokamaks and mirror reactors there is a massive shield to protect the superconducting field coils; this also serves to shield effectively most of the external components from neutron irradiation. In RFPs operating with toroidal field coils constructed from normal conductors the coil shielding is relatively light so that external components become activated, even those outside the fusion power core such as the biological shield. A full assessment of the radioactive inventory would need to take account of this.

The six systems have not been developed to the same level. In particular, none of the studies considered material outside the fusion power core which may become radioactive. However, where possible, the waste arising from refuelling, additional heating and current drive equipment has been estimated and included in the results. These components add less than 2% to the waste totals. The RFPs (CRFPR and TITAN) do not require auxiliary heating, whilst their oscillating field current drives use the main field coils and are, therefore, included implicitly in the waste inventory.

Some of the waste, whilst active enough to require repository disposal, may be suitable for recycling into future fusion reactors - thus reducing the disposal requirements. This recycling is discussed in the following section.

5. Recycling of low activation materials

The variation in neutron induced activity among the reactor components is large and whilst most material inside the shield must necessarily be consigned to a nuclear waste repository, some of the low-level material could be recycled. However, the recycled material must eventually undergo disposal, so that its volume for n cycles of use will be 1/n times the figure obtained without recycling. The limit for n will depend on a number of factors, in particular the permissible operator dose. Moreover, the long-lived activation associated with this material will not be reduced by recycling, but may increase since for multi-stage reactions the activation is more than linearly proportional to neutron fluence.

Economic and resource considerations may also influence decisions on recycling, particularly for breeders and multipliers. It has been assumed that all breeder and multiplier materials will be reprocessed to give long effective replacement times (20 years in the case of STARFIRE, the reactor lifetime for the other reactors). In addition, a simplified recycling option may be considered in which the breeder, multiplier and some shield materials are recycled for five reactor lifetimes (n=5). For STARFIRE, however, this further recycling of breeder and multiplier has been excluded since these materials contain zirconium and aluminium which give rise to long lived activity and may limit recycling [1].

In the STARFIRE study the possibility of recycling some of the shield and magnet material, which would otherwise be repository waste, was considered in detail. A criterion for recycling was based on the more restrictive of two requirements:-

(i) a contact dose equivalent rate $< 25\ \mu Sv/hr$, and

(ii) a specific activity $< 3.7\ GBq/m^3$

after a 50 year post-irradiation cooling period. The first requirement corresponds to the "hands on" limit for radiation workers (50 mSv for a full working year). On the basis of these requirements the shield and magnets yielded a substantially smaller residual waste volume, since 45% was non-repository waste and of the remainder about two-thirds could be recycled.

The effect of recycling on the volumes of repository waste for three of the reactors considered in this study is shown in Table 2. Where material is both non-active and recycleable it has been placed in the former category. Since (1/n) times the recycleable material, where n is the number of cycles, is included in the totals, the volume with recycling represents a steady state level for a continuous sequence of fusion reactors. The volumes of waste for disposal are reduced by 30 to 50% by recycling.

	DIRECT DISPOSAL		WITH RECYCLING	
REACTOR SYSTEM	m^3 GWe net	m^3/GWe packaged	m^3/GWe net	m^3/GWe packaged
STARFIRE	1940	10400	1210	6500
CRFPR	906	4870	478	2570
TITAN	1120	6030	800	4300

Table 2 EFFECTS OF RECYCLING ON VOLUMES OF WASTE

6. Comparison of Waste from Different Fusion Reactors

The volumes of packaged waste shown in Table 1 range from 4870 to 43400 m^3/GWe. The two reversed field pinches give the lowest volumes, and the PCSR-E tokamak reactor gives the highest volume.

The lower volumes of waste from the reversed field pinches are a direct reflection of the fact that they are more compact reactors than the tokamaks or the mirror, the fusion power cores of the reversed field pinches being a factor 5 to 8 smaller than the tokamaks by volume, for equivalent electrical outputs. The differences in repository waste generated are, however, much less since the components of the RFPs require more frequent replacement. Furthermore, all the waste in the fusion power core of the reversed field pinches is likely to be repository waste, even after a 50 year cooling period, whereas in the tokamaks a portion becomes non-repository waste after this period. The advantage of the RFPs, in waste terms, may also be further eroded, or even eliminated, when material outside the fusion power core is included, since the lighter shielding of the RFPs may result in more activation of external material such as buildings and auxiliary equipment than in tokamaks.

