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The impact of materials selection on long-term activation in fusion power plants

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Abstract

Neutron-induced transmutation of materials in a D–T fusion power plant will give rise to the potential for long-term activation. To ensure that the attractive safety and environmental characteristics of fusion power are not degraded, careful design choices are necessary. An aim of optimising power plant design must be to minimise both the level of activation and the total volume of active material that might ultimately be categorised as waste requiring disposal. Materials selection is central to this optimisation. In this paper we assess the influence of materials choices for a power plant on the waste volume and the potential to clear (i.e. remove from regulatory control) and recycle material. Although the use of low activation materials in regions of high neutron flux is an important part of the strategy to minimise the *level* of activation, different choices may result from a strategy aimed at minimising the *volume* of active waste.

1. Introduction

In the safety and environmental performance of fusion power a key issue is the potential to generate active waste that might need long-term disposal in a waste repository. This material arises in the final decommissioning of a power plant at end of life, together with that accumulated from replacement of components such as divertor and blanket modules during operation. The total volume of activated material could be at least as large as that from a fission reactor plant of similar electrical generating capacity, but the biological hazard potential of the material is much lower and the rate of decay very much more rapid [1].

The development of low activation materials for fusion has allowed considerable advances in reducing the likely impact of waste from fusion power [2]. These have included the application of recently proposed principles for removing material from regulatory control ('clearance') when its activity has dropped to very low levels. With the aid of such principles, it is appropriate to increase the attention given to reducing the total *volume* of waste arising from a fusion power plant, in view of the high importance that this is likely to have in the perception of public and media in developing attitudes towards fusion.

In this paper we concentrate on the volume of material that may be classified as active waste from a fusion power plant. In particular the influence of material selection, in those parts of the plant exposed to a neutron flux, is studied. This is done with reference to a series of neutronic and activation calculations with simple models based on the tokamak power plant designs studied in the second phase of the European Safety and Environmental Assessments of Fusion Power (SEAFP-2) [3], augmented by a lithium metal/vanadium concept from ANL [4] and a silicon carbide variant of one of the SEAFP blankets.

2. Radioactive waste volumes

2.1. Sources

For an indication of the volumes of active material originating in different parts of the power plant, it is

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useful to look at the outcome of conceptual plant design studies. In the SEAFP-2 [3], analyses were performed of the activation in all regions of each of three alternative power plant concepts [5,6]. From these data, an assessment was made of the total active material arising at the end of plant life including that resulting from routine blanket and divertor replacements during operation, and options for its disposal in radioactive waste repositories [7]. Although it is relatively straightforward to evaluate the total mass of material, the volume also depends on assumptions affecting the overall density such as the extent to which components have been dismantled, cut up and materials separated.

A typical set of results from Ref. [7] are illustrated in Fig. 1. This shows the active material following the 25 years life of a tokamak power plant of 3 GW fusion power, with in-vessel components including the first wall (FW) constructed of low-activation martensitic steel, and with a helium-cooled lithium orthosilicate ceramic pebble bed blanket. The in-vessel shield is water-cooled stainless steel 316, and the vacuum vessel (VV) is a reduced-activation austenitic steel. The volume values in Fig. 1 are the total active material volumes of the components grouped into the major regions of the plant from the toroidal field coil (TF) windings and structure on the inboard (ib) side through to those on the outboard (ob) side. They are based on preparing the material for disposal according to the requirements of a Swedish waste repository.

The regions in the volume break-down of Fig. 1 are ordered according to increasing activity at 50 years after plant shutdown. Note that the logarithmic activity scale spans ten decades of total activity, illustrating the large range of activation levels encountered. It is evident that a large proportion of the volume is at a relatively low level of activation, in fact 52% of the volume contains only 0.11% of the total activity.

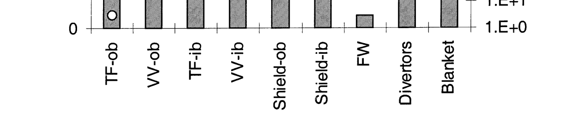

Fig. 1. Total volume of active material from an SEAFP-2 power plant, and activities 50 years after shutdown, originating in various regions (data from Ref. [7]). TF – toroidal field coils, VV – vacuum vessel, ib – inboard, ob – outboard, FW – first wall.

