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SEAL Subtask 6.2, Final Report

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ABSTRACT

The European Union has been engaged since 1989 in a programme to develop tritium breeding blankets for application in a fusion power reactor. There are four concepts under development, namely two of the solid breeder type and two of the liquid breeder type. At the Forschungszentrum Karlsruhe one blanket concept of each line has been pursued so far with the so-called breeder outside tube (BOT) type representing the solid breeder line. In the BOT concept, Li_4SiO_4 is used as ceramic breeding material in the form of pebble beds in combination with beryllium pebbles serving as neutron multiplier. Breeder and multiplier materials are arranged in radial-toroidal layers, separated by cooling plates. The coolant is high pressure helium which is circulated in series, at first through the first wall structure and subsequently through the cooling plates.

The safety and environmental impact of the BOT blanket concept has been assessed as part of the blanket concept selection exercise, a European concerted action aiming at selecting the two most promising concepts for further development. The topics investigated are: (a) blanket materials and toxic materials inventory, (b) energy sources for mobilisation, (c) fault tolerance, (d) tritium and activation product release, and (e) waste generation. No insurmountable safety problems have been identified for the BOT concept. The results of the assessment are described in this report. The information collected is also intended to serve as input to the EU 'Safety and Environmental Assessment of Fusion long-term Programme' (SEAL). The unresolved issues pertaining to the BOT blanket which need further investigations in future programmes are outlined herein.

Gesichtspunkte der Sicherheit und Umwelteinflüsse des heliumgekühlten Feststoffblankets vom Typ BOT für einen DEMO Reaktor

KURZFASSUNG

Im Fusionsprogramm der Europäischen Union werden seit 1989 tritiumbrütende Blankets für den Einsatz in Fusionsreaktoren entwickelt. Bislang befanden sich vier Varianten in der näheren Untersuchung, nämlich je zwei Varianten mit festem und flüssigem Brutstoff. Im Forschungszentrum Karlsruhe wurden beide Linien verfolgt. Für die Linie mit festem Brutstoff ist es das sogenannte BOT-Konzept (englisch für breeder outside tube). In ihm sind Li_4SiO_4 als keramischer Brutstoff in Verbindung mit Beryllium zur Neutronenmultiplikation, beide in Form von Kugelschüttungen, vorgesehen. Brutstoff und Multiplikator sind in abwechselnden radial-toroidal verlaufenden Schichten angeordnet, die durch Kühlplatten voneinander getrennt sind. Die ganze Anordnung wird durch gasförmiges Helium gekühlt. Dieses wird unter hohem Druck zunächst durch die Kanäle in der ersten Wand und anschließend durch solche in den Kühlplatten geleitet.

Die Sicherheit und Umwelteinflüsse des BOT-Konzeptes wurden im Rahmen des europäischen Blanket-Auswahlverfahrens betrachtet, welches zum Ziel hatte, zwei der vier Varianten für die weitere Entwicklung auszuwählen. Die dabei behandelten Themen waren a) Art und Mengen der beteiligten radiotoxischen Stoffe, b) Energieinhalte als mögliche Ursache für Aktivitätsfreisetzungen, c) Störfallverhalten, d) Freisetzung von Tritium und anderen Aktivitätsprodukten und e) Erzeugung radioaktiver Abfälle. Für das BOT-Konzept wurden wie bei den anderen Varianten keine schwerwiegenden Sicherheitsprobleme erkannt. Die Ergebnisse zu den untersuchten Themen werden im Bericht dargestellt. Am Schluß werden einige Aspekte herausgestellt, die noch weiterer Analysen bedürfen. Das Datenmaterial dient auch als Beitrag zu einem laufenden Vierjahresprogramm der EU über Sicherheit und Umwelteinflüsse der Fusionstechnologie (SEAL).

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1 INTRODUCTION

The European Union has been engaged since 1989 in a programme to develop tritium breeding blankets for application in fusion power reactors. There are four concepts under development. Two of these concepts use lithium ceramics as breeder material (solid breeder blanket), the other two concepts employ a eutectic lead-lithium alloy as breeder material (liquid metal breeder blanket).

At the Forschungszentrum Karlsruhe one blanket of each line has been pursued so far with the so-called breeder outside tube (BOT) type representing the solid breeder blanket line. In the BOT concept, Li_4SiO_4 is used as breeding material in the form of pebble beds in combination with beryllium pebbles serving as neutron multiplier. Both breeder and multiplier materials are arranged in alternating radial-toroidal layers, separated by cooling plates. These plates equipped with integral cooling channels were originally designed as arrays of imbedded cooling tubes, giving the concept the name „breeder outside tube". The coolant is high pressure helium, being circulated in series, at first through the first wall structure and subsequently through the cooling plates in the breeding region.

The present state of development of the BOT concept is currently being documented in a comprehensive status report [1]. It is part of the blanket concept selection exercise (BCSE) which aims at selecting two of the four concepts for further development. This is a joint effort among the associations developing the four concepts and industry under the auspices of the European Commission. The present report is essentially a stand-alone version of the contribution to the status report, dealing with the safety aspects of the BOT helium cooled solid breeder blanket for a DEMO reactor. It covers the topics (a) blanket materials and toxic materials inventory, (b) energy sources for mobilisation, (c) fault tolerance, (d) tritium and activation products release, and (e) waste generation and management.

In the frame of the safety and environmental assessment of fusion power long-term programme (SEAL) a task (SEAL 6) is being performed on „blanket safety analysis for the four European blanket concepts" in support of the BCSE. The task is co-ordinated by UKAEA. FZK agreed to provide input information under subtask SEAL 6.2 „Input for the dual coolant blanket and ceramic breeder blanket". Besides the above mentioned status report character, this report is also intended to contain the necessary information presumably needed in SEAL 6 as far as the BOT concept is concerned. In particular it includes supplemental data and references on mass inventories in different regions of the blanket, material composition in radial zones, initial impurities in solid materials, afterheat and thermal inertia, energy potentials, and cooling circuits.

2 BLANKET MATERIALS AND TOXIC MATERIALS INVENTORY

2.1 Volumes and volume fractions in the blanket segments

The total volume in different parts of the ceramic breeder blanket and the volume fractions of steel, breeder material, beryllium multiplier, primary helium coolant, purge helium and void have been calculated for both outboard and inboard blanket. The basis for the assessment is the vertical cross section for the DEMO reactor and the midplane cross sections of outboard and inboard blanket segments as shown in [2]. The main dimensions used in the calculations are displayed in Table 1.

TABLE 1: MAIN DIMENSIONS OF BLANKET

Item	Length (cm)
Outboard blanket	
Major radius to first wall at midplane	827.4
Radius of curvature of first wall (poloidally)	475
Poloidal angle relative to midplane of circular central blanket part (degrees)	-45.5 to + 45,5
Radius of first wall of upper blanket part	643
Length of upper blanket part	183
Thickness of first wall/breeding zone/manifold zone	2.5/56.1/41.4
Inboard blanket	
Radius of first wall at midplane	430
Length of vertical (central) blanket part	810
Length of upper 45 degrees inclined blanket part (top)	238
Length of lower -45 degrees inclined blanket part (bottom)	238
Thickness of first wall/breeding zone/manifold and shield zone	2.5/30/53.1

The computed volumes for the blanket parts (outboard central plus upper part, and inboard central part with upper and lower parts behind the divertors) are listed in Table 2. The total blanket volume (all inboard and outboard segments, full toroidal coverage) with the dimensions defined in Table 1 results to 823 m^3 with volume fractions of 35.5/3.7/19.6/15.4/6.9/18.9 percent for steel/breeder/beryllium/primary helium/purge helium/void, respectively. The total volume of all outboard blankets amounts to 63.5 % of the total blanket volume, leaving 36.5 % for the inboard.

The numbers quoted for the breeder and beryllium refer to the pebble volume (not to the packed bed volume) assuming a packing factor of 0.623 for the Li_4SiO_4 pebbles and 0.808 for the beryllium pebbles [2]. Then the volumes taken from Table 2 and multiplied by the material densities of 7700 kg/m^3 for steel, 2400 kg/m^3 for breeder, and 1850 kg/m^3 for beryllium yield a

- total weight of one outboard segment (without shield): 29.6 tons
- total weight of one inboard segment (including shield): 37.5 tons

A more detailed list of the steel volumes in the first wall, breeding zone, and manifold zone (respectively manifold plus shield zone for the inboard) is given in Table 3 for

each blanket part. Here it can be seen that 51 % of the total steel volume is in the outboard and 49 % in the inboard. Most of the steel is located in the manifold zone.

The percentages in first wall/breeding zone/manifold are 6/35/59 in the outboard. In the inboard the corresponding percentages are 4/17/79.

Note that the removable shield is not included in the volumes given for the outboard.

TABLE 2: VOLUME OF SOLID BREEDER BLANKET (in m^3)

Blanket Part	Steel	Breeder	Beryl-lium	Primary Helium	Purge Helium	Void
Outboard central part	118.6	18.3	96.9	68.5	34.1	95.7
Outboard upper part	31.2	3.6	19.0	14.8	6.7	15.2
Outboard total	149.8	21.9	115.9	83.3	40.8	110.9
Inboard central part	76.8	5.0	26.5	25.0	9.3	25.9
Inboard upper part	32.7	1.8	9.5	9.2	3.3	9.5
Inboard lower part	32.7	1.8	9.5	9.2	3.3	9.5
Inboard total	142.1	8.6	45.5	43.4	16.0	44.9
Outboard plus Inboard total	292.0	30.5	161.4	126.7	56.8	155.8
Total blanket volume	823.1					

TABLE 3: STEEL VOLUMES IN RADIAL BLANKET ZONES (in m^3)

Blanket section	First wall	Breeding zone	Manifold and shield
Outboard central part	6.2	40.6	71.8
Outboard upper part	2.8	12.4	16.0
Outboard total	9.0	53.0	87.8
Inboard central part	3.4	12.5	60.9
Inboard upper part	1.2	6.0	25.5
Inboard lower part	1.2	6.0	25.5
Inboard total	5.7	24.5	111.9
Outboard plus Inboard total	14.7	77.5	199.7
Total steel volume in blanket	291.9		

Multiply by the density of 7700 kg/m^3 to obtain masses.

2.2 Coolant inventories

Coolant inventories in the primary helium cooling circuits of the solid breeder blanket have been calculated for the piping added on estimates for other components like steam generators, circulators and clean-up systems.

The loop arrangements and thermal power for the total outboard and inboard blanket have been taken as described in [2], accounting for redundancies in the primary

circuit layout. This leads to larger power capacities as compared to the nominal power to be removed from the blanket segments. Hence, the loops (except for the feeders connecting the ring collectors and the blanket segments) become larger than they would be required for the nominal power capacity. Table 4 gives an overview of the nominal thermal power to be removed by the primary helium circuits from the inboard and outboard blanket segments (including the first wall). It also shows the redundancy factors discussed below and the required power capacity of the external loops which results from equation 1. As a consequence of this redundant circuit layout, the external loops (including the ring collectors and the components) have to be dimensioned for the case that one circuit of the subsystem has failed (Table 4).

TABLE 4: THERMAL POWER AND REDUNDANCY FACTORS FOR HELIUM COOLING CIRCUITS

Blanket Region	Thermal power per segment or total (MW)	No. of blanket segments	No. of primary circuits	Redun-dancy factor	Power capacity per circuit or total (MW)
Outboard blanket	37.9	48	2 x 6	1.2	182.1
Inboard blanket	21.2	32	2 x 3	1.5	169.8
Total blanket	2500	80	18	1.28	3204

The redundancy factor results from the subsystems layout as follows: All subsystems consisting of n circuits (n=3 or 6) are designed in a 2-out-of-n redundancy, meaning that two circuits of a subsystem must fail for the plant to be unavailable. For instance, for a 2-out-of-3 subsystem chosen for the inboard to be unavailable, 2 circuits must fail. Hence, it is available if 1 circuit fails, in which case the remaining 2 circuits have to take 1.5 times (the redundancy factor) their nominal share. In the same way the redundancy factor of 1.2 corresponds to a 2-out-of-6 subsystem chosen for the outboard.