The three tokamak reactors produce volumes of waste which differ by a factor of four. The PCSR-E gives the largest volume for several reasons, including the fact that all material from the shield is assumed to be repository waste whereas for STARFIRE and DEMO a substantial proportion of the shield is non-repository waste. This emphasises the point that the shield contributes the largest component of the waste, and suggests that the use of low-activation steels and suitable design may have a substantial influence on the levels of waste produced. It should also be noted that extensions of the shielding around pentrations for divertors, vacuum pumping, auxiliary heating systems, etc, are often not well quantified in present studies, but could contribute significantly to the waste produced. The quantity of waste in DEMO is certainly underestimated for this reason.

In the UK, bulky LLW waste with a specific activity in the range 0.4 to 4 MBq/te may, subject to approval, be disposed of in near-surface landfill. For STARFIRE, for which detailed activities are given, about 960 te/GWe of material (mostly from the magnet helium vessel) fall within this range. The corresponding reduction in packaged repository waste would be about 650 m^3/GWe or 6% of the total.

In calculating the volumes of packaged waste, a specific NIREX container has been assumed together with a packing factor of 30%. These assumptions give values of the ratio of packaged volume to mass of waste in the range 0.6 to 1.25 m^3/te, which is consistent with the rough guide of 1 m^3/te often used in estimating quantities of fission waste. It is interesting to compare these values with specific calculations [11] for the disposal of waste from NET in German and Swedish repositories, yielding values of 0.34 to 0.52 m^3/te which appear optimistic, but which may be appropriate to waste from NET.

In the USA a Senior Committee on Environmental, Safety and Economic Aspects of Magnetic Fusion Energy, ESECOM, [12] have estimated the radioactive waste produced by ten tokamak and reversed field pinch reactors, including D-T and D-^{3}He fuelled reactors and hybrid fission-fusion reactors. These cases were developed and analysed with the assistance of a computer model and were chosen to permit exploration of a wide range of materials, power densities, etc. These reactors all produced net electric powers of 1200 MWe, and were assumed to have a 30 year life time and to operate with a capacity factor of 0.65. The unpackaged waste volumes quoted for six of these reactors using D-T fuel and conventional electrical generation plant are shown in Table 3, and fall within the range 400 to 2600 m^3/GWe, the lowest value corresponding to a compact reversed field pinch and most of the high values to tokamaks.

	NEUTRON WALL LOADING	REPLACEMENT PERIOD FOR BLANKET	TOTAL WASTE VOLUME
REACTOR	MW/m^2	years	m^3/GWe
1 V-Li tokamak	3. 2	6	1320
2 RAF-He tokamak	3. 2	6	2010
3 RAF-LiPb reversed field pinch	16. 6	1	400
4 Vi-Li reversed field pinch	14. 6	1	2010
5 SiC-He tokamak	2. 5	6	2600
6 V-FLiBe tokamak	3. 7	6	530

Table 3 WASTE VOLUMES (UNPACKAGED) ESTIMATED IN THE ESECOM STUDY

The volumes of unpackaged waste deduced in this study and the volumes quoted in the ESECOM study are plotted in Figure 1 as a function of the mass power density, defined as the power output per unit mass of the fusion core. (The volume for ESECOM Case 4 has been adjusted since in this case it was originally assumed that the whole of the 0.43 m thick shield would be replaced each year, whereas the shield is not replaced in the other cases. Assuming only 20% of the shield is replaced each year gives a waste volume of 550 m^3/GWe.) The two sets of volumes are comparable, and the trend to lower waste volumes for designs with a high mass power density is noticeable.

Since the shield of a reactor is a major component of the waste volume and is also a major component of the mass of the fusion power core, it is possible to make a simple comparison between the waste volumes from different reactors. In Figure 1 the dashed line corresponds to the volume of fully compacted waste from the core of a reactor, assuming that all components are constructed from steel (ρ = 7.9) and last the whole life of the reactor. In practice the actual volumes of waste requiring disposal in a repository may fall above or below this line.