While it is valuable to reduce the relatively high activity in the in-vessel regions represented by the three bars at the right of the figure, such efforts will have little impact on the total volume of material that might be classified as radioactive. The vessel and ex-vessel regions, on the other hand, may have the potential to be reduced in activity below the threshold for inclusion in the volume regarded as active waste. The four bars at the left represent the vacuum vessel and the TF coils with their casing and related structure, which comprises 38% of the total active volume in this case. As will be seen in Section 3.1, efforts to decrease the peak activation levels in the in-vessel components may have an adverse effect on the potential to reduce the waste volume from these ex-vessel regions. Eventually an optimisation (or compromise) will be needed.

2.2. Reduction strategies

It is clear that the leftmost bar in Fig. 1 represents a large volume which is at a very low level of activity at 50 years after plant shutdown. However, national and international regulations governing radioactive waste disposal do not currently offer an easy path to removal of this material from control. Unless this can be done, even though it is at a very low level, it would still be classified as 'waste', with the associated costs of handling, storage and disposal, as well as public image.

Proposals by the IAEA for the removal from regulatory control of solid material [8] provide a procedure which, if adopted by national nuclear licensing authorities, could allow formerly active material to be 'cleared' as non-radioactive. This clearance procedure is based on nuclide-by-nuclide levels defined in the IAEA proposals, leading to the definition of a 'clearance index' for any specific mixture of nuclides. If this index is less than unity, the material may be declassified as non-active, and treated as normal scrap material for disposal or reuse.

Studies such as those in SEAFP-2 [9] have shown the potential of this approach. For the example case of Fig. 1, Rocco and Zucchetti [10] showed that all of the outboard TF coil region, and much of the inboard TF material, could be cleared 50 years after shutdown. The outboard parts of the vacuum vessel are also close to clearance level, and could be cleared after a further delay or by mixing some with lower activity steel from cooling pipes (if this is permitted). The study also assessed the potential for recycling of those materials which do not reach the clearance level, assuming that recycling operations are limited only by the ability to handle the material, either hands-on or remotely. This approach is also being applied within the ARIES-AT study. Thus 'recycling limits', based on the contact gamma-dose rate, are adopted, but this approach does not address whether the recycling of such materials would be economically viable. However, other studies have established the feasibility of recycling at least some selected fusion materials [11–14].

It is evident from such studies that the possibility of clearance and recycling of materials from a fusion power plant provides a potential reduction in the total waste volume that requires disposal. However in present conceptual designs the realisation of this potential is incomplete or marginal. In the following sections the influence of materials selection on the success of the clearance and recycling approach is examined.

3. Influence of materials selection

3.1. First wall and blanket

The fusion community has spent considerable effort to develop low activation materials for use as first wall and blanket structure [15,16]. The concept of 'low activation' relies on the reduction in the effects of activation, and in particular aims to limit:

1. decay heat – which might drive a temperature transient during postulated loss-of-coolant accidents;
2. gamma radiation – which might limit handling operations on the intermediate time scale of plant maintenance, disassembly and possible materials recycling; and
3. low residual activity, particularly in terms of biological hazard potential, on the longer time scale of ultimate decommissioning and disposal.

A material is low activation either because its constituents have low neutron capture cross sections or because the products of neutron reactions are benign (e.g. short-lived decay), or a combination of both of these characteristics. Ideally all neutron lives would terminate in a 'useful' absorption in lithium-6, producing a triton, but in a practical blanket a significant fraction of volume is occupied by structural material. If this structure is of a low activation material selected by virtue of a low absorption cross section then the blanket is likely to be more transparent to neutrons, leading to increased activation in the shield (see Section 3.2).