The required power capacity per circuit (in the event that one circuit of a subsystem has failed or is in stand-by) has been calculated by use of the following equation. Note that due to the redundancy the overall installed power capacity of the cooling system (3204 MW) is 28 % larger then the nominal power to be removed from the blanket (2500 MW).

$$P_{circuit} = \frac{P_{segment} \cdot n_{segment}}{n_{circuit}} \cdot f_{redundancy} \qquad (1)$$

where $P_{circuit}$ = required power capacity per circuit

$P_{segment}$ = thermal power per segment

$n_{segment}$ = number of blanket segments

$n_{circuit}$ = number of installed primary circuits

$f_{redundancy}$ = redundancy factor (see Table 4)

The helium inventory in the primary cooling system has been estimated based on the pipe dimensions proposed in [3] and on a first layout of the steam generators. For the circulators and the clean-up systems a guess has been made. Table 5 summarises the helium inventory in the different circuit components for the outboard and for the inboard blanket, assuming that all circuits are operating simultaneously at reduced power. In this case the total helium inventory would amount to 22000 kg, 62 % of which would be in the outboard and 38 % in the inboard cooling systems. The largest subsystems would thus contain about 7000kg/4000kg of helium (outboard/inboard). These rounded numbers will be used for further safety considerations. Most of the helium is contained in the pipework (78 % of the outboard and 82 % of the inboard system's inventory) with only 3 to 4 % being in the blanket segments.

Note: If the redundant circuits (1 out of 6 outboard or 1 out of 3 inboard) were isolated from the operating subsystems the inventory in the subsyst would be reduced by 11 and 17 %, respectively).

TABLE 5: HELIUM INVENTORY IN CIRCUIT COMPONENTS

Component	Total Volume (m^3)	Total Mass (kg)
Outboard, Nominal power 1821 MW		
- Primary helium piping (12 circuits)	1709	10686
- 48 Blanket segments	83,3	505
- 12 Steam generators	404	2428
- 12 Circulators	12	88
- 2 Clean-up systems	10	50
Subtotal outboard circuits	2218	13757
Inboard, Nominal power 679 MW		
- Primary helium piping (6 circuits)	1094	6789
- 32 Blanket segments	43,4	263
- 6 Steam generators	194	1163
- 6 Circulators	6	43.8
- 2 Clean-up systems	5	25
Subtotal inboard circuits	1342	8283
Total outboard plus inboard circuits	**3560**	**22041**

2.3 Toxic inventories

Radioactive inventories in the different regions (inboard, outboard) and radial zones (first wall, breeding zone, manifold) of the blanket as well as in the cooling and purge system have been assessed in [2] with supplemental information given in [4]. The inventories relevant for safety considerations are summarised below in two categories, i.e., tritium (Table 6) and activation products (Table 7).

2.3.1 Tritium inventory

Tritium in breeder: The inventory in Li_4SiO_4 has been calculated from the measured residence time for the reference pebbles and is small (≈10 g) compared to the tritium

generation rate which is of the order of 370 g/d. So the tritium concentration in Li_4SiO_4 by weight is on the average 0.14 wppm.

Tritium in multiplier: The inventory in the beryllium is the largest amounting to 1278 g at the end of the blanket life time [2]. This is about one third of the tritium generated in the beryllium (≈3400 g) according to activation calculations [4] and is strongly temperature dependent. Most of the tritium generated in the beryllium occurs in the outboard center parts (90 %), only 6 % in the inboard center part, and the rest in the upper and lower part of the blanket segments.

Tritium in structural material: The structural material MANET will receive tritium from two sources, i.e., from migration (from the plasma, coolant, purge gas, breeder, multiplier) and from transmutations. The first fraction has not been investigated here, but is assumed to be small. The second effect results from FISPACT calculations [4]. Presuming that all the tritium generated in the neutron and gamma field is retained in the structure, the tritium in the blanket structural material will accumulate to a total of 12 g after 20000 hours of operation in 2250 tons of steel. This is an average tritium concentration by weight of 5.3 wppb. The spatial distribution is very different, though, ranging from 75 wppb in the first wall of the outboard central part down to 0.14 wppb in the manifold zone of the outboard upper part. The corresponding range for the inboard is 51 wppb down to 0.4 wppb. Tritium inventories for different blanket regions and radial zones are given in Table 6. In any case they are small, amounting to 12 g in the entire blanket structure.

TABLE 6: BLANKET MATERIALS AND TRITIUM INVENTORY

Blanket zone or system	Total mass inventory (tons)	Total tritium inventory (g)
Breeder material (Li_4SiO_4) OB/IB	52.6/20.6	7/3
Neutron multiplier (beryllium pebbles) OB/IB	214/84	1278 (total)
Structural material (MANET) total OB/IB	1153/1095	9.0/3.1
First wall OB/IB	69/44	4.0/1.5
Breeding zone OB/IB	408/189	4.7/0.6
Manifold and shield zone OB/IB	676/862	0.3/1.1
Primary coolant (helium) OB/IB	14/8	0.29/0.17
Purge gas (helium)	0.015	0.1

OB/IB refers to Outboard/Inboard

Tritium in primary coolant: The H-3 concentration in the helium coolant flow has to be kept low in order to minimise the H-3 losses through the steam generators into the steam cycle as well as leaks from the entire cooling system. Therefore, the H-3 influx into the helium stream (mainly from the first wall, but to some extent also from the breeding region) has to be permanently removed in a H-3 extraction unit via a bypass flow of typically 0.1 % of the main stream. The equilibrium H-3 concentration in the helium loop is a function of the H-3 influx into the helium, the H-3 losses from the helium circuits (which are relatively small), the extraction efficiency of the H-3 extraction unit, and the flow rates in the bypass going through the extraction unit. A simple parametric study on the equilibrium H-3 concentration as function of the above mentioned parameters is given below.

The schematic flow diagram is shown in the insert in Figure 1. Mass balance at the nodes of the network yields the following equations for the tritium mass concentrations at the outlet of the blanket, c_2, and at the inlet to the blanket, c_1:

hot leg:
$$c_2 = \frac{\dot{m}_B - \dot{m}_S}{\alpha \cdot \varepsilon \cdot \dot{M}} \qquad (2)$$

cold leg:
$$c_1 = c_2 - \frac{\dot{m}_B}{\dot{M}} \qquad (3)$$

where:

$\dot{m}_B$ H-3 influx into the main helium stream within the blanket
$\dot{m}_S$ H-3 losses in the steam generators including leaks from the circuits
$\dot{M}$ Helium mass flow rate in the main circuit
c_1, c_2 Mass concentration of H-3 in helium at blanket inlet, outlet
c_3, c_4 Mass concentration of H-3 in helium at extraction unit inlet, outlet
ε Fraction of $\dot{M}$ as bypass flow through the extraction unit

$\alpha \equiv \frac{c_3 - c_4}{c_3}$ Extraction efficiency of the H-3 extraction unit

In Figure 1 is plotted the equilibrium H-3 concentration in the hot leg of the helium circuit, c_2, vs. the H-3 influx, $\dot{m}_B$, into the main helium stream within the blanket with the extraction efficiency, α, as parameter. The other parameters ($\dot{M}$, ε, $\dot{m}_S$) have been chosen as typical for the DEMO ceramic breeder blanket. Since $\dot{m}_S << \dot{m}_B$ the H-3 concentration is proportional to the H-3 influx and proportional to the inverse of the extraction efficiency, α, and the flow rate in the bypass through the extraction unit, $\varepsilon \dot{M}$. There is a large uncertainty in determining the H-3 influx, $\dot{m}_B$. A likely value is of the order of $\dot{m}_B \approx 3$ g/d, but substantially larger values cannot be excluded at present. For $\dot{m}_B = 3$ g/d and α=0.7 the equilibrium H-3 concentration in the hot leg would be 21 wppb and in the cold leg nearly the same. Hence, the total amount of tritium in the primary helium according to Table 6 amounts to 0.46 g which is very low.

Tritium in purge gas: It is assumed that the full purge gas flow of approximately 0.6 kg/s is continuously circulated through the tritium extraction system. The same rationale applies as explained for the primary cooling system with the fraction of mass flow passing the extraction units ε=1 in equation 2. Using again an extraction efficiency α=0.7 and a tritium influx into the purge gas $\dot{m}_B$ = 370 g/d the equilibrium tritium concentration in the purge gas at the blanket outlet would result from equation 2 to 10 wppm and at the inlet from equation 3 to 3.1 wppm, or approximately 6.6 wppm on average. This corresponds to a tritium inventory of 0.1 g in an assumed helium inventory of 15 kg (about 3 kg in piping, 6 kg in blanket segments, 6 kg in extraction components).

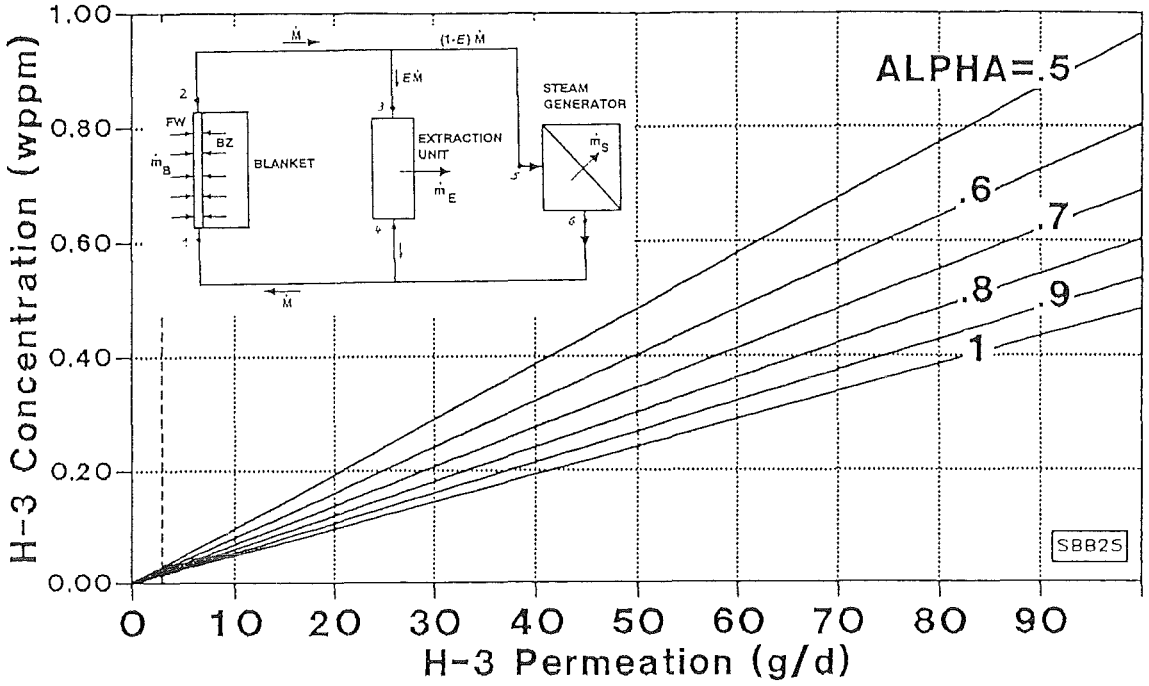

Figure 1: Equilibrium tritium concentration in the helium cooling system.

2.3.2 Activation products inventory

Activation products inventories in the blanket materials (breeder, multiplier, and structural material) are reported in detail in [4]. To give an overview of the specific and total activity in different regions (outboard, inboard) and radial zones (first wall, breeding zone, manifold and shield zone) as referred to in this report the data from FISPACT calculations [4] have been reconfigured and are presented below.

The specific activity taken as the mean of the respective material in different blanket regions and radial zones is illustrated in Figure 2 for typical times after shutdown. At time zero (after 20000 hours of operation) the specific activity per unit mass is not very different among the three blanket materials of the breeding zone (breeder, multiplier, MANET BZ). It ranges between about $3x10^{13}$ and $1x10^{14}$ Bq/kg. For the structural material curves are also plotted for the first wall (FW) and the manifold and shield zone (SH) which differ by about two orders of magnitude. Please note that the value for specific activity in the breeder (Li_4SiO_4) does not include any tritium. In that case it drops by four orders of magnitude within the first year after shutdown, whereas for MANET it declines only by a factor of 3 to 8. There is no significant difference in the specific values for inboard and outboard.

The specific activity in beryllium is initially dominated by He-6 which decays rapidly (half life 0.8 s). Thereafter the tritium takes the lead, contributing more than 99 %. This implies that all the tritium were trapped in the beryllium. Actually, only about one third of the tritium will be retained in the pebbles so that the specific activity in the time span of minutes to several 10 years ought to be reduced by half an order of magnitude.

