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THE IMPACT OF D–^{3}He FUSION REACTORS ON WASTE DISPOSAL

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The suggestion that the surface of the moon may be mined for ^{3}He to be used as a fuel in terrestrial fusion reactors has recently been made. A fusion reactor based on the D–^{3}He reaction would have the advantage that most of the power produced would be in the form of charged non-radioactive particles. However, secondary D–D and D–T reactions also occur. A study is made of the consequences of the radioactivity induced by the neutrons from these reactions with respect to waste disposal. A generic first wall and shield 0.4 m thick consisting of 7% structure, 73% H_2O and 20% void was used as a test case. The structural materials considered were two austenitic stainless steels (PCA and Tenelon), two ferritic alloys (HT-9 and a low activity modification of HT-9), and a vanadium alloy (V15Cr5Ti). The results of the calculations show that for operation at a fusion power loading of 1 MW/m^2 for a thirty year reactor lifetime, Tenelon, the low activity HT-9 and the vanadium alloy meet surface waste disposal requirements consistent with those published in the U.S. Code of Federal Regulations (10CRF61). If five percent boron is added to the water to suppress (n, γ) reactions, HT-9 and PCA structures are acceptable. Calculations of a first wall and shield designed for a 600 MW reactor D–^{3}He operating at a fusion power loading of 2.94 MW/m^2 indicate that Tenelon may be used as structure and meet waste disposal requirements after thirty years of operation. It is concluded that the use of a D–^{3}He cycle allows surface burial of activated reactor components and results in a significant reduction in the volume of waste.

1. Introduction

The focus of most of the development of plasma research into a practical device to supply electric power from fusion has been on the deuterium–tritium (D–T) reaction. The reasons for this are well known and include the availability of the fuel, deuterium from its natural abundance in water and tritium from the absorption of neutrons in lithium, the relatively large amount of energy produced in the reaction, namely 17.6 MeV, and that the confinement of a self-sustaining D–T plasma is likely to be the easier to attain than that of other energy producing reactions. The disadvantages are also well known. Most of the energy of the reaction appears as the kinetic energy of 14.1 MeV neutrons. To make use of this energy it must be converted into heat in a blanket surrounding the plasma and this thermal energy used to drive a Rankine or Brayton cycle. The blanket must also contain lithium to continue to breed replacement fuel. High energy neutrons also result in radiation damage in the structure of the blanket and induce nuclear reactions which leave the blanket radioactive. The accomodation of all these factors make blanket design a difficult task and much of the effort in fusion design work has centered around how best to engineer a structure that will absorb this energy in a useful way yet have an acceptable lifetime and minimal induced activity.

Since the blanket structure has a finite lifetime in the reactor it must be periodically replaced and the used radioactive blanket sections discarded. With proper selection of materials it has been determined that these blanket sections may be disposed of in near surface burial consistent with the guidelines used for the disposal of fission reactor waste as reflected in the Nuclear Regulatory Commission rules. For example, in the MINIMARS [1] reactor design study it was found that by using a low activity version of the ferritic alloy HT-9, blanket segments operated for their expected lifetime of 5 years could be disposed of in near surface burial. It was also found that, as designed, the components of the blanket reflector which would be expected to last for the life of the reactor could not be disposed of in the same fashion.

While most of the effort has been on the D–T system there has been continuing interest in other fu-

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sion reactions. Of these the D–^{3}He reaction has some rather desirable properties. The principal one being that in the reaction $D + {}^3He \to {}^4He + p + 18.3$ MeV, the energy appears as kinetic energy of the charged particles. This opens the possibility of utilizing direct conversion with an attendant increase in the efficiency of conversion to electricity. In addition none of the products of this reaction are radioactive. While a higher temperature is required to contain the D–^{3}He plasma than the D–T plasma, the primary consideration limiting development of the D–^{3}He concept has been the lack of a source of ^{3}He of sufficient magnitude for anything but demonstration experiments.

The work of Wittenberg, Santarius and Kulcinski [2] has pointed out that the soil of the moon has been accumulating ^{3}He from the solar wind. They state that the Apollo lunar samples indicate that the surface soils of the moon contain $\sim 10^9$ kg of ^{3}He. If this amount of ^{3}He were to be used in a 50% efficient D–^{3}He reactor, it would provide $\sim 10^7$ GW (electric) yr of electric power. They estimate that when the efforts to mine the fuel and transport it to terrestrial power plants are considered, the energy payback ratio is ~ 250.

These preliminary results must be verified by further work and detailed schemes for implementing the various steps in the process of bringing ^{3}He from the moon to earth must be developed to determine their practicality. However, with the prospect of a source of fuel for reactors based on the D–^{3}He reaction, it would seem reasonable to look at some of the environmental effects of this type of reactor, in particular to investigate the long term waste problems that would be present.

2. Neutron production

Although the principal reaction in the plasma is the D–^{3}He reaction, there are other reactions taking place. Because deuterium is present there will also be D–D reactions. There are two branches of this reaction that occur with about equal probability:

$$D + D \to T\ (1\ \text{MeV}) + p\ (3\ \text{MeV}),$$

$$D + D \to {}^3He\ (0.8\ \text{MeV}) + n\ (2.5\ \text{MeV}).$$

In the first reaction tritium is produced, which is radioactive, and in the second a neutron is produced which, upon absorption in the structure of the reactor, could induce radioactivity. The tritium produced in the first reaction may not be immediately removed from the plasma and could interact via the D–T reaction to produce 14.1 MeV neutrons which will also induce radioactivity. These two secondary reactions are source terms for most of the radioactivity produced in the structure of the reactor.