To illustrate this behaviour, calculations have been performed of the activation in all regions of a variety of conceptual power plant designs. Simple one-dimensional radial mid-plane models were used, with neutron flux spectra computed using the discrete ordinates code ANISN [17], employing 175-group neutron cross sections from the FENDL/E-2 library [18]. Time-dependent nuclide inventories were then calculated in each radial region using the European Activation System (EASY-99) comprising the FISPACT-99 code [19] and the EAF-99 nuclear data library [20]. This one-dimensional modelling approach is considered to be adequate for the scoping calculations required to illustrate the points under discussion, although it should be appreciated that 3-D effects such as neutron streaming along blanket and shield penetrations are not represented in these calculations. Such effects could result in local peaks in activation significantly above the levels predicted here – however the trends identified for the bulk of the material are not affected.

The FISPACT-99 code includes the direct calculation of Clearance Index according to the IAEA recommended clearance levels [8]. The results of this for five different design concepts are shown in Fig. 2, for each radial region on the outboard side of the models. The legend summarises the features of each design concept in terms of the tritium breeding material used, the first wall and blanket structural material, and the primary coolant. The vacuum vessel and all ex-vessel regions are identical in all cases, and based on 316 stainless steel. The shield is water-cooled 316 stainless steel, only the thickness varies between cases. The space available for this in-vessel shielding depends on the blanket thickness required to achieve sufficient tritium generation in each design.

There are a number of parameters which influence the behaviour of the different models in Fig. 2, including the variation of the magnitude and shape of the energy spectrum of the neutron flux exiting the back of the blanket. This is much affected by the choice of breeding material, neutron multiplier and coolant water leading to a significantly increased rate of neutron moderation to thermal energies [21]. Since the only differences between the five plant designs are in the in-vessel regions, it

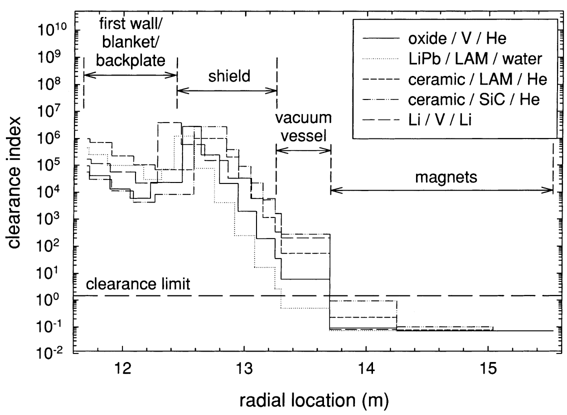

Fig. 2. Clearance index of radial regions in a variety of power plant designs 50 years after shutdown. Legend indicates breeder/structure/coolant, abbreviations: oxide – lithium oxide ceramic, LiPb – liquid lithium–lead eutectic, ceramic – lithium orthosilicate ceramic, Li – liquid lithium metal, V – vanadium alloy V–4Cr–4Ti, LAM – low-activation martensitic steel, SiC – silicon carbide composite.

is clear that choices here have a profound influence on ex-vessel activation. In particular the possibility of clearance of the vacuum vessel and parts of the coil structure appears to be sensitive to these choices.

In order to concentrate on the influence of the choice of structural material, further calculations were made of four plant models which are identical (helium-cooled lithium orthosilicate/beryllium pebble bed) apart from the structure. Four alternatives were compared: silicon carbide composite (SiC/SiC), a vanadium alloy (V–4Cr–4Ti), a low-activation martensitic steel, and 316 stainless steel. The impact of these alternatives on the waste volume is illustrated by the clearance index of ex-vessel components, for example the steel casing of the toroidal field coils. Fig. 3 shows this parameter plotted as a function of time – the time after end of plant life at which the steel casing of the magnet could be cleared varies from 33 to 47 years according to the selected first wall/blanket structural material. In a practical optimised design such effects would be partially offset by changing the shield design, for example by employing a thicker shield if SiC/SiC is selected, but there may be serious economic penalties incurred here (see Section 3.2).