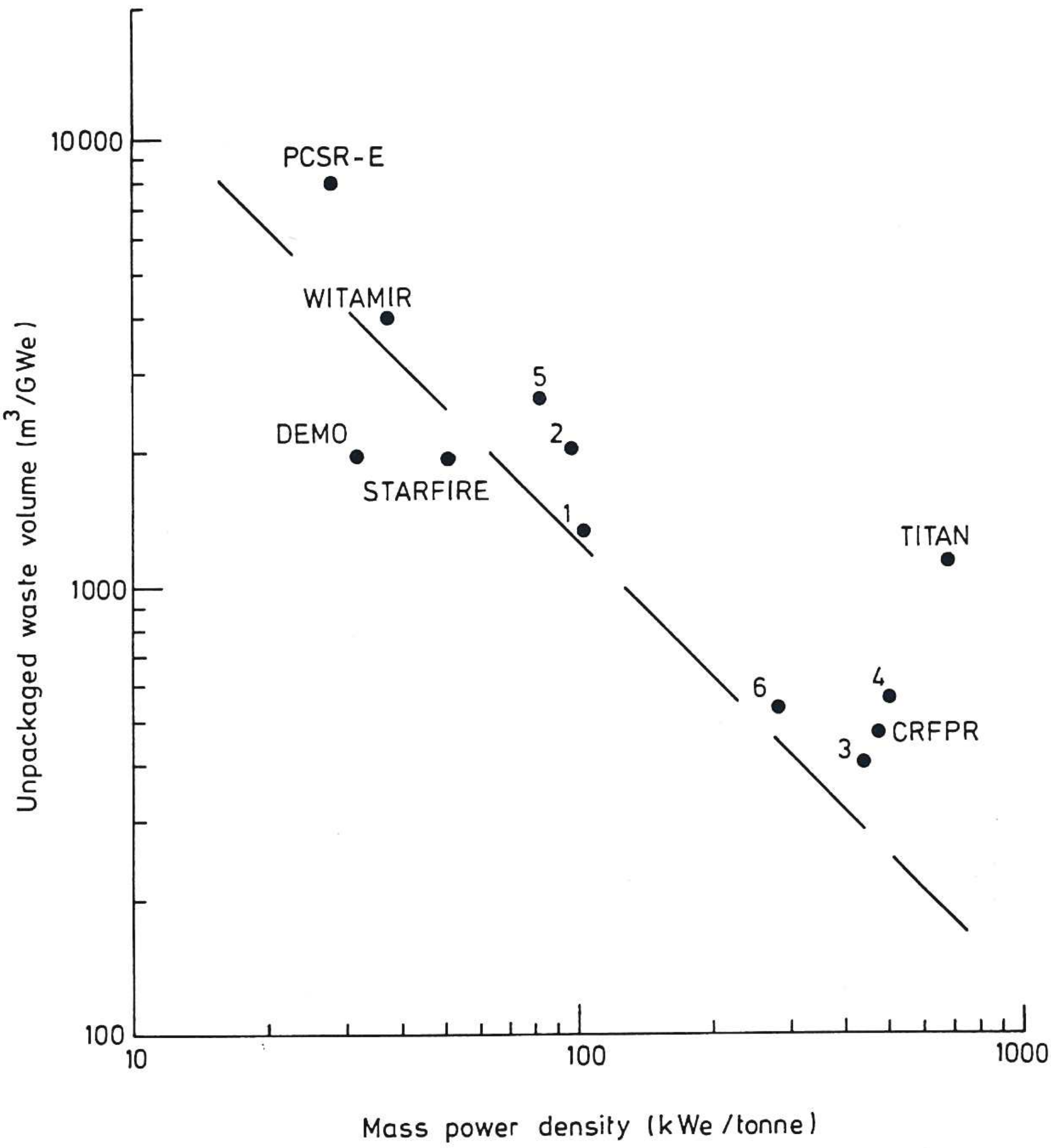

FIGURE 1 VOLUMES OF UNPACKAGED WASTE AS A FUNCTION OF THE MASS POWER DENSITY

Reasons for higher waste volumes are -

- components are replaced several times during the life of the reactor,
- components are included from outside the fusion power core, e.g. the lithium breeder in external pipework of TITAN,
- components are constructed from materials of significantly lower density that steel e.g. the graphite reflector in STARFIRE or
- components contain larger voids or uncompacted volumes are quoted e.g. the blanket structure in PCSR-E.

Reasons for lower waste volumes are -

- some materials have a sufficiently low level of activity that they do not require disposal in a repository or can be recycled, or
- not all active components have been identified at this conceptual stage of the reactor design, e.g. the absence of shielding around ducts in the DEMO study.

In general the volumes of waste will not fall to very low levels for compact reactors with a high mass power density because the frequent replacement of the first wall and blanket structure represents a minimum volume which is dependent on the blanket design.

The scatter in results suggests that the volume of waste from a fusion reactor is not known within an accuracy of a factor 3. A mass power density of greater than 100 kWe/tonne has been proposed as a criterion for the economic viability of a fusion reactor [13], and if this is accepted the volume of unpackaged waste should not exceed 1000 to 2000 m^3/GWe provided components with large voids are reasonably compacted and the shield design is optimised so that a significant proportion of the waste does not require disposal in a repository. The corresponding maximum level of packaged waste is 5000 to 10000 m^3/GWe.

Two other previous studies have also considered the quantities of waste generated by tokamak fusion reactors. Miyahara [14] used the STARFIRE design in his study and estimated a packaged waste volume of 39000 m^3/GWe. This is 3.4 times greater than the figure estimated here, based on the same data. The difference is explained by the fact that no reduction was made for material qualifying as non-active or recycleable and a large proportion of the total volume consisted of additional lead shielding around the packaged waste. Although shielding will be required during interim storage and transportation it will not be needed for final disposal.