The relative contribution to the power from the neutron producing fusion reactions may be calculated from a simple one temperature model. The total fusion power can be written as:

$$P_{Fus} = n_D n_{He} \langle \sigma v \rangle_{DHe} E_{DHe} V + \frac{n_D^2}{2} \langle \sigma v \rangle_{DD}^{n} E_{DDn}^{tot} V + \frac{n_D^2}{2} \langle \sigma v \rangle_{DD}^{T} \left(E_{DDT}^{tot} + f_B E_{DT} \right) V,$$

where P_{Fus} is the total fusion power, n_D is the deuterium density, n_{He} is the helium-3 density, E_i the energy produced in the ith reaction, $\langle \sigma v \rangle_i$ the reaction rate for the ith reaction, f_B the fraction of the tritium that reacts, and V the volume of the plasma. Similarly the neutron power is:
From DD reactions,

$$P_{DD}^{n} = \frac{n_D^2}{2} \langle \sigma v \rangle_{DD}^{n} E_{DD}^{n} V$$

and from DT reactions,

$$P_{DT}^{n} = f_B \frac{n_D^2}{2} \langle \sigma v \rangle_{DD}^{T} E_{DT}^{n} V.$$

Combining these relations

$$\frac{P_n}{P_{Fus}} = \left\{ \langle \sigma v \rangle_{DD}^{n} E_{DD}^{n} + f_B \langle \sigma v \rangle_{DD}^{T} E_{DT}^{n} \right\} \times \left\{ 2 \frac{n_{He}}{n_D} \langle \sigma v \rangle_{DHe} E_{DHe} + \langle \sigma v \rangle_{DD}^{n} E_{DDn}^{tot} + \langle \sigma v \rangle_{DD}^{T} \left(E_{DDT}^{tot} + f_B E_{DT} \right) \right\}^{-1}.$$

These equations show that fraction of the fusion power carried by neutrons depends on the temperature of the plasma, the ratio of ^{3}He to D, and the burn fraction of the tritium. Figs. 1 and 2 show the percent of fusion power carried by neutrons for 100% and 50% tritium burnup. In each case the amount of power in neutrons is reduced as the temperature of the plasma is raised. This behavior is a result of the nature of the cross sections of the various reactions. The reaction rate per interacting pair for D–T decreases after peaking at 60 keV while the reaction rate for ^{3}He–D increases faster than that for the D–D reactions as the plasma temperature is increased. In both cases the fraction is reduced as the ^{3}He/D ratio is increased reflecting the reduced

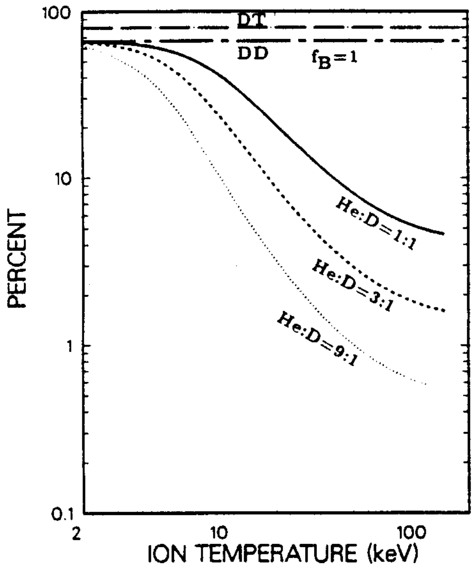

Fig. 1. Percent of fusion power in neutrons for 100% tritium burnup as a function of ion temperature for ^{3}He : D ratios of 1 : 1, 3 : 1 and 9 : 1.

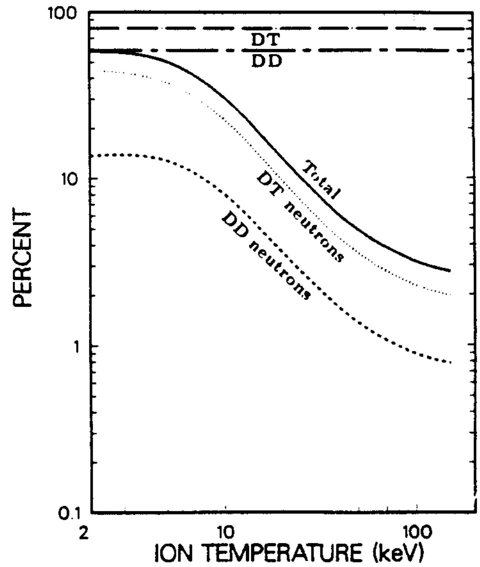

Fig. 3. Percent of fusion power in DD and DT neutrons for 50% tritium burnup as a function of ion temperature for a ^{3}He : D ratio of 1 : 1.

likelihood of D–D reactions. As is expected, the fraction of energy in neutrons is increased if all the tritium is assumed to burn up.