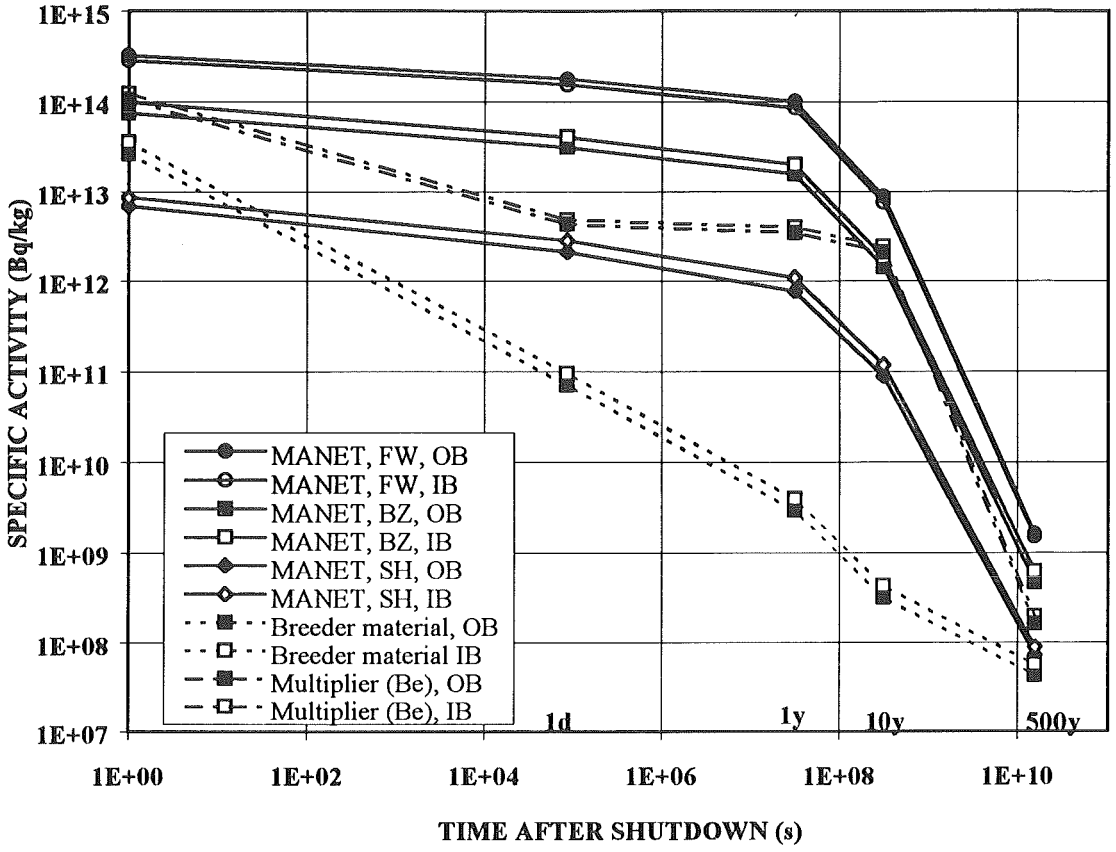

Figure 2: Specific activity in blanket materials

The total activity of the blanket materials has been evaluated by multiplying the average specific activities for the outboard and inboard plotted in Figure 2 by the mass inventories (Table 7). The total activity in the outboard (48 segments, all materials) after decay times of 0s, 1d, 1y, 10y, 500y amounts to 8.2×10^{19}, 2.8×10^{19}, 1.5×10^{19}, 1.7×10^{18}, and 3.9×10^{14} Bq, respectively. In the inboard the corresponding values reach about 60 % of the outboard values. The numbers quoted for the multiplier include the contribution of the tritium as calculated by FISPACT (about 3.4 kg of tritium in the whole multiplier which is actually too high), whereas for the breeder material tritium has been excluded because the amount of tritium being retained in the breeder is comparably small.

TABLE 7: TOTAL ACTIVITIY IN BLANKET MATERIALS (in Bq)

Material and region	Mass (tons)	Time after shutdown				
		0 s	1 d	1 y	10 y	500 y
Breeder material, OB	52.6	1,4E+18	3,7E+15	1,5E+14	1,7E+13	2,3E+12
Breeder material IB	20.6	7,4E+17	2,0E+15	8,2E+13	8,8E+12	1,2E+12
Breeder material OB+IB	73.2	2,2E+18	5,6E+15	2,4E+14	2,6E+13	3,5E+12
Multiplier (Be), OB	214.4	2,2E+19	9,2E+17	7,4E+17	4,4E+17	3,5E+13
Multiplier (Be), IB	84.2	1,0E+19	4,1E+17	3,4E+17	2,0E+17	1,7E+13
Multiplier (Be) OB+IB	298.6	3,3E+19	1,3E+18	1,1E+18	6,5E+17	5,1E+13
MANET, FW, OB	69.3	2,3E+19	1,2E+19	6,9E+18	6,2E+17	1,1E+14
MANET, FW, IB	43.9	1,3E+19	6,7E+18	3,8E+18	3,4E+17	6,5E+13
MANET, FW, OB+IB	113.2	3,5E+19	1,9E+19	1,1E+19	9,5E+17	1,8E+14
MANET, BZ, OB	408.1	3,1E+19	1,3E+19	6,4E+18	5,8E+17	1,9E+14
MANET, BZ, IB	188.7	1,9E+19	7,6E+18	3,8E+18	3,4E+17	1,2E+14
MANET, BZ, IB+OB	596.8	4,9E+19	2,0E+19	1,0E+19	9,2E+17	3,1E+14
MANET, SH, OB	676.1	4,7E+18	1,4E+18	5,2E+17	5,9E+16	4,9E+13
MANET, SH, IB	861.6	7,5E+18	2,4E+18	9,4E+17	1,0E+17	7,7E+13
MANET, SH; OB+IB	1537.7	1,2E+19	3,9E+18	1,5E+18	1,6E+17	1,3E+14
All materials OB	1420.4	8,2E+19	2,8E+19	1,5E+19	1,7E+18	3,9E+14
All materials IB	1199.0	5,0E+19	1,7E+19	8,8E+18	9,8E+17	2,8E+14
All material OB+IB	2619.4	1,3E+20	4,5E+19	2,3E+19	2,7E+18	6,7E+14

To elaborate on the effect of the tritium which is time dependent the total activity for the outboard plus inboard versus decay time is plotted in Figure 3.

For the breeder material both cases are shown, i.e., without any tritium and including the predicted amount of tritium retained in the breeder (10 g, Table 6). One can see that at 10 years after shutdown the small amount of tritium dominates the activity of the breeder by two orders of magnitude, although at a relatively low level compared to steel. At short decay times (<1d) and very long decay times (>100y) the influence of tritium is small.

For the multiplier, too, the curves without and with tritium are plotted, the latter being adjusted to the predicted tritium content in beryllium of 1278 g (Table 6). Here, the effect of tritium on the total activity reaches more than two orders of magnitude in the time interval 1 to 10 years after shutdown, approaching the curves for MANET pertaining to the shield and manifold zone.

For MANET the effect of tritium on the total activity is negligible, since the amount of tritium is small and the activity level is rather high which is dominated by the material in the first wall and in the breeding zone. So the total activity in all blanket materials amounts to 1.3×10^{20} Bq at time zero declining slowly for the first 10 years and thereafter more rapidly, irrespective of the tritium contained in the breeder and multiplier.

It is concluded that the total activity is dominated by the structural material (predominantly by that contained in the first wall and breeding zone). The activity in the breeder is typically 3 to 4 orders of magnitude lower and the activity of the multiplier ranges in between, depending on the amount of tritium trapped.

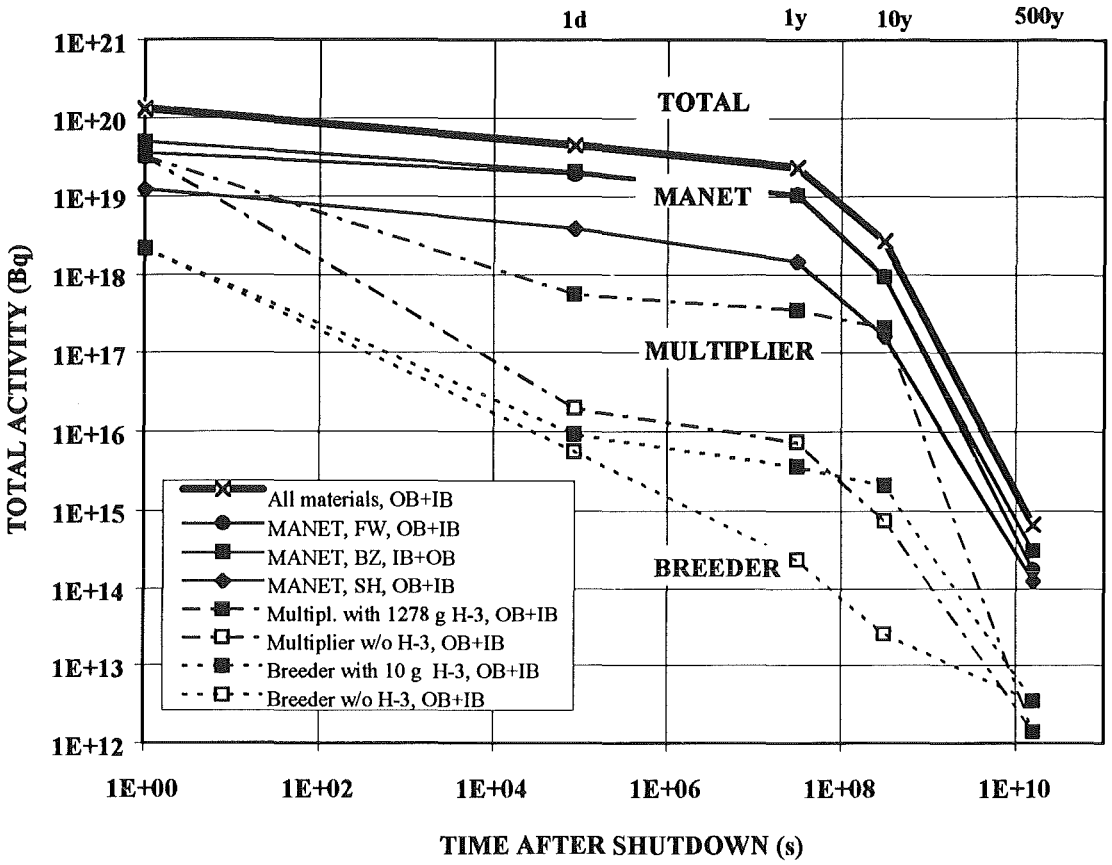

Figure 3: Total activity in blanket materials (tritium adjusted to predicted amount)

2.3.3 The dominating nuclides

To give an overview of the dominating radioactive nuclides with respect to total activity and gamma dose rate those nuclides which contribute with more than 10 % to the activity or dose rate are compiled in Table 8 for technically relevant time intervals, e.g., operation, accident management, maintenance and repair, decommissioning, and long term disposal. For the breeder material most of the dominating nuclides (for instance Na-24, Ar-37, Fe-55 for activity or Mn-54, Sc-46, Co-60 for gamma dose rate) are produced from impurities specified for the FISPACT calculations. The isotope Al-26, however, which is of interest with regard to the long term gamma dose rate results from both the aluminum content as initial impurity (two thirds) and from transmutation from silicon (one third). The latter fraction cannot be avoided.

In the multiplier the basic constituent is beryllium with a small addition of oxygen. Again a large number of specified impurities are responsible for the dominating nuclides (like Fe-55, Ni-63, C-14 for activity and Na-24, Mn-56, Co-58, Co-60, Ag-108 for gamma dose rate). The initial aluminum content is lower than in the breeder (about one sixth) and the silicon content is much lower (0.01 % vs. 23.2%) so that the Al-26 long term gamma dose rate ranks only second.

In the structural material MANET the nuclides of Fe, Mn, Ni, and Nb stem from alloying elements. Co is a specified impurity (0.02 %) but is to some extent (about 5 % of the initial amount) produced from Fe and/or Ni. Tc is a product of the alloying element Mo. The contribution of Al (initial impurity 0.05 %) is insignificant in this

composition. So a reduction of manganese by at least two orders of magnitude, cobalt by at least one order of magnitude, and niobium by perhaps two orders of magnitude would be effective means to reduce the gamma dose rate of MANET. On the other hand in the time intervals relevant for maintenance and decommissioning Fe-55 is the strongest and unavoidable contributor to the activity in ferritic steel.