The tendency for some low-activation materials, when used in the blanket, to result in increased activation ex-blanket, is due to the higher neutron transparency as noted above. But not all low-activation materials exhibit this behaviour. The radial variation of the total neutron flux through the outboard blanket is shown by the upper group of curves in Fig. 4, for three cases with different structural material (but otherwise identical). This energy-integrated flux presentation obscures differences in neutron spectrum, but nevertheless it is clear that the SiC/SiC case differs substantially from the two steels. Part of the reason is revealed in the lower group of curves, which show the total neutron absorption rates

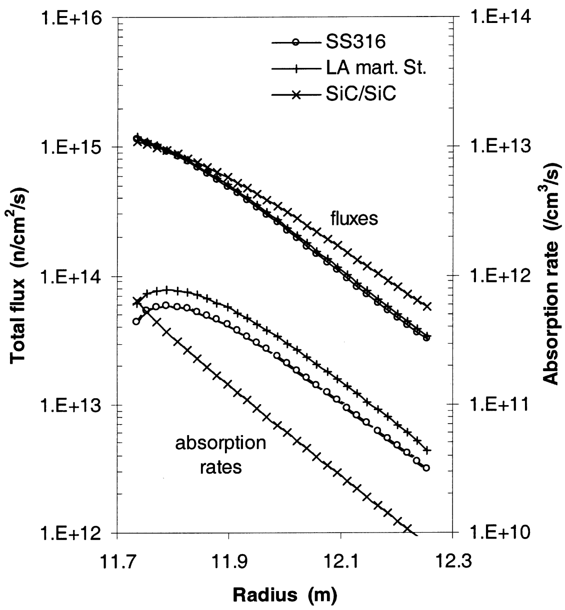

Fig. 4. Radial variation in blanket of total neutron flux and absorption rates in alternative structural materials.

in the three structural materials. The silicon carbide has much lower absorption, leading to the more gradual decrease in total neutron flux, and also to the low activation of the SiC/SiC itself. But the low activation martensitic steel, which also achieves a low level of activity compared with 316 stainless steel, actually has an absorption rate *higher* that that of 316 steel. There is also a difference in the shape of the absorption profile, the steels exhibiting a small peak about 5 cm into the blanket, due to the shift of neutron energy spectrum towards lower energies at which (n, γ) reactions are important in the steels, but almost absent in the SiC/SiC.

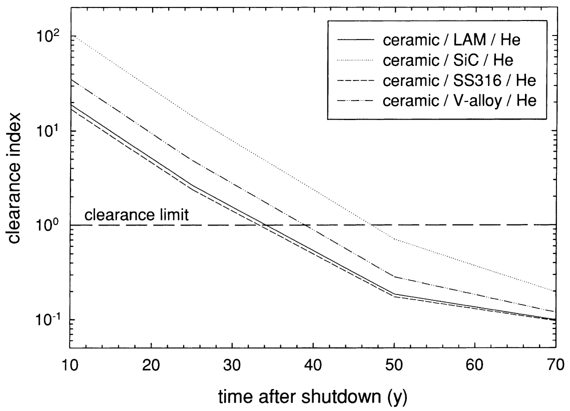

Fig. 3. Clearance index for outboard TF coil casing, for various FW/blanket structural materials (V – vanadium alloy V–4Cr–4Ti, LAM – low-activation martensitic steel, SiC – silicon carbide composite, SS316 – 316 stainless steel).

3.2. Shield

The primary purpose of the in-vessel shield is to reduce the radiation damage in the vacuum vessel, to avoid degradation of its mechanical properties and to permit re-welding operations as well as to reduce nuclear heating and damage in the toroidal field coils. The shield also provides the major attenuation of neutron flux that reduces the activation of the vessel and all ex-vessel components. But because of the intentionally high absorption rate, the shield itself is likely to exhibit a moderately high activation. As seen above, this is partly dependent on the level of transmission of neutrons through the blanket, but also on the activation characteristics of the shielding material itself.

Activation calculations, such as those shown in Fig. 2, indicate that the shield material has a clearance index too far above unity to have any prospect of being cleared on a practical time scale. But at least some of the material may be a candidate for recycling. Fig. 5 shows the contact gamma dose-rate at 50 years after shutdown in the steel–water shield of a plant based on a water-cooled lithium–lead blanket. The adopted limits for hands-on access to the material, and recycling operations by remote handling, are also indicated. These remote handling limits are indicative, not regulatory requirements at present, and will be revised in future work to reflect the most recent equipment survivability data in radiation fields and the latest worker radiation protection criteria. The two lines are for shields which differ only in the steel used, 316 stainless steel in one case, a reduced-activation austenitic, OPTSTAB, in the other. Clearly the use of this steel with improved long-term activation characteristics allows a greater volume to have the potential for recycling – more than half of it by hands-on techniques, and all of the remainder by remote handling.