Botts and Powell [15] studied four tokamak designs (UWMAK-I, UWMAK-II, BNL and PPPL) and estimated total packaged waste volumes of 240 to 450 m^3/GWe-yr. These values are similar to the volumes of packaged repository waste for direct disposal shown in Table 1.

7. Conclusions

1) A survey of six D-T fusion reactors including three tokamaks, two RFPs and a tandem mirror has provided estimates for the packaged radioactive waste, normalised to the reactor rating, of 5000 to 43000 m^3/GWe.

2) Only the fusion power cores have been considered. Contaminated steam piping and boilers cannot be included due to the lack of design information. For the same reason auxiliary equipment or shielding around penetrations has not always been included, and there are differences between the reactors as to the consideration of liquid metal breeder/coolant in external circuits. The inclusion of these items might significantly increase the waste volumes. Secondary wastes from fuel processing and general maintenance operations are also excluded, because at this conceptual stage of fusion reactor design there is insufficient information on these wastes.

3) The volumes of waste from the tokamak and mirror reactors are dominated by the shield, which is required to reduce the neutron damage and heating in the superconducting coils. By careful design and the use of low activation materials, a significant fraction of this component can be disposed of as non-active waste, or partly recycled, and the active waste significantly reduced.

4) The volumes of waste from the reversed field pinch reactors are at least a factor two less than those from tokamaks, partly due to their compact construction and partly due to the reduced requirement for neutron shielding for normal magnet coils. However, since the RFP designs are relatively lightly shielded they may produce further waste due to irradiation of external components not included in present assessments.

5) The estimates in this report are consistent with most previously published estimates, including those in the ESECOM study. The normalised volume of waste falls as the mass power density of the reactor increases, and the volume of packaged waste is therefore unlikely to exceed 5000 to 10000 m^3/GWe for an economic power reactor. Never-the-less, estimates are uncertain to within a factor three.

8. Acknowledgements

The authors gratefully acknowledge the assistance of Dr W J Worraker in the preparation of parts of the report.

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Appendix

Data for sample reactors

The following tables, with notes, are included in the Appendix -

REACTOR SYSTEM	STARFIRE	DEMO (R254)	PCSR-E	CRFPR	TITAN	WITAMIR
Date of publication	1980	1985	1986	1985	1987	1980
Participant Organisations	ANL (a)	UKAEA	NET	LANL	UCLA (b)	U Wiconsin
Net electrical power GW	1. 20	1. 07	1. 25	1. 00	1. 00	1. 53
Average neutron wall loading (MW/m²)	3. 6	2. 6	2. 2	19. 0	18. 1	2. 4
Major radius (m)	7. 0	6. 8	9. 3	3. 9	3. 9	165 (e)
Minor radius (m)	1. 9	1. 6	2. 4	0. 71	0. 60	0. 72
Plasma current (MA)	10. 1	9. 3	16. 6	18. 4	17. 8	
Toroidal Field (T)	5. 8 (c)	6. 0 (c)	6. 4 (c)	- 0. 4 (d)	- 0. 38 (d)	3. 6 (c)
Blanket Breeder	$LiAlO_2$	Li_2Si0_3	$Li_{17}Pb_{83}$	$Li_{17}Pb_{83}$	Li	$Li_{17}Pb_{83}$
Multiplier	Zr_5Pb_3	Be			Li	
Coolant	H_2O	He	H_2O	$Li_{17}Pb_{83}$	Li	$Li_{17}Pb_{83}$
Design life (years)	40	20	25	30	40	30
Assumed availability (%)	75	75	75	76	75	80
First wall fluence limit (MWy/m²) (e)	16	10	10	15	14	4. 8
Mass power density kWe/te (f)	50	34	28	480	690	38

Table A1 Main Parameters of Reactors

Notes for Table A1 Main Parameters of Reactors

(a) STARFIRE was designed by the Argonne National Laboratory, with the participation of McDonnell Douglas Astronautics Co., General Atomic Co., and Ralf M Parsons Co.

(b) TITAN was a collaborative study organised by the University of California at Los Angeles, and including General Atomic Tech., Rensselaer Polytechnic Inst., the Fusion Engineering Design Centre at Oak Ridge, and Culham Laboratory. The net electrical power appears as 1000 MWe in some references and 970 MWe in others; the former value is used in this study.

(c) Field measured at plasma major axis.

(d) Field measured at plasma surface.

(e) Length of central cell of Linear reactor.

(f) Mass power density is the ratio of the net electrical power and the mass of the fusion power core. The masses are taken from Tables A2 to A7, except that in the case of CRFPR and TITAN allowance has been made for breeder/coolant in external circuits. These values of the mass power density are roughly consistent with values quoted in the literature, except that values for CRFPR and TITAN are frequently calculated without any breeder/coolant.