TABLE 8: ACTIVITY AND GAMMA DOSE RATE DOMINATING NUCLIDES (SORTED ACCORDING TO THEIR IMPORTANCE)

Time Interval	Activity dominating nuclides			Gamma dose rate dominating nuclides		
	Breeder	Multipl.	Structure	Breeder	Multipl.	Structure
0 s Operation	Al-28	He-6	Mn-56 Fe-55	N-16 Al-28	N-16 He-6	Mn-56
10 s to 1 day Accident management	Si-31 Na-24	Fe-55	Mn-56 Fe-55	Na-24	Na-24 Mn-56 Co-58 Co-60	Mn-56 Mn-54 Co-60
1 day to 1 year Maintenance and repair	H-3 Na-24 Ar-37 Fe-55	H-3 Fe-55	Fe-55 Mn-54	Mn-54 Sc-46 Co-58 Zn-65	Co-58 Co-60 Mn-54	Mn-54 Co-60
1 year to 100 years Decommissioning	H-3 Fe-55 Ar-39	H-3 Fe-55 Ni-63	Fe-55 Ni-63 Nb-93	Co-60 Na-22 Al-26	Co-60 Ba-137	Co-60 Nb-94
> 100 years Long term disposal	C-14 Cl-36	Be-10 C-14	Nb-94 Tc-99 Nb-91 Ni-59	Al-26	Ag-108 Al-26	Nb-94

3 ENERGY SOURCES FOR MOBILISATION

Energy sources in upset or accidental conditions are seen in (a) plasma disruptions, (b) delayed plasma shutdown after cooling disturbances, (c) decay heat, (d) work potential for pressurised coolants, and (e) exothermic chemical reactions. This section summarises the energy sources for the BOT blanket based on the inventories described in 2. An overview of the energy sources (a) to (e) is given in Table 9. The values are elaborated in the subsections to follow.

TABLE 9: ENERGY SOURCES FOR MOBILISATION

Energy source	Value (GJ)
Plasma disruptions (localised)	≈1
Delayed plasma shutdown	11
Decay heat integrated over:	
1 minute	1.9
1 hour	76
1 day	370
1 month	3900
Work potential of helium coolant	
1 subsystem	11.2
all subsystems	35.3
Chemical energy potential	
beryllium/water reaction	12 000
beryllium/oxygen reaction	21 200

3.1 Energy liberated during plasma disruptions

Plasma disruptions can cause local evaporation of first wall material or mobilisation of adhesive dust. This is a common and unresolved problem of first wall protection and dust processing and is not considered here. The energy source is essentially the stored energy in the plasma which is typically of the order of 1 GJ and, hence, is small compared to other energy sources discussed in the following subsections.

3.2 Energy due to delayed plasma shutdown

Delayed plasma shutdown after a sudden cooling disturbance will bring any first wall to melt within seconds. The energy source is essentially the time integral of the fusion power from the beginning of the cooling disturbance to complete shutdown. For instance, for the reference plasma shutdown scenario (plasma is shutdown 1 s after cooling disturbance; at that time the internal heat source disappears instantaneously and the first wall surface heat flux decreases linearly down to zero in 20 s) the energy transmitted to the first wall is then 4.2 GJ and to the blanket 0.8 GJ per GW of fusion power (ignoring that a large fraction of the 4.2 GJ goes to the divertor). This scales up to 9.2 GJ to the first wall and 1.8 GJ into the blanket for a 2.2 GW reactor.

3.3 Decay heat

Decay heat is the governing feature in managing the cooling disturbances like LOCA, LOFA (Section 4) and, in particular, loss of site power or loss of heat sink. The decay heat has been assessed in the neutronics analyses [4] and is presented there as time dependent afterheat power density (per unit volume) and total afterheat per outboard and inboard blanket segment. Here we give additional information on specific afterheat (per unit mass), total afterheat, and adiabatic temperature rise in blanket domains referred to in this report, that is, in different blanket regions (inboard, outboard) and radial zones (first wall, breeding zone, manifold and shield zone) as specified in 2.1.

3.3.1 Specific afterheat

The specific afterheat in the materials (breeder, multiplier, and steel) and for steel in the blanket domains mentioned is plotted in Figure 4 as a function of time after shutdown. The curves represent average values taken over the entire radial zone (thickness of radial zones as defined in Table 1) in the outboard (central part plus upper part) and, respectively, in the inboard (central part plus upper and lower part).

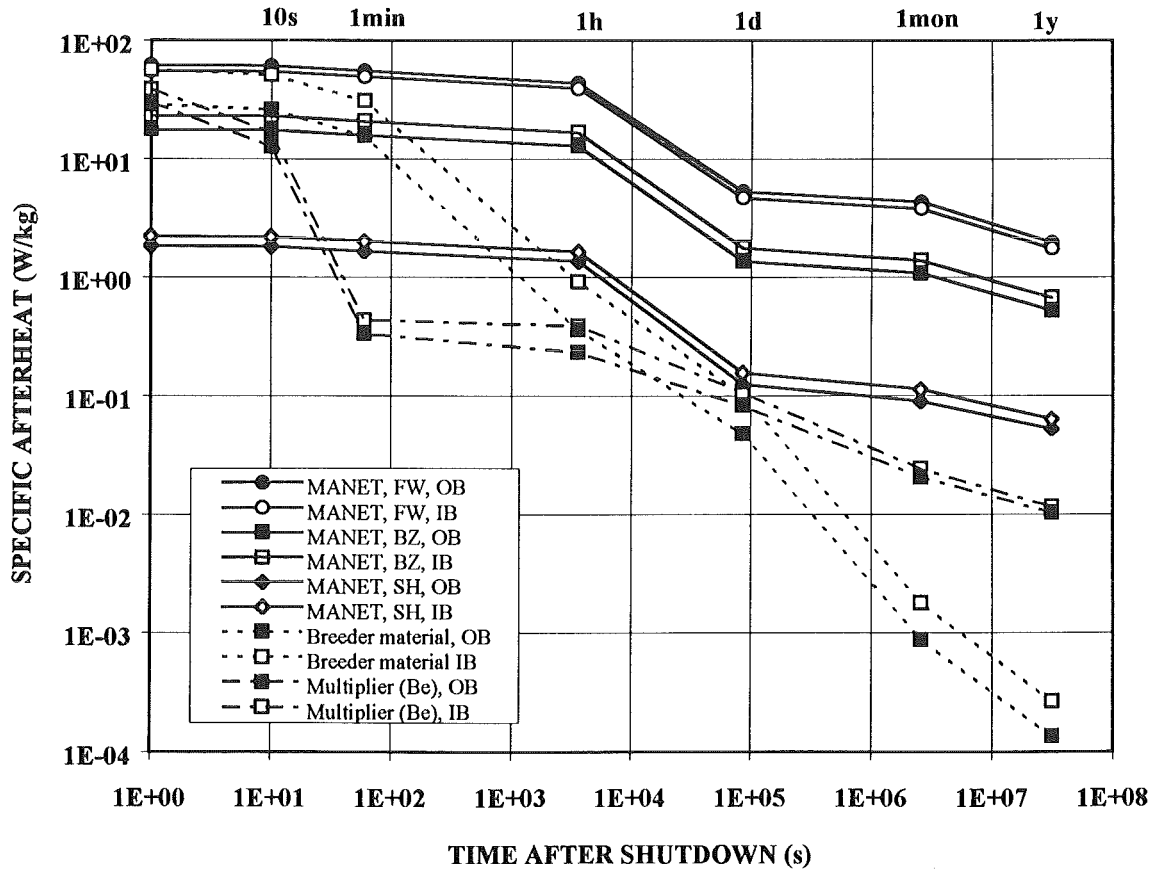

Figure 4: Specific afterheat in blanket materials in the outboard (OB) and inboard (IB) in different radial zones, i.e., first wall (FW), breeding zone (BZ), and manifold plus shield zone (SH).

At time zero (after 20000 hours of operation) the specific afterheat for the three blanket materials of the breeding zone ranges between 18 W/kg for the steel (curve MANET, BZ, OB) and 57 W/kg for the breeder material. Note that for all pairs of

curves for inboard and outboard, except for the steel in the first wall, the average specific afterheat in the inboard is higher by a factor of between 1.2 and 3 than in the outboard.

The time behaviour of the steel and the relation of the specific afterheat in the three radial zones is similar to that of the specific activity (Figure 2) with the typical decrease by one order of magnitude after 1 day. The specific afterheat in the breeder material starts out at a high level, however, after 1 minute it drops steadily in the log-log scale by five orders of magnitude within the first year. It does not include the contribution from the retained tritium which is insignificant. The multiplier steps down by two orders of magnitude within the first minute and then declines gradually staying below the specific afterheat of the steel in the manifold and shield zone.

In a similar way as the averaged values discussed above also the corresponding values at the blanket midplane have been plotted (not shown) and all the qualitative results presented apply in the same way, of course at a higher level. Values at the midplane are greater by typically a factor of 1.8 to 2.6 compared to the zone average values.

3.3.2 Total afterheat

The total afterheat of the blanket materials in the different regions and radial zones (Figure 5) has been evaluated by multiplying the average specific afterheat plotted in Figure 4 by the corresponding mass inventories listed in Table 7. The total afterheat in the blanket after decay times of 0s, 1h, 1d, 1 month, 1 year amounts to 34, 16, 1.7, 1.3, and 0.6 MW, respectively, 60 % of which is generated in the outboard and 40 % in the inboard. The afterheat is dictated by the structural material (MANET) and among the three radial zones the first wall contributes most with its relatively small mass fraction.

For thermal transient analyses on the time scale beyond a few minutes the contribution of the breeder and multiplier materials to the total afterheat can be neglected (Figure 5). As already discussed in 2.3.2 the curves for the breeder material do not contain the contribution from the trapped tritium, whereas the curves for the multiplier do include the full tritium produced in the beryllium (assuming zero release). As 1 g of tritium produces an afterheat of 0.327 W it becomes evident that the amounts of tritium predicted to be retained in the materials (Table 6) can be neglected in this context.

From the curves for the total afterheat in the entire blanket (Figure 5, curve labelled all materials OB+IB) we can estimate the decay energy integrated over typical time intervals of relevance in safety discussions in comparison to other energy sources for mobilisation. This accumulated decay energy amounts to 1.9 GJ after 1 minute rising up to 3900 GJ after 1 month (Table 9).

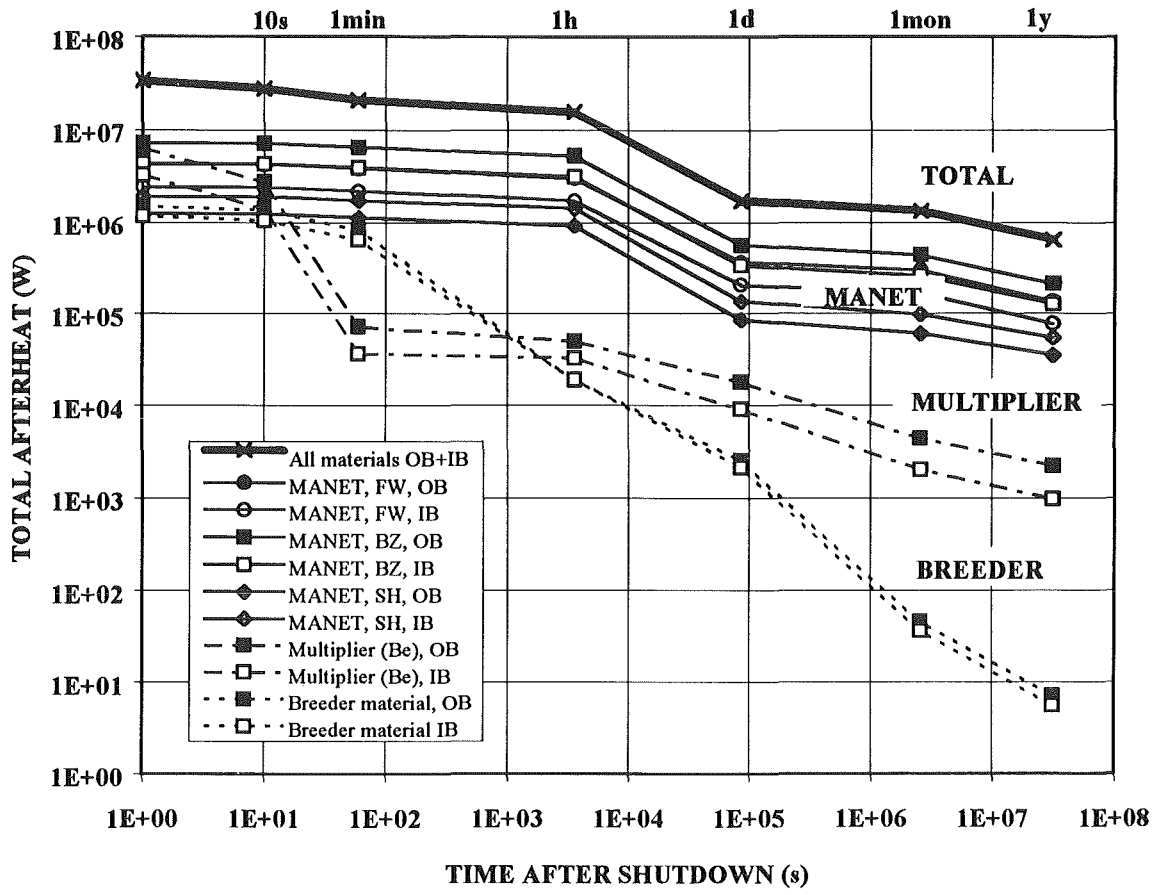

Figure 5: Total afterheat in blanket materials in the outboard (OB) and inboard (IB) in different radial zones, i.e., first wall (FW), breeding zone (BZ), and manifold plus shield zone (SH).