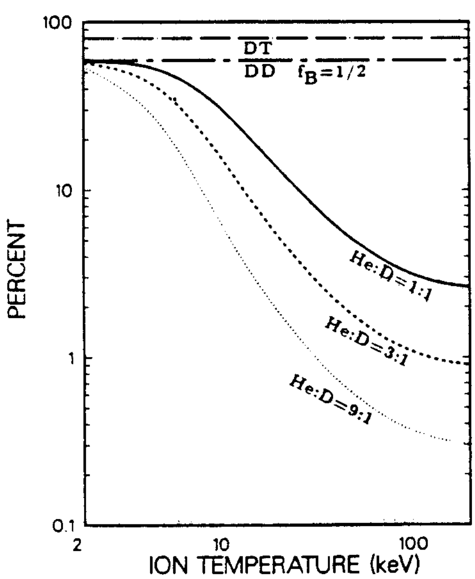

Fig. 2. Percent of fusion power in neutrons for 50% tritium burnup as a function of ion temperature for ^{3}He : D ratios of 1 : 1, 3 : 1 and 9 : 1.

To use these results a set of operating conditions must be chosen. If other plasma related considerations, such as power density and stability, are taken into account, reasonable operating conditions may be chosen [3]. These considerations indicate that a ^{3}He–D ratio of 1 : 1 is preferred with a 50% tritium burnup being a reasonable choice. The operating temperature would be in the range of 50–60 keV for a tokamak and 90–110 keV for a tandem mirror. This corresponds to about 4.5% of the power in neutrons for a tokamak and about 3.2% in a tandem mirror. Increasing the ^{3}He : D ratio to 3 would lower the values by about a factor of 3. Although the latter option would lead to reduced activity it is not as desirable from a reactor standpoint because of a significant reduction in the power density of the plasma and a consequent increase in size and cost.

The relative contribution of DT and DD neutrons is shown in fig. 3 for the 1 : 1 case with 50% tritium burnup. At the operating temperatures the power from DT neutrons is approximately three times the power from DD neutrons. Because of the difference in the energy carried per neutron in each case, there are almost twice as many D–D neutrons as D–T neutrons. Fig. 4 illustrates that similar remarks may be made in the 3 : 1 ^{3}He to D case.

An additional potential source of radioactivity is the 14.3 MeV protons produced in the primary D–^{3}He

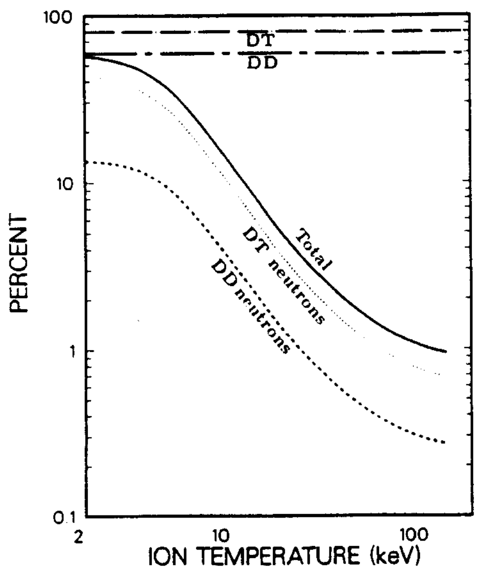

Fig. 4. Percent of fusion power in DD and DT neutrons for 50% tritium burnup as a function of ion temperature for a ^{3}He : D ratio of 3 : 1.

reaction. If these protons strike structural elements of the reactor, nuclear reactions may be induced leading to radioactive products. If the reaction results in a neutron being emitted, it in turn may interact and create additional radioactive nuclei. If this type of reactor is to be successful, most of the protons must be directed out of the plasma and into the direct convertor where they will lose their energy in the conversion process. Almost all of the remaining protons will lose their energy in the plasma through collisions with electrons. There is only a small fraction of the protons that would strike the reactor structure. Some insight into the magnitude of this process may be obtained by considering the previous case of a 1 : 1 mixture of ^{3}He to D with 50% tritium burnup. At a plasma temperature of 50 keV about 4.5% of the power is carried off by the neutrons. For this case the ratio of the proton production rate to the total neutron production rate is 7.3.

If a fraction, F, of these protons strike structural elements of the reactor producing Y neutrons per proton through a (p, n) or (p, np) reaction, the number of neutrons from protons induced reactions to neutrons from fusion reaction is

$$\frac{N_{p,n}}{N_{f,n}} = 7.3\ FY.$$

Borchers, Overley, and Wood [4] have measured thick target neutron yields for a number of elements at proton energies up to 13 MeV. An extrapolation of their data to 14 MeV gives a maximum value for Y of about 1×10^{-3} for medium Z nuclei. If a value of 1×10^{-2} is taken as a reasonable upper limit for F then

$$\frac{N_{p,n}}{N_{f,n}} = 7.3 \times 10^{-2} \times 10^{-3} = 7 \times 10^{-5},$$

that is, neutron production from the protons is over four orders of magnitude lower than that from fusion reactions. In view of this result, only neutrons from fusion reactions will be considered here. This result also implies that at least for the purposes of this paper the radioactive products of the proton induced reaction can also be neglected. Since the range of protons in solids is small, there may be regions where the local activity is affected by proton bombardment but this should not affect the total activity in the reactor. An investigation of this aspect requires a more detailed reactor design and plasma analysis than is presently available.