COMPONENT	MATERIALS	MASS (tonne)	VOLUME (m^3)	REPLACEMENT PERIOD Calendar years	WASTE MASS (te/GWe)	WASTE VOLUME (m^3/GWe)	WASTE CATEGORY UK (a)
First wall	PCA (b)	28	3.6	6	160	20	ILW
Second wall	PCA	20	2.6	6	112	14	ILW
Pumped limiters	Ta-5W	31	1.9	6	172	10	non-active (k)
Blanket: Structure	PCA	356	45.1	6	1980	250	ILW
Multiplier	Zr_5Pb_3	328	36.9	20	574	62	ILW
Breeder	$LiAlO_2$	606	178.4	20	1010	297	ILW
Reflector	79%C 21%PCA	208	109.8	20	347	183	ILW
Shield	(c)	8664	1332.9	40	7220	1110	LLW (d)
Magnet dewar	Fe - 1422	1044	132.2	40	872	110	LLW (e)
Magnet He vessel	AISI 304	1119	141.6	40	932	118	LLW
TF Coils	(f)	4075	522.4	40	3400	432	LLW (g)
Central support cylinder	Glass/epoxy	281	147.9	40	236	123	non-active
OH and EF Coils	(h)	375	54.3	40	312	45	non-active
Vacuum Pump shield	(l)	6536	1054.2	40	5440	878	LLW (j)
Heating & Current Drive		116	14.9	40	96	12	LLW (m)
FUSION POWER CORE TOTAL		23790	3779		22800	3660	
		TOTAL ILW			4200	830	
		TOTAL LLW			7410	1110	

Table A2 Masses, volumes and Waste generated by STARFIRE

Notes for Table A2 Masses, Volumes and Waste generated by STARFIRE

(a) Waste categories are based on 50 years post-irradiation cooling time. It is assumed that this is sufficient for the afterheat to decay sufficiently to allow for further storage/disposal to be uncooled. Note that in the STARFIRE study a 5 year irradiation time was assumed for all components. This is inconsistent with the cycle time for the shields and magnets which is 30 full power years. Thus the activation for these components has been scaled up by a factor of 6 to obtain the waste categories for this study. This linear scaling on fluence is valid for single stage activation products but gives an underestimate for multi-stage ones.

(b) PCA is the Primary Candidate Alloy and is a stainless steel (AISI316) based on 65% Fe 16% Ni 14% Cr 2% Mo 2% Mn

(c) 68.6% Fe1422 (Fe 14% Mn 2% Ni 2% Cr), 15.1% TiH_2, 9.7% W, 5.2% B_4C + 1.4% Ti6Al4V.

(d) The LLW category is based on the average activity (6.9 GBq/tonne). The individual component activities range from $0 \rightarrow 2.1$ TBq/tonne and segregating the shield would give 24 tonne/GWe ILW, 3360 tonne/GWe LLW and 3840 tonne/GWe non-active.

(e) If the dewar is segregated into inboard and outboard components the waste becomes 60 tonne/GWe LLW and 810 tonne/GWe non-active.

(f) 39.2%Cu, 56.7% AISI304, 1.25% Nb_3Sn, 1.5% insulation (G10), 1.4%NbTi

(g) Note that 73% of the TFC activity is concentrated in ~ 60 tonnes (~ 15% total mass) of copper on the inboard side. Segregation of components could give 3000 tonne/GWe non-active plus 390 tonne/GWe LLW.

(h) 5.1% AISI304, 27.5%Cu, 5.3% glass/epoxy, 2.0%NbTi

(j) Specific activity of the vacuum pumping shield is not given in the STARFIRE report, and therefore the waste category is assumed to be same as for the bulk shield, as are segregation effects.

(k) The activity quoted for Ta-5W (~10^{-22}MCi/m^3 (2.3 x 10^{-7}Bq/tonne) 30 years after irradiation to 18 MWyr/m^2) seems to be too low. However the quantity of waste concerned is very low.

(l) 87.7% Fe 1422, 6.7% B_4C, 5.1%TiH_2 4.6%Ti6Al4V.

