Radioactivity and Associated Problems in Thermonuclear Reactors

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RADIOACTIVITY AND ASSOCIATED PROBLEMS IN THERMONUCLEAR REACTORS

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The need for radioactivity calculations in thermonuclear reactors is reviewed. The type of data needed to make the calculation is discussed along with the calculational scheme for processing the data and making the calculations. The results are reviewed for Tokamak reactors along with a discussion of various forms of presenting the results. The calculations are applied to calculate the shutdown close behind the shield in selected CTR designs. The results show that further work needs to be done with more explicit paths and means for radioactivity to be released in order to properly assess the hazards of fusion reactors.

In the conceptual design of fusion-based reactor systems, there are numerous, oftentimes conflicting, considerations which enter into the choices made for materials in the blanket and structure surrounding the plasma. One of these considerations is that of the radioactivity induced in these materials. Although the other considerations, such as radiation damage, mechanical stresses, heat removal, etc. place such severe limitations on the design that induced radioactivity may be treated as an after-the-fact consideration, nevertheless it is still important affecting such topics as accident analysis, routine maintenance, blanket replacement and repair, waste storage, blanket reprocessing, and even may be reflected in resource requirements for fusion systems.

The first systematic treatment was that of Steiner and Fraas[1] who calculated the activity and associated quantities for two possible structural materials (niobium and vanadium) for the Tokamak reactor design of Fraas. Since that time, more attention has been given to the problem and a section on induced activity may be found in most reactor design reports. This work is summarized in several papers, for example, those of Vogelsang, Kulcinski, Lott, and Sung[2], Dudziak and Krakowski[3] and Williams and Santoro[4]. Taking a somewhat different approach, Powell and his group[5,6] have proposed reactor designs in which the blanket designs have been chosen to minimize the activity through the use of low activity (or at least low residual activity) materials. Conn, Sung, and Abdou[7] showed that with proper choice of blanket material, the neutron spectrum can be altered and thus affect the induced activity.

The information needed to calculate the radioactivity and associated quantities is substantial. Among the data needed are the composition of the blanket, the energy dependent neutron flux at each point, the energy dependent cross sections, the nuclear decay constants, and the type and energy of the emitted radiation. The nuclear constants are not only needed for those isotopes present initially in the reactor but, in many instances, the cross sections and decay information for nuclei produced (either stable or radioactive) are also needed. Of these quantities, the cross sections are the most difficult to obtain. The calculations require both particle emission and capture cross sections over a range from 14 Mev down to thermal energies. When the experimental data is reviewed, it is found to be insufficient or absent in many cases. Consequently, reliance must be placed on calculated cross sections. Fortunately, through the work of the Brookhaven group in developing the THRESH[8] code, there now exists a body of calculated cross sections in a form convenient for use in reactor calculations. While these cross sections may not be as accurate as desired, they seem to provide an adequate base for the kinds of calculations and comparisons needed at this stage of reactor development. It seems reasonable to expect that by the time more precise calculations are required, even more reliable cross sections will be available.

In the calculations of activity, the materials are exposed to the neutron flux for times in the range of from one year up to the reactor lifetime (~30 years). Because of this long exposure and because one of the areas of interest is the residual activity at long times after shut-

down, it is necessary not only to calculate activity from isotopes originally present in the reactor but also to calculate effects due to interactions with isotopes produced by transmutation. Techniques have been adapted to treat this problem via such methods as the use of linear chains, however, difficulties arise in trying to decide when to terminate a chain. It is not sufficient to look at the various activities at shutdown since the relative contribution of a long-lived isotope at that time may be insignificant while at some later time, after shorter-lived isotopes have decayed away, its relative contribution may be quite appreciable. The development of suitable procedures for terminating chains is one of the more difficult problems in setting up a general scheme for calculating radioactive inventories.