But the improved long-term activation performance of OPTSTAB compared with 316 stainless steel is balanced by less favourable short-term properties. On the time scale of importance to postulated loss-of-coolant accidents, the decay heat generated by OPTSTAB is significantly higher than that of 316 stainless steel, as illustrated by the plot of the total decay heat in the shield, Fig. 6. The initially three times higher decay heat, due to a much higher manganese content, may result in higher peak temperatures in all in-vessel components in some accident scenarios [22]. Thus a balance must be struck between the requirements of waste volume reduction and those of accident consequence limitation. It is possible that an advanced shield design, optimised against *both* these requirements rather than simply minimum neutron flux at the vessel, may provide satisfactory performance. This will be the subject of future work.

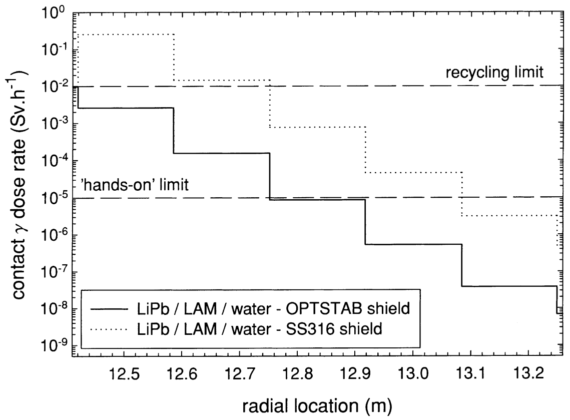

Fig. 5. Contact gamma dose-rate of outboard in-vessel shield, 50 years after shutdown, with alternative shielding materials.

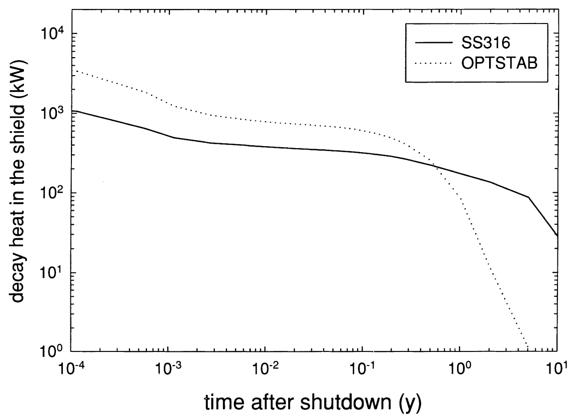

Fig. 6. Total decay heat in an in-vessel shield of different steels.

Clearly optimisation of the shield design must be done together with the blanket design. In Section 3.1 it was noted that different blanket materials choices lead to very different neutron flux transmission and thus variation in shielding requirements. The thickness of the shield is constrained by the available space within the vessel, and increasing dimensions here have a severe economic penalty on the cost of many major plant components.

3.3. Vessel and ex-vessel structure

In the vacuum vessel and certainly in the bulky components outside it, the neutron flux levels may be low enough for clearance to be achieved within a practical time period of tens of years after shutdown. As discussed in Section 3.1 and seen in Figs. 2 and 3, this is partly dependent on the neutron attenuation provided by the blanket and shield, with only one of the five designs compared in Fig. 2 allowing clearance of the vessel at 50 years.

But the potential to clear these components is also, of course, influenced by the material chosen for their construction. The use of a reduced-activation austenitic steel as an alternative to 316 stainless steel offers a marked improvement, as illustrated by Fig. 7, which compares the clearance index of parts of a vacuum vessel constructed of OPTSTAB with one of 316 steel, in the same plant. It takes some three times longer decay time for the 316 vessel to reach the clearance level.