3. Reference case

For the purposes of evaluating the waste disposal consequences the 1 : 1 case with 50% tritium burnup is used as the reference for both the tokamak and tandem mirror cases. The reactor is modeled as a cylinder with a first wall-shield region extending from 0.5 m to 0.9 m. This region contains 7% structure, 73% H_2O and 20% void. A second region 0.6 m thick consisting mostly of structure (90%) surrounds the first wall-shield region to provide a more realistic boundary condition at the outside of the first wall-shield. The thickness and composition of the first wall-shield was not optimized but was picked on the basis of experience with D–T systems as a reasonable starting point for comparison purposes.

To be able to make comparisons with different types of plasmas the calculations are normalized to a fusion *power* loading of 1 MW/m^2 of first wall area. This power loading does not correspond to actual power striking the first wall but is representative of the fusion power produced in the volume enclosed by the first wall. In the case of a D–T reactor the equivalent neutron wall loading would be 14.1/17.6 = 0.8 MW/m^2. Since radiation damage effects will greatly be reduced in D–^{3}He systems the calculations are made considering that the first wall-shield will last for the full life of the reactor which is taken to be thirty years. The use of only one blanket shield assembly for the full life of the reactor has the additional advantage of reducing

Table 1
Elemental composition used for BCSS activation analysis

Element		Structural materials				
		PCA (wt%)	V15Cr5Ti (wt%)	HT-9 (wt%)	Modified HT-9 (wt%)	Tenelon Fe 1518 (wt%)
1	H					
2	Li					
4	Be					
5	B	0.005		0.01	0.001	0.001
6	C	0.005	0.005	0.2	0.15	0.15
7	N	0.01	0.01	0.05	0.001	0.005
8	O		0.01	0.01	0.007	0.007
9	F					
11	Na					
12	Mg					
13	Al	0.03	0.02	0.01	0.008	0.008
14	Si	0.5	0.03	0.35	0.2	0.2
15	P	0.01	0.003	0.02	0.013	0.013
16	S	0.005	0.001	0.02	0.004	0.004
17	Cl		0.0001			
19	K	0.0003	0.00001	0.0003	0.0003	0.0003
20	Ca					
22	Ti	0.30	5.0	0.09	0.1	0.003
23	V	0.10	79.80	0.3	0.3	0.002
24	Cr	14.0	15.0	12.0	11.0	15.0
25	Mn	2.0		0.55	0.53	15.0
26	Fe	64.88	0.004	85.0	Balance	Balance
27	Co	0.03		0.02	0.005	0.005
28	Ni	16.0	0.0004	0.5	0.006	0.004
29	Cu	0.02	0.0002	0.09	0.003	0.003
30	Zn					
33	As	0,02	0.0002			
38	Sr					
40	Zr	0.005		0.001	0.001	0.001
41	Nb	0.03	0.004	0.11	0.00011	0.00011
42	Mo	2.0	0.001	1.0	0.00027	0.00027
47	Ag	0.0001		0.0001	0.00009	0.00009
48	Cd	0.0002		0.0001	0.0001	0.0001
50	Sn	0.005		0.003	0.003	0.003
51	Sb	0.001		0.001	0.0005	0.0005
56	Ba	0.001		0.001	0.0002	0.0002
65	Tb	0.001		0.0005	0.0002	0.0002
73	Ta	0.01	0.001	0.001	0.0004	0.0004
74	W	0.05	0.0002	0.5	2.50	0.01
77	Tr	0.001		0.0005	0.0002	0.0002
82	Pb	0.001		0.001	0.0005	0.0005
83	Bi	0.001		0.001	0.0002	0.0002

the total amount of material to be stored. A decay time of one year is assumed prior to disposal for all cases.

The calculations were made using the DKR [5] code package with activation cross sections taken from the ACTL [6] library. The waste disposal limits are those calculated by Fetter [7]. Five different alloys are considered as structure materials. The materials, which are taken from the Blanket Comparison and Selection Study (BCSS) report [8] as likely candidates for fusion reactors, are the ferritic alloy HT-9, an austenitic stainless steel PCA, a modification of HT-9 which would have lower long term activity, a high manganese austenitic stainless steel known as Tenelon, and a vanadium alloy V15Cr5Ti. The composition of these materials is that presented in the BCSS report and is given in table 1.

4. Results

The level of activity which would be expected in this first wall-shield is shown in fig. 5 for the case of a tokamak with equal amounts of D and ^{3}He and 50% tritium burnup operating for thirty years at full power. In general the short and medium term behavior is governed by activity from the major constituents while the very long term activity is due to trace impurities. Since waste disposal characteristics are determined by long lived activities it is clear that Tenelon, modified