3.3.3 Thermal inertia

The heat capacity relative to the specific afterheat of a given material or a mixture of materials represents the thermal inertia of a given system. High thermal inertia limits temperature transients and amplitudes in upset or faulted conditions and, thereby, contributes inherently to limit potential consequences for primary initiating events and eases plant control. A measure for the thermal inertia (more precisely, of its inverse value) is the adiabatic temperature evolution of the isolated blanket or of portions of the blanket, like the radial zones or individual materials. It can be described by equation 4

$$\frac{dT}{dt} = \frac{h(t)}{c_p(T)} \qquad (4)$$

where T = mean temperature of the system

h(t) = mean specific afterheat of the mixture (weighted by mass fractions) as function of time, t

$c_p(T)$ = mean heat capacity of the mixture (temperature dependent and weighted by mass fractions)

The resulting temperature vs. time curves are plotted in Figure 6 for various portions of the outboard blanket as indicated in the figure legend. In computing equation 4 the average specific afterheat according to Figure 4 was used for individual materials.

The transient for the fictitious isolated first wall is the steepest, reaching a temperature rise of 237 K in the first hour. In contrast, the isolated manifold and shield zone (without removable shield) would warm up very slowly (about 40 K in the first day). Even smaller is the heat-up of the breeder material (7 K/day) and of the multiplier (4 K/day). For a mixture of the whole outboard blanket the temperature rise amounts to 137 K for the first day, levelling off to, e.g., 53 K/day at the end of the first 24 hours. Likewise the curve for the isolated breeding zone (mixture of steel, breeder material, and multiplier) is plotted, which is similar to the former.

The same exercise has been done for the inboard and also for an isolated slice of material mixture taken from the blanket midplane, using the specific afterheat histories at midplane which are higher as discussed earlier. Typical results are tabulated in Table 10 for the four cases.

In general, one can conclude that the adiabatic temperature transients for isolated portions of the blanket are high for the first wall, moderate for the mixture and very low for the breeder material and multiplier. Temperature transients for the inboard are higher than for the outboard (except in the first wall). Please note that the temperature rise in the first wall can never reach the values quoted because of the thermal coupling with the breeding zone. It may rather approach the curve obtained for the mixture of the whole blanket.

TABLE 10: ADIABATIC TEMPERATURE RISE IN DIFFERENT PORTIONS OF THE BLANKET (in K after 1 day; in the first line in K after 1 hour)

Blanket material and zone	In total radial zone		In a slice at midplane	
	Outboard	Inboard	Outboard	Inboard
MANET, FW (after one hour)	237	214	475	426
MANET, BZ	358	459	836	1030
MANET, SH	41	50	74	106
Breeder material, BZ	7	17	19	25
Multiplier, BZ	4	6	9	11
Total blanket (FW+BZ+SH)	137	132	296	280
Total BZ (all materials)	116	162	269	355

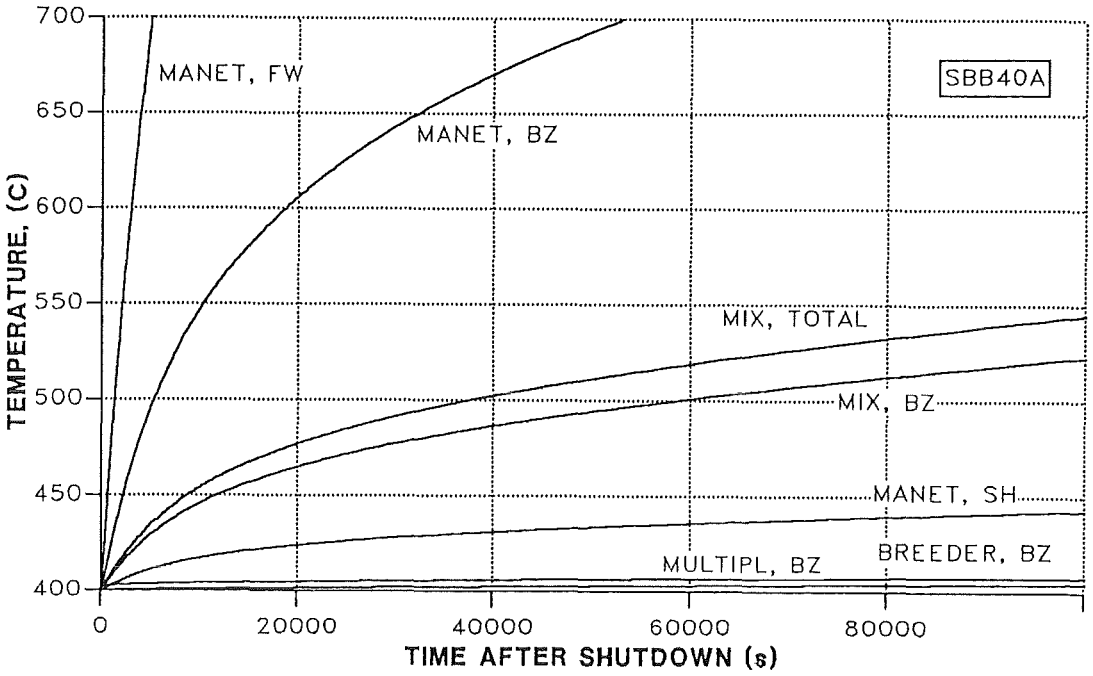

Figure 6: Adiabatic temperature evolution of blanket materials and mixtures of materials in isolated radial zones of the outboard blanket.

3.4 Work potential of helium coolant

The work potential of the helium coolant, the release times in case of a double-ended pipe break, and the momentum forces of the outflowing gas stream have been assessed analytically and by use of a simple RELAP model [5].

The main helium cooling system contains 22000 kg of helium at 8 MPa and inlet/outlet temperature of 250/450 °C, i.e., a mean temperature of about 350 °C. The work potential relative to ambient conditions for the total helium inventory is 35 GJ. A more realistic scenario is the adiabatic expansion of the helium inventory from one outboard cooling subsystem, comprising six primary loops with an inventory of about 7000 kg (see 2.2). The corresponding work potential would be 11 GJ. Upon release this helium would pressurise the vacuum vessel (assuming a free volume of 5000 m^3) in the event of an in-vessel pipe rupture to 1.5 MPa. This is beyond the expected design pressure and, hence, would require an extra expansion volume. Assuming 0.2 MPa as an absolute end pressure, the required expansion volume would be 70000 m^3.

The release times for the helium of one of the two independent outboard cooling subsystems in the case of a double-ended break in pipe sections of different sizes were investigated. Analytical computations assuming reversible adiabatic expansion of the pressurised helium yielded draining times of 1.8/2.9/46 s for inner pipe diameters of 1080/540/300 mm which are typical for the hot leg, circulator bypass, and feeders, respectively.

To obtain an estimate of the effects of wall friction and heat transfer from the pipe walls to the helium during outflow, a simple RELAP model was established. The simulated pipe lengths were 100/100/30 m for the pipe diameters mentioned above. These are chosen as example with regard to the envisaged cooling circuits.

Computations with this model resulted in draining times of 2.5/4.3/72 s for the respective diameters. The longer release times are due to dissipation of kinetic energy by friction and due to decreasing density of the outflowing gas stream by heat supply from the pipe walls. Figure 7 shows the depressurization of one of the two outboard first wall cooling subsystems after a double-ended break in either the hot leg, circulator bypass, or in one feeder.

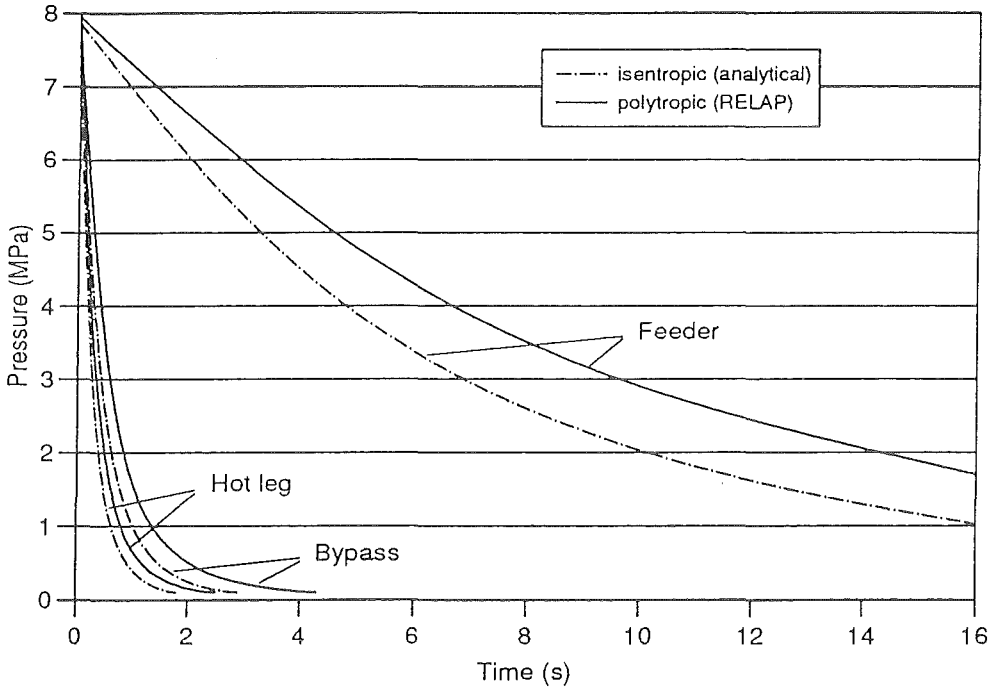

Figure 7: Depressurization of one of the two outboard cooling subsystems. Curves refer to a double-ended pipe break in either the hot leg (1080 mm), circulator bypass (540 mm), or in the feeder (300 mm).

The momentum forces of the gas stream immediately at the beginning of the discharge were calculated according to the mass flows and velocities obtained with the RELAP model to 4220/1140/250 kN per side for the pipe diameters of 1080/540/300 mm.

The minimum release time for the helium in case of a guillotine break of a single cooling channel in the first wall (cross sectional area 14 mm x 18 mm, reversible adiabatic expansion) was estimated to 3.4 hours. This long time span is due to the small cross sectional area of a first wall cooling channel which limits the outstreaming mass flow to some kilograms per second.

It should be noted that there exists a potential for the release of the helium from both cooling subsystems at the same time into the vacuum vessel in case of a crack in a segment box extending over adjacent cooling channels, each one pertaining to one cooling subsystem (see 5.2). Then the end pressure and expansion volume discussed above would be doubled if the failure occurred in an outboard first wall. A similar occurrence in the inboard first wall would have less consequences because of the smaller helium inventory.