(m) Not included in the STARFIRE report, so LLW category assumed

COMPONENT	MATERIALS	MASS (tonne)	VOLUME (m^3)	REPLACEMENT PERIOD Calendar yrs (f)	WASTE MASS (te/GWe)	WASTE VOLUME (m^3/GWe)	WASTE CATEGORY (g)
First wall(a):							
Tiles	Tungsten	28	1. 5	2	254	14	repository
Wall	Copper alloy	12	1. 5	5	44	5	repository
Cooling tubes	Inconel	6	0. 7	2	54	6	repository
Blanket(a):							
Structure	S/S FV448	570	74. 0	5	2130	277	repository
Breeder	Li_2SiO_3	650	250. 0	20	590	227	repository
Multiplier	Be	110	62. 0	20	100	56	repository
Divertor(b)	W/26Re+Sn	1	0. 1	2	10	1	repository
Shield(c)	Fe1422 + AISI 316 + B_4C+H_2O+He+Pb	21000	3200. 0	20	19100	2910	(g)
Toroidal field coils (d)	S/S 316L + Cu + CuNi + NbTi/Nb_3Sn+He +epoxy etc	6400	1000. 0	20	5820	909	non-active
Poloidal field coils (e)	As for TF coils	2400	400. 0	20	2180	364	non-active
Neutral beam heating		460	68. 0	20	418	62	repository
TOTALS		31637	5058		30700	4830	
		TOTAL REPOSITORY WASTE			12400	1990	

Table A3 Masses, volumes and waste generated by DEMO R254

Notes for Table A3 Masses, Volumes and Waste generated by DEMO R254

(a) Volumes of first wall and blanket components can be deduced from the DEMO report, from the known first wall area (492 m^2) and relative volumes as assumed for the neutronics calculations. The following densities (in te/m^3) have been assumed: tungsten or W-26Re 19.3 [1] copper 8.25 [1], inconel 8.42 [2], stainless steel FV448 as for FV1.4914 [1], beryllium 1.85 [1], lithium metasilicate 2.6 [3]. The figures for the blanket structure include the coolant manifold as in [4].

(b) The divertor volume is also deduced from the DEMO report, assuming a total cooled (poloidal) length of plate of 1.4 m. Density of liquid tin at 650°C is taken as 6.7 te/m^3 [4], that of the W-26Re as above.

(c) The shield is taken to be a scaled up version of the INTOR shield [5], with the same composition as the INTOR outboard shield reference design with an average density of 6.7 te/m^3 based on densities (in te/m^3) of 7.94 for Fe1422, 2.52 for B_4C, and 11.35 for lead [6]. The INTOR shield volume of 1650 m^3 deduced from [5] is scaled up for DEMO (i) in the ratio (fusion power/neutron wall loading) to allow for area effects and (ii) by 20% as a nominal degree of thickening required by neutronic considerations.

(d) The mass of the TF coils is found by assuming the same density as for the INTOR TF coils, which have a total mass of about 3100 te [5] and by taking the volume to be scaled in proportion to $(\text{area})^{3/2}$ or $(\text{fusion power/neutron wall loading})^{3/2}$. The overall density is taken to be the same as for the PF coils, for which a figure of 6.17te/m^3 can be deduced from the DEMO report.

(e) From the dimensions given in the DEMO report the total PF coil volume can be deduced directly. The mass then follows from the density as noted in (d) above.

(f) It is assumed for the first wall tiles and cooling tubes, and for the divertors, that the high heat flux and plasma bombardment mean replacement of these items with the maintenance cycle. In the case of the high strength copper-alloy first wall the radiation damage life has been taken as 10 MWy/m^2, which for conservatism is half the figure that has been suggested [7]. For blanket structural steel the fluence limit is taken as 30 MWy/m^2, twice the figure appropriate for the first wall [8]. It is assumed that breeder and multiplier can be recycled with negligibly small loss of material (eg burn-up of ^6Li) and therefore last for the design life of the reactor. The shield and coils are also assumed to last for this time.

(g) Activation data are not currently available for the DEMO reactors, R254 and R278. It has therefore been assumed by analogy with STARFIRE that the magnets can be classed as non-active waste, that the shields emerge as a mixture of non-active and repository waste in the ratio 0.54:0.46, and that components within the shield are repository waste.

COMPONENT	MATERIALS	MASS (tonne)	VOLUME (m^3) (d)	REPLACEMENT PERIOD Calendar year	WASTE MASS (te/GWe)	WASTE VOLUME (m^3/GWe)	WASTE CATEGORY (f)
First wall: Structure	Steel	232	44	6. 25 (e)	742	140	HLW
Multiplier	Pb	73	(8)	6. 25 (e)	234	(6)	HLW
Coolant	H_2O	7					
Blanket: Structure (torus)	Steel	777	844	6. 25 (e)	2490	2700	HLW
Structure (flange)	Steel	1560	705	6. 25 (e)	4990	2260	MLW
Breeder	$Li_{17}Pb_{83}$	4660	496	25	3730	390	MLW
Coolant	H_2O	153					
Divertor		37	5	1. 25 (e)	590	78	HLW
Shield: Structure	Steel	19285	3090	25	15428	2470	HLW
Coolant	H_2O	649					
Toroidal field coils		8620	1300	25	6900	1040	
Poloidal field coils		961	145	25	770	116	
Coil support structures		2660 (a)	337	25	2130	270	
Other support structures		2937 (b)	372	25	2350	300	
Cryostat		2200	278	25	1760	223	
TOTALS		44800 (c)	7620		42200	9990	
	HIGH LEVEL UNCOMPACTED WASTE (g)				21400	5400	
	MEDIUM LEVEL UNCOMPACTED WASTE (g)				7150	2650	

Table A4 Masses,.volumes and waste generated by PCSR-E

Notes for Table A4 Masses, Volumes and Waste generated by PCSR-E

(a) Consisting of intercoil support structure (2520 te) and inner support structure (138 te).