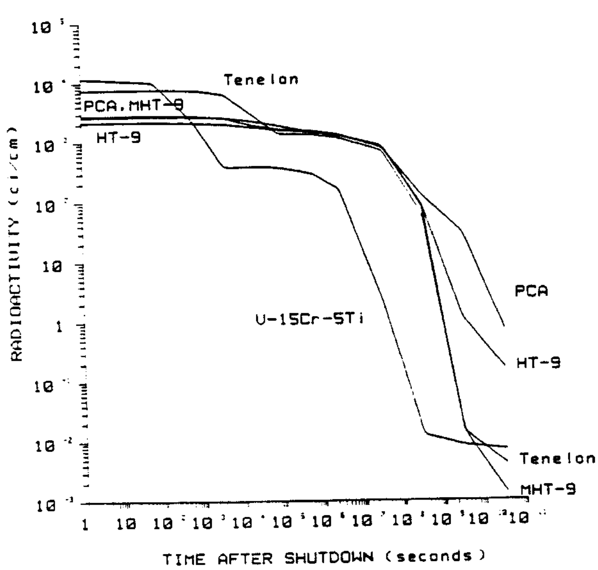

Fig. 5. Activity per centimeter after shutdown of the first wall-shield for the tokamak case. ^{3}He:D ratios of 1:1, 30 years operation at a fusion power loading of 1 MW/m^2.

HT-9, and the vanadium alloy would be expected to have superior disposal properties. However, both Tenelon and V15Cr5Ti show significantly higher initial activity implying a possible higher initial afterheat. Since afterheat is also a problem in certain accident scenarios, this aspect cannot be ignored in making a choice of material to be used.

The waste disposal ratings for the reference case are given in table 2 for the five different structural materials. The waste disposal rating (WDR) [9] is defined as the sum of the ratio of the concentration of a particular isotope to the maximum allowed concentration of that isotope taken over all isotopes. Thus a WDR less than one represents an permissible situation. The distinction between the two modes of disposal, class A and class C, is that of 10CFR61 and basically means burial with no restriction or burial with restriction on form and with a 500 year barrier. Although a class A WDR is preferred, the classification of waste as class C would be satisfactory for these reactor components.

As indicated in the table only the vanadium alloy easily meets class A limits. Modified HT-9 and Tenelon also meet the limits but only by a factor of about two. Since reactors of this type may be expected to operate at a power loading of 2–3 MW/m^2 it is unlikely that the goal of a WDR < 1 would be met. If class C limits are considered then modified HT-9 and Tenelon are acceptable. In all cases one isotope is the major contributor to the limit. In the case of HT-9, PCA and V15Cr5Ti it is ^{94}Nb ($t_{1/2} = 2 \times 10^4$ a) produced from ^{93}Nb or ^{94}Mo. In the case of modified HT-9 and Tenelon it is ^{108m}Ag ($t_{1/2} = 1.3 \times 10^2$ a) produced from ^{107}Ag. Principal secondary contributors are ^{99}Tc ($t_{1/2} = 2.1 \times 10^5$ a) from ^{98}Mo in PCA (class A) and ^{94}Nb in modified HT-9 and Tenelon (class C).

In performing the previous calculations of the WDR the activities used were averaged over the first wall-shield region. Alternatively one could consider averaging over the entire region enclosed by the outer rim of the first wall-shield, i.e. average over a volume including the plasma region in the center. The results of this calculation are shown in table 3. The effect is to lower the WDR by about 30% and to bring both modified HT-9 and Tenelon into consideration for class A disposal. HT-9 and PCA remain unacceptable for either class A or class C.

At the other extreme it might well be desirable for the purposes of reducing the volume for ease of transportation to compact the first wall-shield. The WDR for these cases are also shown in table 3. The effect is to increase the WDR by a factor of approximately 14. Modified HT-9, Tenelon and the vanadium alloy are

Table 2
Comparison of waste disposal rating (WDR) 1 year after shutdown
1st wall-shield
[Tokamak 1:1 ^{3}He:D (1 MW/m^2 fusion power loading, 30 years operation)]

	HT-9	PCA	MHT9	Tenelon	V15Cr5Ti
WDR (Class A)	22.9	8.04	0.52	0.41	0.053
Major contributor	94Nb 95.2%	94Nb 72.9%	108mAg 89.9%	108mAg 89.9%	94Nb 95.3%
2nd major contributor	108mAg 2.4%	99Tc 11.1%	60Co 5.4%	60Co 5.3%	26Al 3.9%
3rd major contributor	99Tc 1.95%	63Ni 6.92%	94Nb 4.22%	94Nb 4.24%	99Tc 0.64%
WDR (Class C)	2.19	0.60	0.0076	0.006	0.0053
Major contributor	94Nb 99.6%	94Nb 92.7%	108mAg 69.6%	108mAg 69%	94Nb 96%
2nd major contributor	108mAg 0.27%	99Tc 0.92%	94Nb 28.7%	94Nb 28.7%	26Al 3.9%
3rd major contributor	99Tc 0.13%	108mAg 0.9%	26Al 1.3%	26Al 1.8%	99Tc 0.39%

still suitable for class C disposal even in this rather severe case.

The effect of alternate modes of operation is shown in table 4. If the fuel ratio in the tokamak is changed to 3:1, ^{3}He to D, the neutron production is reduced by about a factor of three. HT-9 and Tenelon structure is now suitable for class A disposal and PCA is suitable for class C disposal. If a change is made in reactor type to a tandem mirror the neutron production is reduced due to the change in the D–D and D–T reaction rate relative to the ^{3}He–D reaction rate. The effect is a reduction in WDR by 29% relative to the reference case. Tenelon is now class A waste for practical cases and modified HT-9 is borderline. PCA is borderline as class C waste. A further reduction may be obtained by operating the tandem mirror in a 3:1 ^{3}He to D mode. As is shown in table 4 all structural materials except perhaps HT-9 are class C but HT-9 and PCA remain unsuited for class A disposal.