3.5 Exothermic chemical reactions

The largest chemical energy potential results from the vast amount of beryllium multiplier (≈300 tons). The exothermic reaction per ton of beryllium with water or oxygen generates 40 GJ or 67.4 GJ, respectively, resulting in a total chemical energy potential of 12000 GJ for a beryllium/water reaction and 21200 GJ for a beryllium/air reaction. This corresponds to between 100 and 300 GJ per blanket segment (Table 11).

It must be emphasised that the beryllium pebble beds are encapsulated in approximately 150 separate chambers per segment, so that the accessibility of the beryllium to chemical agents is small, as is its mobility (compare 5.2).

TABLE 11: CHEMICAL ENERGY POTENTIAL PER BLANKET SEGMENT (GJ)

Chemical reaction (at 600 °C)	Outboard	Inboard
Beryllium/water reaction	179	105
Beryllium/air reaction	301	177

4 FAULT TOLERANCE

The following analyses of temperature transients during a loss of coolant and loss of flow accident and of the temperature profile during handling without active cooling have been performed in order to get a first idea of whether the BOT blanket is tolerant in accidental or abnormal conditions.

4.1 Loss of coolant accident

Temperature transients in the first wall of the outboard blanket were analysed for a loss of coolant accident (LOCA) [6]. The LOCA sequence has been defined as a working hypothesis for the BOT blanket as follows: Coolant flow is interrupted instantaneously at time zero in one of the two independent helium subsystems. Plasma shutdown is assumed at t=1 s. At the same time the internal heat source in the blanket disappears instantaneously and the first wall surface heat flux decreases linearly down to zero within 20 s.

The calculations were performed in two steps. In the first step a rather coarse three-dimensional mesh was used to compute the transient helium temperatures along typical coolant channels, applying the average value of the surface heat flux of 0.4 MW/m^2. In the second step a finer two-dimensional mesh was used to compute the temperature distribution in the structure. Here the local peak surface heat flux of 0.5 MW/m^2 occurring at the blanket midplane was taken into account. Since it was found in the steady state calculations [7] that for this concept the heat exchange between the first wall and the breeding zone cannot be neglected, the models contain both a representative part of the first wall and of the adjacent breeding zone consisting of the attached cooling plate and portions of the pebble beds (Figure 8). The boundary of the model representing the beryllium pebble bed has been chosen such that the temperature contour lines are approximately normal to the boundary which means zero heat flux as boundary condition.

The temperature history at three selected points of the first wall during the transient is plotted in Figure 8. The maximum temperature is reached at the edge of the segment in the equatorial midplane, opposing the failed cooling channel (point A). Its peak value is 590 °C (70 K above the steady state temperature) occurring 10 s after LOCA initiation. Temperatures in the other points next to the rib (point B) and opposite to the intact cooling channel (point C) immediately start to decrease.

A transient stress analysis has not been carried out so far. The yield strength and the ultimate tensile strength of the structural material MANET at the peak temperature (590 °C) are reduced by 20 % compared to the values at operating temperature (520 °C). The stress analysis for steady state conditions [7] resulted in stresses well below the admissible values according to the ASME code. It can be assumed that the primary stresses do not increase significantly during the transient and that the secondary stress amplitudes occur locally. Hence, the short-term temperature transients are not expected to endanger the integrity of the structure. However further investigations are needed.

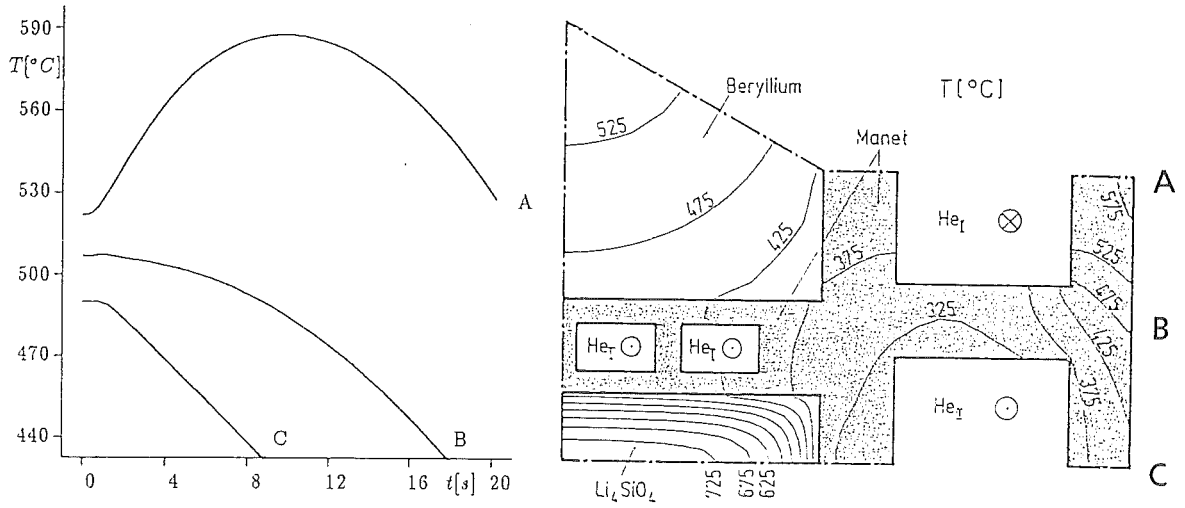

Figure 8: Temperature history at three points of the first wall during a LOCA (left) and temperature distribution across the first wall plus breeding zone model at t=10 s (right).

4.2 Loss of flow accident

Temperature transients in the first wall of the outboard blanket were analysed for a loss of flow accident (LOFA) in a similar way as described in 4.1 for the LOCA, using the same geometrical model shown in Figure 8 [6]. For the LOFA sequence it has been assumed that the mass flow rate in both cooling subsystems decreases linearly down to 1 % of its nominal value within 4 s after the start of the accident at t=0 s. The history of the heat loads is the same as in the case of a LOCA.

The temperature history at three selected points of the first wall during the transient is plotted in Figure 9. The maximum temperature is reached at the edge of the segment in the equatorial midplane, opposing the failed cooling channel with the higher initial temperature (point A). Its peak value is 565 °C (45 K above the steady state temperature) occurring 14 s after LOFA initiation. Temperatures in the other points next to the rib (point B) and opposite to the other failed cooling channel (point C) show a similar behaviour, namely: until about 3 s the shut-off internal heat load leads to a temporal temperature drop. Then the surface heat flux, declining slower than the mass flow rate, causes the temperature hump at t=14 s.

Also shown in Figure 9 is the history of the coolant temperature during the first 60 s at the inlet, at the centre, and at the outlet of the plasma facing part of the first wall. One can see that at t=14 s the coolant temperatures are still rising. The fact that the first wall temperatures at this time start to drop is caused by further decreasing surface heat flux and by temperature balance within the model. It is to be noted that the subsequent temperature development in the first wall and in the whole blanket depends on the afterheat production discussed in 3.3 and on the ascribed cooling conditions (natural circulation, heat dissipation to surrounding structures). Those long term transients have still to be assessed. They can exceed the short term temperature peaks regarded here.

The generated thermal stresses have not been assessed. They are smaller than in the case of LOCA due to the smaller temperature amplitudes and gradients.

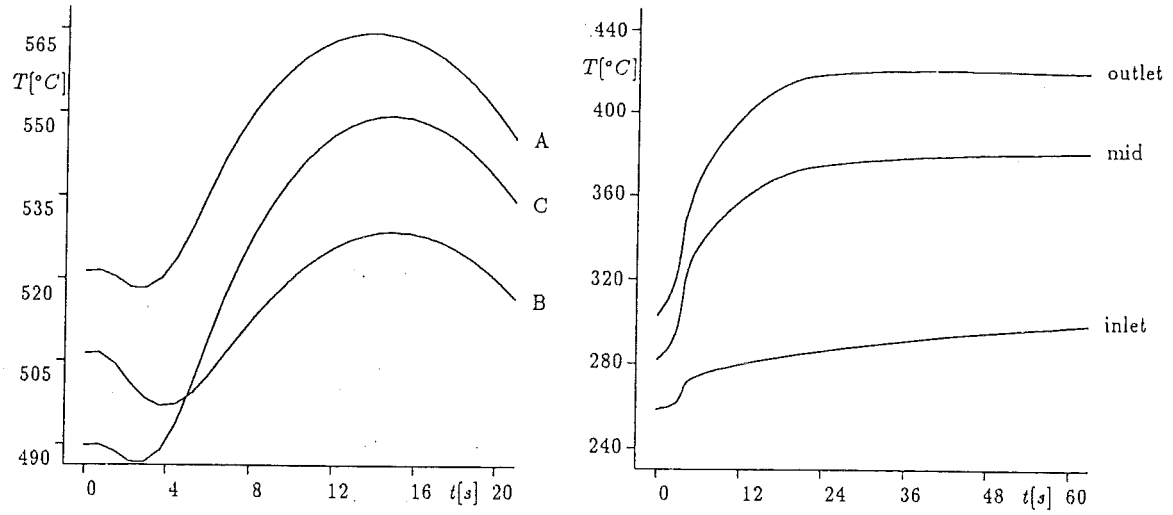

Figure 9: Temperature history at three points (compare Figure 8) of the first wall during a LOFA (left) and history of the coolant temperature in the plasma facing part of the first wall (right).

4.3 Temperature distribution in a segment during handling

Within the blanket concept selection exercise performed for the four European blanket concepts the question arose, whether active cooling of the blanket segments during handling is required or not. To answer this question for the BOT solid breeder blanket, calculations with the finite element code FIDAP [8] were performed.

The calculations were performed using the model shown in Figure 10. It is a poloidal-radial cut through the toroidal midplane of an outboard blanket segment. In poloidal direction the model is bounded by cuts through the centre of the beryllium and Li_4SiO_4 pebble beds, respectively, which were assumed to be adiabatic boundaries. In radial direction the model includes the first wall, the breeding zone with one cooling plate, and the manifolding zone consisting in the simplified model of an air filled gap and the back plate. Heat is only dissipated by convection and radiation from the front side, whereas the side walls as well as the rear side are set as adiabatic boundaries. The heat transfer within the pebble beds (both conduction and heat resistance at pebble bed/solid structure interfaces) was modelled based on correlations developed in [9]. In the air filled gap conduction and convection by natural circulation in a closed vertical cavity [10] was considered. As heat sources the radial profiles of afterheat generated in the different materials one day after shutdown were used according to the neutronics analyses [4].

Two calculations were carried out. In the first case heat transfer across the front wall due to convection and radiation (emissivity =1) was considered. In the second case only convection was taken into account. The equilibrium temperature profile in radial direction for the first case is shown in Figure 10. It ranges from 300 °C at the front side to 420 °C at the rear side, involving a significant temperature jump of about 40 K in the air filled gap. In the second case the temperature profile looks similar, however due to the neglect of the heat losses by radiation, it is shifted to a higher level, that is, the temperature would now range between 740 °C and 830 °C.

The analysis shows that the afterheat generated in an outboard segment one day after shutdown can be dissipated by convection and radiation via the front wall alone in an air environment of 30 °C, without exceeding acceptable temperature limits in the structure (case 1). The comparison with case 2 shows that the contribution from radiation is substantial and must not be neglected in the assessment. The analysis implies a number of conservative assumptions, i.e., the adiabatic side walls, the adiabatic back wall, and the air filled gap which will more likely be filled with helium. On the other hand, the emissivity will perhaps be much less than unity. Overall, active cooling of a segment during handling will not be required by all means, although it would ease the temperature control in terms of profile and level.