These and other results for the vacuum vessel presented here are for the outboard portion of the vessel at the mid-plane. Other calculations have shown that in cases where the vessel falls below the clearance limit, there may remain a small section of the vessel on the inboard side for which clearance is not possible. This is due to the smaller volume of blanket and shield on the inboard side, dictated by geometric constraints, and a

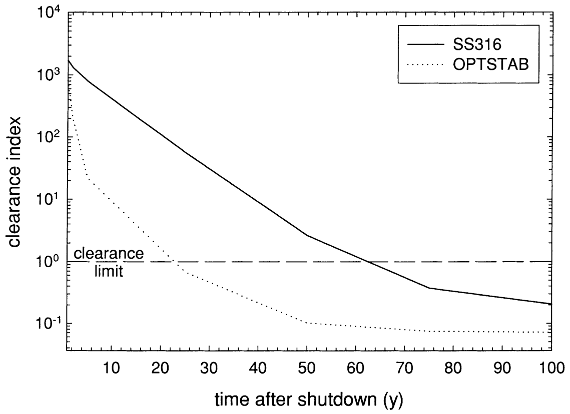

Fig. 7. Clearance index of outboard parts of a vacuum vessel constructed of different steels, in a plant using a water-cooled lithium–lead blanket and low-activation martensitic steel structure.

consequent higher neutron flux in the vessel wall, at least near the mid-plane.

In all these calculations, assumed materials compositions have included a comprehensive inventory of impurities, determined from an optimisation procedure [23] based on measured compositions. As has been recognised in various earlier studies [24–27], impurities play a significant role in the long-term activation properties – in the results presented here the activity in the outermost regions, the magnets and their outer coil case, are dominated by impurities. Thus in these regions the effect of swapping the type of steel used is negligible if the impurity content remains unchanged.

4. Conclusions

Results from earlier studies have indicated that a substantial part (typically 50%) of the active volume of materials arising from a fusion power plant are in the regions outside of the blanket and shield. In these outer components, the activation is low enough that the potential exists to clear the material as non-active after a practical time period, typically 50 years. Other parts of the plant, including the shield, may be candidates for recycling. But materials selection throughout the plant has an essential impact on realising this potential.

Efforts to reduce the maximum levels of activation in the first wall and blanket, while important, may lead to increased activation in the shield, vacuum vessel and magnet structures, with a consequent increase in the volume of material that must be classified as waste requiring repository disposal. Calculations of activation properties in a range of conceptual plant designs have shown how the possibility of achieving clearance and recycling of bulky components is influenced by choice of structural material for the first wall and blanket, as well as by the materials chosen for the components themselves.

The potential to achieve clearance of the vacuum vessel and permit recycling of the in-vessel shield can be realised only in some of the plant designs considered. The success of waste volume reduction is sensitive to material choices in all parts of the plant. In particular it is influenced by:

- Structural material choice for first wall and blanket, with some low-activation materials exhibiting markedly different neutron absorption characteristics to others. In particular, very low short-term activation of silicon carbide is due to its low absorption, so that its use leads to greater shielding requirements outside the blanket, with economic penalties, as well as higher activation of the shield, and consequent increased volume of materials that cannot be cleared or recycled.
- Blanket breeder and multiplier materials and choice of coolant.
- Choice of material for the shield. The 'best' shielding materials do not necessarily have the best activation properties. But the reduced-activation steel, OPTSTAB, assessed in calculations, while providing much greater recycling potential for the shield, generates higher short-term decay heat that may be a concern for some postulated loss-of-coolant accident scenarios.
- Material for the vacuum vessel and some ex-vessel structure, in particular the inner coil case. Here a reduced activation austenitic steel provides a significantly improved prospect of clearance as non-active. But in the outermost regions the longer-term activity is dominated by impurities.

These choices are part of an overall design strategy that has to balance and optimise a number of parameters. The active waste volume issue is important enough to be included in design optimisations with high priority. The conclusions of this work show that it must be included in design requirements for material selection in *all* parts of the plant, not just those in which the waste volume minimisation is expected to be made. In particular, low-activation material choices for the plasma facing and blanket components must be made with due regard to the influence on the total active waste volume from the plant as a whole.

Acknowledgements

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