As mentioned, the design chosen for the cases discussed was based on a simple extrapolation of previous designs and while it appears to be adequate it would be

Table 3
Comparison for WDR 1 year after shutdown
1st wall-shield
[Tokamak 1:1 ^{3}He:D (1 MW/m^2 fusion power loading and 30 years operation)]

WDR	HT-9		PCA		MHT9		Tenelon		V15	
	Class A	Class C	Class A	Class C	Class A	Class C	Class A	Class C	Class A	Class C
Over volume W/O plasma	22.9	2.2	8.0	0.60	0.52	0.0076	0.41	0.006	0.053	0.0053
Over volume including plasma	15.8	1.5	5.6	0.42	0.36	0.0053	0.28	0.0042	0.037	0.0036
Compressed	327	31.3	115	8.6	7.4	0.11	5.8	0.086	0.76	0.075

Table 4
Comparison for WDR 1 year after shutdown
1st wall-shield
[Tokamak and tandem mirror (1 MW/m^2 fusion power loading and 30 years operation)]

WDR	HT-9		PCA		MHT9		Tenelon		V15	
	Class A	Class C	Class A	Class C	Class A	Class C	Class A	Class C	Class A	Class C
TOK 1:1	22.9	2.2	8.0	0.60	0.52	0.0076	0.41	0.0060	0.0053	0.0053
TOK 3:1	8.9	0.85	3.1	0.234	0.20	0.0029	0.16	0.0023	0.021	0.0020
TM 1:1	16.3	1.6	5.7	0.43	0.37	0.0054	0.29	0.0043	0.037	0.0037
TM 3:1	5.8	0.55	2.0	0.15	0.13	0.0019	0.10	0.0015	0.013	0.0013

desirable to see what improvements could be made. In all cases the major contributions to the WDR come from nuclei that are formed in (n, γ) reactions. Fig. 6 is a plot of the reaction rate leading to ^{94}Nb the major contribution to the WDR in HT-9 and PCA. It is seen that much of the activity is induced by neutrons of energies less than one electron volt. If the low energy flux could be suppressed, the production rate of ^{94}Nb or any (n, γ) reaction would be reduced. To verify this the reference case with HT-9 was recalculated with the addition of 5% boric acid to the water. The results are shown in table 5. The calculations indicate that the addition of the boron had the desired effect. The WDR is reduced to about 14% of the value without boron.

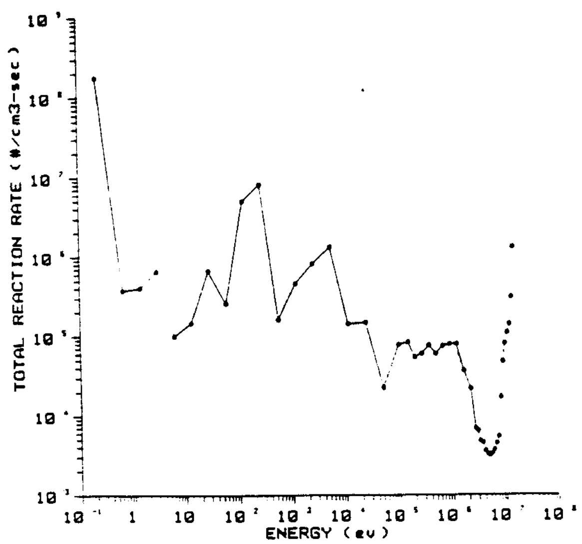

Fig. 6. Reaction rate versus energy for the production of ^{94}Nb in the first-wall-shield of the tokamak reference case.

While this calculation was not repeated for the other possible structural materials, significant reductions would be expected for these cases also.

Another approach to reducing the WDR of these (or any other) reactors is to reduce the level of the impurities which are major sources of long lived activities. In HT-9 for example most of the WDR is due to ^{94}Nb and most of the ^{94}Nb is produced from ^{93}Nb. Thus the WDR of HT-9 is very dependent on the amount of Nb in the alloy. As noted earlier the compositions of HT-9 used in the calculations were taken from the BCSS report. Since the publication of that report there is reason to believe that the Nb level quoted in the BCSS report may be too large. In particular, experiments performed by Lechtenberg at General Atomic indicate that their HT-9 samples have a Nb content of 0.0011% [10] compared to 0.11% quoted in the BCSS report. A reduction in Nb content to this level would have a significant effect on the WDR of HT-9. While there is evidence that the level assumed for HT-9 could be reduced, the approach taken here is to use the values in the BCSS for reference calculations pending a definitive determination of the concentration of all trace elements which contribute to the WDR. However, it is of interest to determine the consequences of a reduction in the Nb content of HT-9. Consequently the reference case of a

Table 5
Comparison of WDR 1 year after shutdown
1st wall-shield HT-9 structure
[Tokamak 1:1 ^{3}He:D (1 MW/m^2 fusion power rounding and 30 years operation)]

	H_2O	H_2O-HBO_2
1st wall Class A	22.9	3.2
1st wall Class C	2.2	0.28

Table 6
WDR 1 year and 10 years after shutdown
1st wall-shield
[Tokamak 1 : 1 ^{3}He : D (1 MW/m^2 fusion power loading, 30 years operation)]