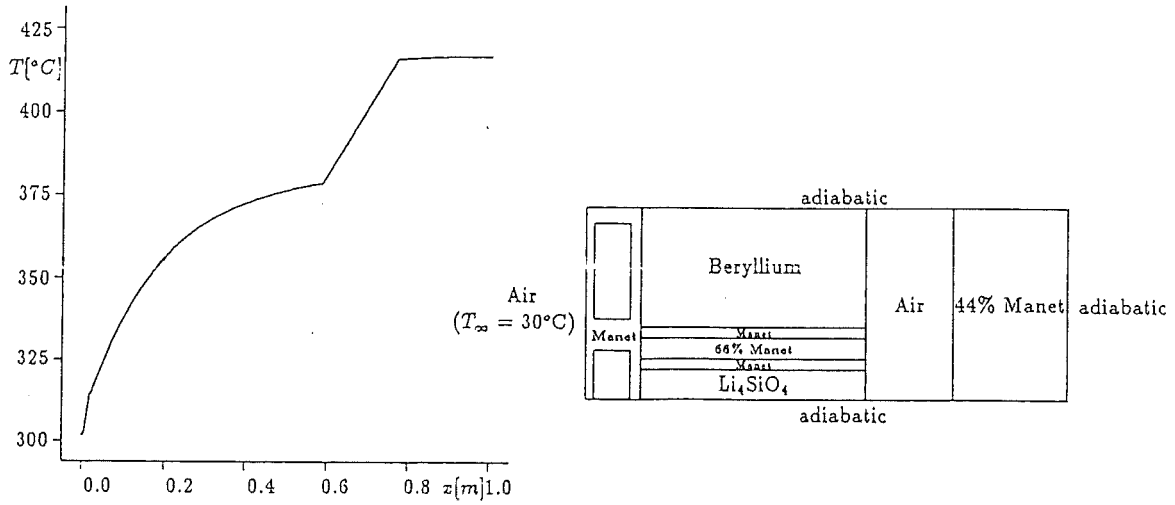

Figure 10: Radial temperature profile in the outboard blanket segment during handling. Heat is dissipated at the front wall by convection and radiation (left). Radial-poloidal model used in FIDAP calculations (right).

5 TRITIUM AND ACTIVATION PRODUCTS RELEASE

5.1 Release during normal operation

In normal operation the total release of tritium and activation products to the environment should not exceed 1 TBq/d as a guide line. The release is expected to be dominated by the effluents arising from the primary helium cooling loops via permeation in the steam generators into the secondary water/steam cycle. For these discharges an upper limit of 20 Ci/d (0.74 TBq/d) has been estimated [1] based on the assumptions made in [11] (incoloy 800 steam generator tubes with a barrier factor of 20). This leaves little margin for other effluents, for instance originating from coolant leakage (into the containment and from there via the plant air detritiation system to the environment), from the purge gas system, and from other fuel cycle equipment. Those have not been assessed here, but conservative estimates performed in the safety and environmental assessment of fusion power (SEAFP) indicate that they may also range up to close to 1 TBq/d so that the total release would exceed the 1 TBq/d goal. Further assessment is needed in this area.

5.2 Tritium release in accidental situations

For a first judgement on the radiological hazard potential of the tritium an assessment has been made on the early dose and chronic dose to the most exposed individual at a distance of 1 km from the point of release caused by the tritium inventory. Part of the tritium will escape into the vacuum vessel and from there to the environment in the case of a postulated severe loss of coolant accident (rupture of the first wall involving both cooling subsystems). In this section the tritium release from the primary coolant, from the purge gas, and from the ejected amount of beryllium pebbles is considered and the radiological consequences are discussed.

The accident scenario postulated which could be considered as an enveloping design basis accident is defined as follows:

- Rupture of the first wall of one outboard segment in toroidal direction across the whole segment width and the full first wall depth
- Escape of primary helium from both subsystems into the vacuum vessel
- Ejection of multiplier from the affected segment chamber of 45 mm height (≈40 kg of beryllium) into the vacuum vessel
- Venting of the vacuum vessel into the expansion volume, end pressure <0.2 MPa
- Escape of total (original) purge gas into vacuum vessel and expansion volume and refill of purge gas system to vacuum vessel end pressure
- Air or water ingress into vacuum vessel
- Chemical reaction of ejected multiplier and release of tritium into vacuum vessel and expansion volume
- Release of tritium from breeder and multiplier pebbles from the whole outboard blanket (except affected chamber) into purge gas due to temperature transients; flushing of a fraction of the released tritium into the vacuum vessel before isolation of affected segment

- Isolation of affected segment from purge gas system within about 1 hour
- Afterheat removal from outboard blanket (including the affected segment) by main cooling systems at low pressure.

The amount of tritium escaping into the vacuum vessel and expansion volume is summarised in Table 12.

TABLE 12: TRITIUM ESCAPING INTO THE VACUUM VESSEL IN CASE OF A POSTULATED FIRST WALL RUPTURE

Tritium carrying system or component	Total tritium inventory (g)	Fraction escaping into Vac. Vessel	Tritium escaping into Vac. Vessel (g)
Primary coolant (helium)	0.46	0.62	0.29
Purge gas (helium)	0.1	1	0.1
Multiplier ejected (beryllium)	0.4	1	0.4
Multiplier in affected segment	20	0.05	1
Multiplier in remainder blanket	1258	0.003	3.8
Breeder in affected segment	0.15	0.5	0.08
Breeder in remainder blanket	10	0.05	0.5
Total			**6.2**

The following explanations are given:

- The primary coolant has a total tritium inventory of 0.46 g (Table 6) in 22000 kg of helium. The two subsystems affected contain 62 % of the total inventory, that is 0.29 g of tritium in about 14000 kg of helium.
- The purge gas carries in steady state a tritium inventory of approximately 0.1 g (2.3.1) in a single and interconnected system for the whole blanket. It is assumed that all the tritium is released into the vacuum vessel.
- The ejected beryllium from one chamber (40 kg) contains about 0.4 g of tritium. Due to the postulated chemical reaction all the tritium is assumed to be liberated into the vacuum vessel.
- The tritium retained in the beryllium of the affected segment amounts to about 20 g. It is assumed that during the temperature transient 5 % of the tritium (3 % corresponding to the initial burst predicted in [2] plus modelling uncertainties) will be released into the purge gas and from there into the vacuum vessel.
- The tritium inventory in the beryllium of the remainder of the blanket amounts to 1258 g (1278 g according to 2.3.1 minus 20 g). It is assumed that during the accident 3 % of the tritium content is released from the beryllium (according to the initial burst) into the purge gas and 10 % of that released fraction has escaped into the vacuum vessel before the isolation of the affected segment from the purge gas system was achieved.

- The tritium retained in the breeder material of the affected segment amounts to about 0.15 g. It is assumed that during the temperature transient 50 % of the tritium will be released into the purge gas and from there into the vacuum vessel.
- The tritium inventory in the breeder material of the remainder of the blanket amounts to 10 g (Table 6). It is assumed that during the accident 50 % of the tritium content is released from the breeder pebbles into the purge gas and 10 % of that released fraction has migrated into the vacuum vessel before the isolation of the affected segment from the purge gas system was achieved.

Under this supposition the total amount of tritium escaping into the vacuum vessel and expansion volume as gaseous effluent accumulates to 6.2 g, 84 % of which being liberated from the beryllium alone. Most of the tritium will be released from the breeder and multiplier in the form of HT. However a large fraction is expected to be chemically reduced to HTO at the steel walls before leaving the blanket. So for the assessment of the dose rate to the public it would be conservative to assume that all the tritium were present in the form of HTO, the more dose effective form compared to HT.

To compute the early dose and the chronic dose to the most exposed individual at a distance of 1 km from the point of release, credit can be taken of a containment retention factor, $f_R<1$, saying that only a fraction f_R of the tritium escaped into the vacuum vessel and expansion volume will be released to the environment. Moreover, for the early dose (ED, 7 days exposure plus 50 years integration, no ingestion) a conversion factor of 0.26 mSv/g-T released to the environment and for the chronic dose (EDE, 50 years exposure plus 50 years integration, with ingestion) a conversion factor of 1.6 mSv/g-T are used according to [12] for a first assessment. These factors vary with the assumptions made for the release scenarios by about 3 orders of magnitude as investigated in [13] and the chosen values range close to the upper bound (of 1 mSv/g-T for ED and 4 mSv/g-T for EDE). Applying these conversion factors and a containment retention factor of $f_R=0.01$ we obtain with the total tritium of 6.2 g from Table 12 (and hence 0.062 g released to the environment) the following doses:

Early Dose = 0.016 mSv
Chronic Dose = 0.1 mSv

Thus, the predicted early dose and chronic dose caused by tritium release in this conservative accident scenario are low. The tolerable dose limits on the other hand are still under discussion and the spectrum of national regulations is rather wide. Recent recommendations discussed in ITER for various classes of event sequences range from 0.1 mSv/a for likely sequences ($f<10^{-2}$/a) over 5 mSv/event for unlikely sequences (10^{-2}/a > f > 10^{-4}/a) to 5-50 mSv/event for extremely unlikely sequences (10^{-4}/a > f >10^{-6}/a), where f is the rate of occurrence. In the SEAFP study [14] a value of 100 mSv/event is suggested as a yardstick with the understanding that doses of this level trigger the consideration of evacuation but do not mandate it.

5.3 Activation products release

Activation products release to the environment has not been assessed for the BOT blanket. The set of accident scenarios investigated in SEAFP [14] show that the maximum early dose to the most exposed member of the public are rather small. It ranges from 1.5E-7 mSv to 6.6E-4 mSv for the design basis accidents (DBA) and

from 7.5E-7 to 2.4E-1 mSv for the postulated beyond design basis accidents (BDBA). Corresponding values for the maximum chronic dose are higher by up to more than two orders of magnitude, viz. 2.6E-6 to 1.3E-1 mSv for DBAs and 3.7E-5 to 21 mSv for BDBAs. It is to be emphasised that in all the cases considered in the SEAFP study the largest contribution to the dose values stated originate from source terms which are not blanket concept specific (namely dust aerosols from beryllium or tungsten as first wall protection material), or from aerosols in the water coolant which is not applicable to the BOT concept. Hence, a similar assessment performed for the BOT blanket as was done in SEAFP would perhaps result in early doses much less than 1 mSv and in chronic doses also below 1 mSv. Again, most of these doses are expected to result from the first wall protection material, especially in the case of tungsten.

In conclusion, the activation products release phenomena need further analyses. Results from the SEAFP study suggest that the dose levels may range in the same order of magnitude as obtained in 5.2 for tritium release, but all being well below tolerable dose limits referred to in 5.2.

6 WASTE GENERATION AND MANAGEMENT

Only the decommissioning waste (no operational waste) is considered here. The masses and volumes for the different blanket materials produced by one complete set of blankets with the dimensions defined in Table 1 are given in Table 13, along with the total radioactive inventory and typical afterheat power densities at different decay times. The data have been compiled from the FISPACT calculations described in [4]. The volumes and power densities quoted refer to compact materials without any void volume. For the breeder and multiplier material the activity and power density are given for two cases, viz. a) without any tritium and b) including the predicted amount of tritium according to Table 6. This is because the influence of the tritium is substantial and for decommissioning purposes it is conceivable to remove most of the tritium from the Li_4SiO_4 and beryllium by baking.

TABLE 13: RADIOACTIVE INVENTORY AND AFTERHEAT IN ONE SET OF BLANKET SEGMENTS AT DIFFERENT DECAY TIMES (20000 HOURS OF FULL POWER OPERATION, ROUNDED NUMBERS)

Blanket Material or Zone	Total Mass	Compact Volume	Total Radioactivity in Bq after				Decay Heat (kW/m^3) after	
	(tons)	(m^3)	1 y	10 y	100 y	10^5 y	1 y	10 y
Breeder incl. 10 g of tritium	75	31	4E+15	2E+15	3E+13	8E+09	8,7E-04	3,3E-04
Breeder excl. tritium	75	31	3E+14	4E+13	1E+13	8E+09	5,1E-04	1,8E-04
Multiplier incl. 1278 g of tritium	300	161	4E+17	3E+17	2E+15	6E+11	5,4E-03	2,2E-03
Multiplier excl. tritium	300	161	5E+15	5E+14	3E+12	6E+11	1,9E-03	4,3E-04
Structural material total	2250	292	2E+19	2E+18	1E+15	2E+13	1,4E+00	1,5E-01
First wall incl. tritium	114	15	1E+19	1E+18	4E+14	4E+12	1,2E+01	6,0E-01
Breeding zone incl. tritium	600	78	1E+19	9E+17	6E+14	9E+12	3,5E+00	3,4E-01
Manifold&Shield zone incl. H-3	1540	200	2E+18	2E+17	3E+14	4E+12	4,0E-01	7,8E-02

The total activity of the 75 tons of breeder material is the lowest of all materials, ranging from 3E14 Bq after 1 year to 8E9 Bq after 10^5 years, if all the tritium were removed. Rests of tritium would increase the activity by up to two orders of magnitude for the first decades. The afterheat power density is also very low (<0.9 W/m^3 after 1 year of decay) as is the gamma dose rate (<10^{-3} Sv/h in the 50 - 100 years time scale [4]). As already outlined in 2.3.3 most of the dominating nuclides are produced from impurities, but the very long term gamma dose rate seems to be determined by Al-26, being both a product of the original aluminum impurity (0.158 % by weight) and from transmutation of silicon.