(b) Consisting of coil support leg (192 te), torus support leg (425 te), ring beam support (580 te), and main leg mass (1740 te).

(c) The mass of 44800 tonnes includes coolant, whereas the mass of 44000 tonnes quoted in Reference [9] excludes coolant.

(d) Where individual volumes are not quoted in the literature they are deduced from the masses, densities, and overall volumes.

(e) Averaged over reactor life, since availability varies during life.

(f) Based on information in Reference [10].

(g) The quoted volumes are of uncompacted components and in some cases, such as the blanket structure, contain substantial voids. High level waste has an activity > 370 TBq/m^3 and medium level waste has an activity > 370 GBq/m^3 [10].

COMPONENT	MATERIALS	MASS (tonne) (c)	VOLUME (m^3)	REPLACEMENT PERIOD Calendar years	WASTE MASS (te/GWe)	WASTE VOLUME (m^3/GWe)	WASTE CATEGORY (U.S)
First wall	Cu (MZC)	2	0.2	1	60	6	Geologic
Second wall	HT-9	10	1.2	1	300	36	Geologic
Limiter {collector	Cu (MZC)	3	0.3	1	90	9	Geologic
{manifold & header	HT-9	6	0.7	1	180	21	Geologic (a)
Blanket: Structure	HT-9	40	5.2	1	1200	156	(b)
Breeder/Multiplier	PbLi	5460 (f)	535.0	30	5460	535	(e)
Shield	AISI 316	160	20.3	30 (d)	159	20	Class C
Toroidal field coil	} Cu/AISI316/	76	11.5	30 (d)	75	11	Class C
Ohmic heating coil	} MgO/H_2O	400	54.1	30	399	54	Class C
Equilibrium field coil	}	413	55.8	30	414	57	Class C
Pellet injector (g)	(h)	1	0.1	30 (i)	0.0	0.0	
SUB TOTAL (FPC)		6571	684		8340	906	
		TOTAL GEOLOGIC WASTE			7290	762	
		TOTAL CLASS C WASTE			1050	144	

Table A5 Masses, Volumes and Waste for CRFPR

Notes for Table A5 Masses, Volumes and Waste generated by CRFPR

(a) This is classed as geologic rather than class C waste due to the Nb^{94} limit which may be overly conservative.
(b) The blanket structure will meet class C rating if filled with concrete, but assumed geologic waste here.
(c) Masses do not include water coolant.
(d) The first wall, blanket, shield and TFC's are replaced every year but the shield and TFC's are recycled via the spare torus assembly.
(e) Assumed to be geologic waste after reprocessing for 30 years.
(f) This includes ~4500 tonnes of PbLi in the external circuits [11].
(g) Successful development of a centrifugal injector must be assumed.
(h) Stainless steel ~ 1 tonne, copper ~ few kgs, carbon fibre ~ 1kg.
(i) Turbine may need more frequent replacement but this is very small.

COMPONENT	MATERIALS	MASS (tonne)	VOLUME (m^3)	REPLACEMENT PERIOD Calendar years	WASTE MASS (te/GWe)	WASTE VOLUME (te/GWe)	WASTE CATEGORY (U.S)
First wall	V-3Ti-1Si	2.h	0. 4	1. 3	76	12	Class C
Divertor collector plates	W(a)	0.4	0. 04	1. 3	12	1	geologic
Blanket: Structure	V-3Ti-1Si	39.2	6. 4	1. 3	1200	197	Class C
Coolant breeder	Liquid Li	212.0 (b)	400	40	212	400	Class C (c)
Reflector structure	V-3Ti-1Si	95.6	15. 5	6. 7	572	94	Class C
Hot shield structure	V-3Ti-1Si	172	28. 0	6. 7	1030	168	Class C
Toroidal field coils	Part of blanket	-	-				
Ohmic heating coils	(d)	289	34. 2	40	289	34	Class C
Equilibrium field coils		315	43. 0	40	315	43	Class C
EF Coil shield	steel/B_4C	394	62. 7	40	394	62	Class C
Divertor: Shield 1	V-3Ti-1Si	14.2	2. 3	1. 3	436	71	Class C
Shield 2	V-3Ti-1Si	41.2	6. 7	6. 7	244	40	Class C
Pellet injector (e)	(f)	1.0	0. 1	40 (g)	0	0	
SUB TOTAL (FPC)		1576	600		4800	1124	
		TOTAL GEOLOGIC WASTE			224	1	
		TOTAL CLASS C WASTE			4580	1123	

Table A6 Masses, Volumes and Waste for TITAN

Notes for Table A6 Masses, Volumes and Waste generated by TITAN.