	HT-9 with 0.11% Nb + H_2O		HT-9 with 0.011% Nb + H_2O		HT-9 with 0.11% Nb + H_2O with boron		HT-9 with 0.011% Nb + H_2O + boron	
	Class A	Class C	Class A	Class C	Class A	Class C	Class A	Class C
1 year after shutdown	22.9	2.2	3.5	0.25	3.2	0.28	0.87	0.049
10 years after shutdown	22.8	2.2	3.4	0.24	3.2	0.28	0.86	0.049

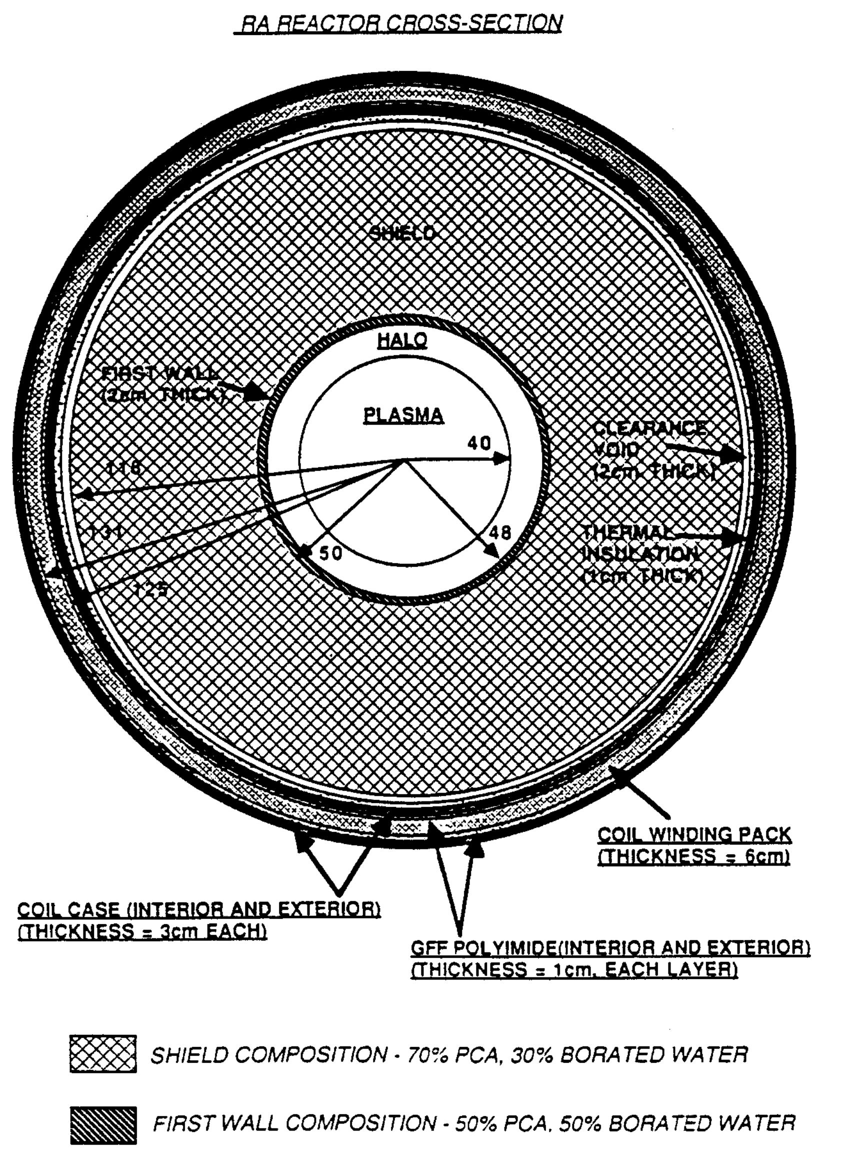

Fig. 7. Cross section of the preliminary design for the RA reactor with a ^{3}He : D ratio of 1 : 1.

tokamak with a 1 to 1 fuel ratio was recalculated assuming the Nb content was reduced by a factor of 10 to 0.011%. The results are shown in table 6. Considering the class A reference case, for a one year wait prior to disposal, of the total WDR of 22.9, 21.8 comes from ^{94}Nb and of this, 21.6 or 99% comes from ^{93}Nb. Reducing the Nb content to 0.011% reduces the WDR to 3.5, i.e. by about a factor of 7. The contribution due to ^{93}Nb is still significant being about 2.2. Even the complete elimination of Nb would not result in a WDR < 1.

Since much of the residual WDR is due to reactions induced by low energy neutrons the addition of boron to the water in the first wall-shield would still be beneficial. As shown in table 6 adding boron to a first wall-shield with HT-9 structure (0.011% Nb) reduces the class A WDR to 0.87 and the class C rating to a low value of 0.049. In both cases very little is gained by deferring disposal until 10 years after shutdown.