The 300 tons of beryllium multiplier carry an activity of between 5E15 Bq after 1 year to 6E11 Bq after 10^5 years, if all the tritium were removed. With the predicted amount of 1278 g of tritium the activity is higher by up to 3 orders of magnitude in the first century after shutdown. Hence, the actual activity will be determined by the degree of detritiation that can be achieved. The afterheat one year after shutdown amounts to a few W/m^3 only. The long term gamma dose rate is governed by the nuclides Ag-108 and Al-26. It ranges around 3E-5 Sv/h, depending on the initial impurities specified. However, there is a substantial amount of alpha emitters (on the order of

5E+5 Bq/g after 10 years) coming from the specified uranium impurity (0.011 weight -%).

The total amount of radioactivity in the blanket structure (MANET) sums up to 2E19 Bq after 1 year for all outboard and inboard segments, declining by 4 orders of magnitude within 100 years (Table 13). Approximately one half of the activity resides in the small volume of the first wall (15 m^3) and only about 10 % in the large volume of the removable shield (inboard only) and manifold zone. Accordingly, the averaged afterheat power density at 1 year varies between 12000 W/m^3 in the first wall and 400 W/m^3 in the manifold and shield zone. The gamma dose rate after 100 years varies in the range of 0.1 Sv/h for the first wall to 0.01 Sv/h for the manifold and shield material.

There is no universal classification scheme for radioactive waste available, since allowable levels will depend on the actual radioactive waste management option and the properties of individual radionuclides. However, based on current international practice IAEA has prepared a safety guide [15] which outlines typical characteristics of waste classes as an orientation. To give an idea of what these classes are and how the different material groups would fit in, Table 14 summarises typical activity and afterheat data for a decay period of 10 years. The bottom line of the table marks the IAEA interpretation of the boundary between high level waste (HLW) and low and intermediate level waste (LILW) [15]. For the LILW there is a subdivision between short lived waste (LILW-SL) and long lived waste (LILW-LL). The LILW-SL has a limitation of long-lived alpha-emitting radionuclides to 4000 Bq/g in individual waste packages and may require either near surface or geological disposal facilities. The LILW-LL and of course the HLW require geological disposal.

TABLE 14: ACTIVITY AND AFTERHEAT IN STRUCTURAL MATERIAL AFTER 10 YEARS OF DECAY AND IAEA WASTE CLASSIFICATION BOUNDARIES

	Activity (TBq/m^3)	Afterheat (kW/m^3)	Activity (Bq/g)	
			Total	from alphas
Breeder incl. 10 g of tritium	7E+01	0.0003	3E+07	≈ 0
Multiplier incl. 1278 g of tritium	2E+03	0.002	1E+09	5E+05
First wall incl. tritium	6E+04	0.6	8E+09	≈ 0
Breeding zone incl. tritium	1E+04	0.3	2E+09	≈ 0
Manifold&Shield zone incl. H-3	8E+02	0.08	1E+08	≈ 0
IAEA HLW/LILW boundary interpretation	≈ 5E+04	≈ 2	for alpha emitters LILW-SL: <4E+03 LILW-LL: >4E+03	

Applying this waste classification scheme one can draw the following conclusions:

- The breeder has a specific activity and afterheat well below the HLW/LILW boundary (by 3 and 4 orders of magnitude, respectively) and can be classified as LILW. Since there are practically no alpha emitters it is supposed to be qualified as short lived waste (LILW-SL).
- The multiplier has a specific activity one order of magnitude and an afterheat 3 orders of magnitude below the HLW/LILW boundary and can also be classified as LILW. There are substantial alpha emitters which call for a qualification as long lived waste (LILW-LL).
- The first wall structural material is close to the HLW/LILW boundary in terms of the specific activity which further decays substantially in the next decades. Also the afterheat is below the limit and according to FISPACT calculations there are no alpha emitting nuclides, so that formally the classification as LILW-SL would be fulfilled. The long term gamma dose rate may call for a classification as long lived waste.
- The activation data of the majority of the structural material from the breeding zone and particularly from the manifold and shield zone are well below the HLW/LILW boundary and again there are no significant alpha emitters. The problem of gamma dose rate is somewhat alleviated compared to the first wall material but not decisively.
- All material groups will require shielding during handling and transportation, since their gamma dose rate exceeds the frequently used hands-on limit of 2.5E-5 Sv/h by several orders of magnitude for decades

7 SUMMARY AND CONCLUSIONS

This report summarises the safety considerations for the Karlsruhe breeder outside tube (BOT) type solid breeder blanket, performed essentially in the frame of the European blanket concept selection exercise. It covers the topics (a) blanket materials and toxic materials inventory, (b) energy sources for mobilisation, (c) fault tolerance, (d) tritium and activation products release, and (e) waste generation and management. The assessment is based on a cooling system layout comprising a total of 18 circuits which are arranged in two 2-out-of 6 subsystems for the outboard plus two 2-out-of 3 subsystems serving the inboard. The terms 2-out-of-6 and 2-out-of-3 mean that two circuits of a subsystem must fail for the plant to be unavailable. The results are summarised below, conclusions with regard to further investigations needed are drawn in 7.2.

7.1 Summary of results

Blanket materials and toxic materials inventory: The total blanket volume (all inboard and outboard segments, full toroidal coverage) amounts to 823 m^3 with fractions of 36/4/20 percent for steel/breeder/beryllium, the rest of 15/7/19 percent for primary helium/purge helium/void. The loop arrangements account for redundancies, leading to an overpower capacity of 28 % from nominal. The total helium inventory in the cooling system amounts to 22 tons, with 7 tons in the largest subsystem. The total mass of beryllium in all blanket segments amounts to 300 tons.

The tritium inventory in the fluids (helium coolant and purge gas) is small, i.e., less than 1 g. In the solids (breeder and structural material) it is moderate, approximately 10 g each in the Li_4SiO_4 and in the MANET. However, a large tritium content of about 1300 g is trapped in the 300 tons of beryllium multiplier. The total activity in all blanket materials amounts to 1.3×10^{20} Bq at shutdown. It is dominated by the structural material (predominantly by that contained in the first wall and breeding zone). The activity in the breeder material is typically 3 to 4 orders of magnitude lower and the activity in the multiplier ranges in between, depending on the amount of tritium trapped. The dominating nuclides in terms of activity and gamma dose rate at different time intervals are summarised in Table 8.

Energy sources for mobilisation: The main energy sources for mobilisation result from decay heat and work potential of helium (Table 9). The chemical energy potential of beryllium is extremely high but a large scale chemical reaction is hardly conceivable. The decay heat would cause fictitious adiabatic temperature rises in isolated parts of the blanket of about 240-480 K in the first hour in the first wall, 40-100 K in the first day in the shield and manifold region, <25 K in the first day in the breeder material, and <11K in the first day in the multiplier. An intimate mixture of all materials in the blanket would heat up ≈135 K in the first day (Table 10).

The helium inventory from one outboard cooling subsystem would pressurise the vacuum vessel in case of a LOCA to 1.5 MPa. This is beyond the expected design pressure and requires an extra expansion volume of around 70000 m^3, assuming 0.2 MPa end pressure. The release times are short (few seconds) and the momentum forces of a double-ended major pipe break are high (up to 4200 kN per side). In the case of a first wall failure the helium would be released from both cooling subsystems at the same time.

Fault tolerance: Short term LOCA temperature excursions in the blanket structure are on the order of 70 K above steady state levels. They are not expected to threaten the blanket integrity, but further investigations are needed. Similar analyses for a loss of flow accident (LOFA) yielded even smaller temperature amplitudes.

Active cooling of a blanket segment during handling is not required from the internal temperature distribution point of view, if the free convection surrounding gas (air or helium) can be kept at a temperature level around the normal ambient temperature.

Tritium and activation products release: The early and chronic dose for a conservative tritium release scenario in case of a postulated accident (rupture of a first wall) have been estimated to be about 0.02 mSv and 0.1 mSv, respectively. This is below currently discussed tolerable dose limits. The release of activation products has not been assessed. Results from literature suggest that the dose levels may range in the same order of magnitude as those obtained for tritium release. Further work is needed in this field.

Waste generation: The specific activity and the afterheat of the blanket materials in different radial zones have been assessed and compared with the latest IAEA waste classification scheme. The 75 tons of breeder material are supposed to be qualified as short-lived low and intermediate level (LILW-SL) waste. The beryllium multiplier is expected to be qualified as long-lived low and intermediate level (LILW-LL) waste. The first wall structural material is close to the boundary between high level waste (HLW) and LILW. The majority of the structural material is well below the HLW/LILW boundary but has a long-term gamma dose rate. All material groups will require shielding during handling and transportation.

7.2 Conclusions

No insurmountable safety problems have been identified for the breeder outside tube blanket concept. Overall, the safety and environmental impact has been judged to be of almost the same level as the one evaluated for the other three European DEMO blanket concepts [16]. Nevertheless, a number of concerns need further investigations or optimisation in the five areas addressed.

Blanket materials and toxic materials inventory: The nominal tritium inventory in fluids is small and does not raise severe safety concerns for the public in accidental situations. On the other hand, minimisation of the routine release via steam generators relies on permeation barriers at the tube bundle. The effectiveness, reliability and durability have still to be proved. Operational tritium losses through pipe walls, component walls and from leakage have not been elucidated but need attention. All those effects are governed by the tritium transport through the first wall into the primary helium coolant which carries large uncertainties and needs further research.

Besides the tritium a certain amount of sputtered and eroded activation products from the channel walls will be carried by the main coolant and by the purge gas. The generation, transport, deposition behaviour, and means for on-line removal of such products have to be studied. In this context the issue of developing low activation structural material is to be seen. Also the potential and technical means to minimise the impurities in the beryllium multiplier (like Co, Al, Fe, Mn, U) have to be elaborated.

Energy sources for mobilisation: The main energy sources for mobilisation result from afterheat and work potential of helium. The afterheat is dominated by the decay

in the structural material rather than in the breeder and multiplier. The heat generation is moderate relative to the heat capacity of the system (leading to a high thermal inertia) and can easily be rejected by one of the two separate cooling systems connected to each segment. It is expected, but needs to be proved, that this is also true in the case of natural convection circulation, particularly at reduced system pressure.

The work potential of the helium coolant is large and requires an extra expansion volume for the case of a LOCA. To cope with pressure transients in the vacuum vessel and other containment compartments a containment strategy and accident management strategy (definition of primary initiating events, allocation of expansion volumes, possibility of venting, effectiveness of flow restrictors in combination with isolation valves) have to be developed.

The large chemical energy potential of beryllium in conjunction with the large amount of tritium trapped calls for investigations into chemical reaction kinetics of beryllium pebbles for conceivable failure sequences, like water or air entrainment via the purge gas system or the primary cooling system.

Fault tolerance: Stresses induced by disruptions have not been addressed in this report but were assessed in the blanket selection exercise. They do not seem to create severe safety concerns at the present state of knowledge. However, large uncertainties still exist in the modelling assumptions with regard to physics phenomena, mechanical boundary conditions, design features, effect of poloidal field coils, and material strength. Further R&D is needed which is not specific to a particular blanket concept.

Short term temperature transients for the LOCA cases investigated are moderate but the scenarios are to be extended to LOCAs occurring simultaneously in two cooling subsystems, and to further LOFA cases. The latter cases imply also the transition periods to achieve natural circulation heat transfer. Since the average specific afterheat is higher in the inboard segments than in the outboard segments the analysis should be extended to the inboard. In these studies the effects of fabrication tolerances, modelling uncertainties, and partial plugging on flow maldistribution in multiple parallel coolant channels need to be considered.

Tritium and activation products release: Some of the inventory and, hence, release issues have already been addressed in the first paragraph of the conclusions. The release scenarios have to be elaborated aiming at reducing overly conservative assumptions. In general, this complex is not considered critical for design basis accidents, but beyond design basis accidents scenarios have to be taken into account. Note that in the whole blanket selection exercise the issues related to the first wall material (chemical reaction, tritium liberation, dust) have been deliberately ignored.

Waste generation: Establishing rules for waste classification is a matter of national policy. According to the IAEA safety guide line, all but the first wall structural material can be classified as low and intermediate level waste. A further reduction of the waste problem is the objective of the world-wide effort in developing low activation materials. Beryllium and breeder materials are envisaged for recycling. Adequate techniques have yet to be developed.

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