(a) W-Re alloy may be preferred but waste category unaffected.

(b) The 212 te includes ~ 34 te of lithium in the reflector and hot shield and ~178 te in the external circuits.

(c) Assumed to be class C waste after reprocessing for 40 years.

(d) OHC coil material 70% Cu 10% insulator (spinal) 10% structure (304 s/s) 10% coolant (He)

(e) Successful development of a centrifugal injector must be assumed.

(f) Stainless steel ~1 tonne, copper ~ few kgs, carbon fibre ~ 1 kg.

(g) Turbine may need more frequent replacement but this is very small.

COMPONENT	MATERIALS	MASS te	VOLUME m^3	REPLACEMENT PERIOD Calender years	WASTE MASS te/GWe	WASTE VOLUME m^3/GWe	WASTE CATEGORY UK (d)
Blanket							
Structure	HT-9	1089	137. 8	4. 3	4982	630	ILW
Breeder/Multiplier	$Li_{17} Pb_{83}$	8387	882. 8	30. 0	5482	577	ILW
Reflector	HT-9	4275	541. 1	30. 0	2794	354	ILW
Shield	77% A-242 Steel, 19% lead, 4% B_4C	17886	2338. 7	30. 0	11690	1529	ILW
Central Cell Coils	81% Al-2219, 17% Al, 2% NbTi	3399	1246. 5	30. 0	2222	815	LLW
Barrier Coils	(a)	1777	210. 2	30. 0	1161	137	non-active
Transition and Yin Yang Coils	62% Cu, 37% 316 SS, 1% NbTi	2923	345. 2	30. 0	1910	226	non-active
Others (b)	(c)	984	175. 1	30. 0	643	114	non-active
FUSION POWER CORE TOTAL		40720	5877		30884	4382	
		TOTAL ILW			24948	3090	
		TOTAL LLW			2222	815	

Table A7 Masses Volume and Waste Generated for WITAMIR-I

Notes on Table A7 Masses Volume and Waste Generated for WITAMIR-I

62% Cu, 35% 316 SS, 1.2% NbTiTa, 0.9 Nb_3Sn, 0.5 NbTi

(a) Direct convertor including cryopump, Neutral beams, ECRH and support structures.

(b) 20% HT-9, 6% 316 SS, 22% Al-2219, 6% Cu, 1.6% Ta10W, 0.1% LaB_6, 44% A-242, 0.8% Mo, 0.5% W.

(c) Waste categories are based on the assumption that material outside the central cell coils is non-active and the

(d) activities deduced from the WITAMIR-I report i.e. blanket = 2.5 TBq/te, reflector plus shield = 0.68 TBq/te and central cell coils = 1.0 MBq/te. The cool-down time in 50 years and the results for the reflector plus shield and the central cell coils have been scaled up by 15 since only a two year irradiation time was assumed.

	TOTAL WASTE (Net)		NON-ACTIVE WASTE (net)		REPOSITORY WASTE DIRECT DISPOSAL (net)		REPOSITORY WASTE DIRECT DISPOSAL (packaged)(a)
REACTOR SYSTEM	MASS te/GWe	VOLUME m^3/GWe	MASS te/GWe	VOLUME m^3/GWe	MASS te/GWe	VOLUME m^3/GWe	VOLUME m^3/GWe
STARFIRE	22800	3660	11200	1720	11600	1940	10400
DEMO R254	30700	4830	18300	2840	12400	1990	10700
PCSR-E	42200	9990	13600	1940	28600	8050	43400
CRFPR	8340	906	0	0	8340	906	4870
TITAN	4800	1120	0	0	4800	1120	6030
WITAMIR	30900	4380	3710	480	27200	3900	21000

Table A8 WASTE VOLUMES

Table A8 Waste totals

(a) The waste is assumed to be packed into similar boxes, at a uniform volume packing factor, which are then with grouting. Thus the ratio of the volumes of packaged to net waste is $V_{BE}/(p.V_{BI})$, where V_{BE} = box e: volume, V_{BI} = box internal volume, and p = packing factor. For the purpose of this study the parameters are on the NIREX Number 2 LLW box for which $V_{BE} = 8.64\ m^3$, and $V_{BI} = 5.36\ m^3$, and a value of $p = 0.3$ is assumed.

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