5. RA design first wall-shield

While these calculations are encouraging, the reference design chosen is not representative of what would be present in an actual design. Subsequent work has better defined the reactor concept to the point where an optimal shield design may be made incorporating the previous concepts. The design, given the designation RA, was based on a tandem mirror reactor with a 1 : 1 ^{3}He to D ratio, 50% tritium burnup and a fusion power loading of 2.94 MW/m^2 [11]. The first wall radius was 0.48 m and the length of the central cell was 99 m. The composition and thickness of the shield was chosen based on a nuclear heating in the magnets of 1 mW/cm^3. Boron was added to the water to suppress the low energy reactions. The structural materials considered were PCA and Tenelon. The ferritic alloys were eliminated from consideration since most of the shield would be operated below the ductile to brittle transition temperature. The vanadium alloy was not considered because of concerns related to cost and availability. The geometry and composition of the shield is shown in fig. 7. The activity per centimeter of central cell is shown in fig. 8. The initial activity calculated corresponds to 0.53 Ci/W$_{fusion}$ for PCA and 0.87 Ci/W$_{fusion}$ for Tenelon. In both cases the initial drop off is rather slow requiring well over a year to be reduced by a factor of 10. The afterheat is shown in fig. 9. The afterheat from Tenelon corresponds to 3.5 MW at shutdown, dropping off to 0.22 MW after one day. The afterheat from PCA is lower by about a factor of three at shutdown being 1.1 MW. However, after one day it is somewhat higher than

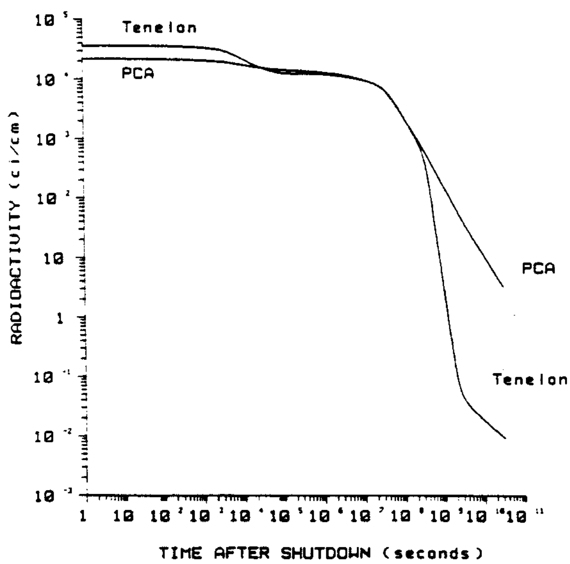

Fig. 8. Activity per centimeter after shutdown of the RA reactor following 30 years of operation.

Tenelon, namely 0.28 MW. The waste disposal rating results are shown in table 7. It is seen that PCA does not meet the requirements for surface disposal although it comes within about a factor of two. Tenelon, however, meets the class C criterion with considerable

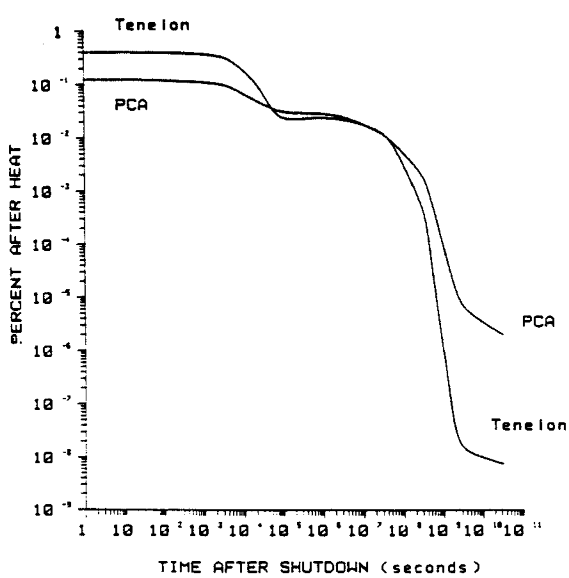

Fig. 9. Afterheat after shutdown of the RA reactor following 30 years of operation.

Table 7
Comparison of WDR for "RA" 1 year after shutdown

Zone	Material			
	PCA		Tenelon	
	Class A	Class C	Class A	Class C
F.W.	358	22.3	11.9	0.20
F.W. and shield W/O plasma	33.9	2.11	1.21	0.019
F.W. and shield including plasma	28.3	1.76	1.01	0.016
F.W. and shield compressed	48.7	3.03	1.74	0.027

margin even for the completely compressed case. If the option of averaging over the whole blanket assembly including the plasma region could be utilized, Tenelon would also qualify for class A disposal. Although PCA does not meet surface burial criteria, if additional absorber were to be introduced into the structure it might be possible for the WDR to be reduced to the point where it too would qualify. In neither case would waiting for a longer time prior to disposal improve the situation materially. For example, in the case of Tenelon, delaying disposal until 10 years after shutdown reduces the WDR by only 8%.

6. Conclusion

The utilization of reactors based on the D–^{3}He reaction results in significant improvement in waste disposal both from the standpoint of waste disposal rating and volume of material. The calculation for the MINIMARS reactors indicated that over a 30 year lifetime 6 complete blankets with a WDR of 0.53 (class C) would have to be disposed of, in addition to a reflector with an unacceptable waste disposal rating of 3.4. In contrast, a first wall-shield assembly in a similar sized reactor operating on the D–^{3}He cycle could operate for the full reactor life with a waste disposal rating < 1 for class C and might be suitable for disposal as class A waste.

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