Aside from this problem, the implementation of a general scheme is involved rather than complex. There are numerous reactions possible, spatially varying fluxes, changing neutron spectra, and varying compositions and dimensions of the blanket. In addition, once the activities have been calculated, it may be necessary to proceed with a gamma transport problem to determine where the energy is actually deposited or what dose rates might be. A block diagram of a program designed to treat this problem for fusion reactors is shown in Figure 1[9]. The basic cross section and decay data is read in either from ENDF format or from other data sources and processed into a suitable group format. This data is then combined with the neutron spectrum and the geometrical and material properties of the blanket. Linear decay chains are constructed and then solved to obtain the activity. These results may be presented directly or transformed into other measures of activity. Provision is made for the results to appear in a form suitable for a gamma transport calculation which may be performed. The procedure is flexible enough to allow the calculation of most quantities which could be of interest, however as it stands, it is best adapted to one dimensional calculations. The results, which will be presented later, are based on this program.

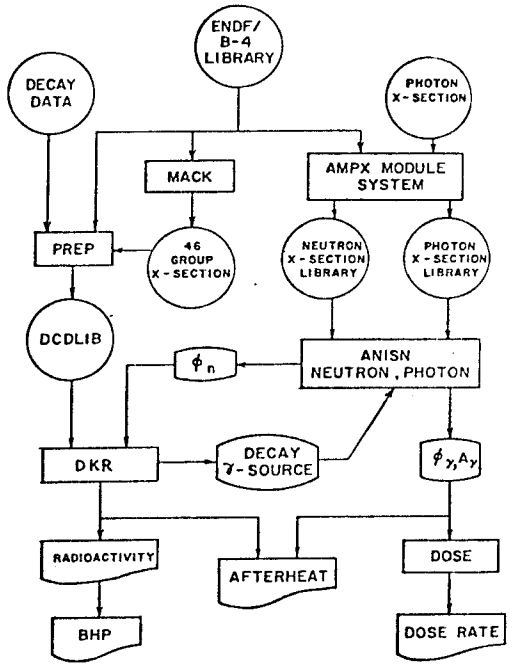

FIGURE 1. Block Diagram of CTR Radioactivity code DKR

Once the calculations of the radioactivity in the blanket have been performed, the question that arises is how to present the results in the most meaningful way. There is a spatial dependence of the activity, and the largest amount is usually in the region closest to the plasma. As the neutron spectrum changes with position in

the reactor, the relative amounts of the various isotopes also changes. There may be different materials in the blanket such that one material or region may dominate the activity at early times while, at later times, the material in a remoter region may provide the largest source of activity. Different users may have different interests, the heat transfer oriented may wish to know what the energy deposition rate is, while those concerned with maintenance wish to know the dose rates. The originator of the calculation wishes to inform everyone of the nature and magnitude of the problem and put it into perspective with other designs. To try and meet these and other needs, several different methods have been used to present the results of the calculation.

To provide an overall picture, the spatial dependence may be ignored and the results are summed over the whole reactor. However, in order to make comparisons between designs, it is convenient to normalize using the thermal output of the plant. The results are presented as curies per kilowatt, for example. However, this is not a very satisfactory description since it makes no allowance for the nature of the radiation but simply counts events. A somewhat more useful way to express the results is to calculate the energy associated with the radioactive decay, i.e., the afterheat, which may be expressed as a fraction or percent of the thermal power of the plant. Neither of these methods takes into account the biological effects of the different isotopes. As an initial attempt to do this, Steiner[1] has suggested using the concept of "biological hazard potential" (BHP). The BHP for an isotope is the amount of air or water required to dilute that isotope to "maximum permissible concentrations" (MPC) again per unit of thermal power from the reactor. These BHP then provide a figure of merit for comparisons of different materials or designs. The concept suffers from the obvious defect that it assumes that all isotopes have equal probability of release, i.e., there is no mechanism for getting the isotopes in a chemical form where dilution and MPC have meaning.

The activity or BHP discussed above represent the radioactivity at one point in time following shutdown. In an effort to account for the different half lives of the various isotopes present, it has been suggested [10] [3] that a better measure of activity or BHP commitment following shutdown is an integral of the activity or BHP from shutdown to infinity. In the case of an isotope decaying to a stable product, this results in weighting the activity or BHP of an isotope by the mean life of that isotope. Additionally, it is equivalent to saying that the quantity of interest is proportional, not to the disintegration rate, but to the number of unstable nuclei produced which will eventually decay. In the case of multiple decay, it represents the number of unstable nuclei times the number of decays required to reach a stable nucleus, each appropriately weighted in the case of BHP. The hope is that this method gives a better relative weight between an intense short-lived isotope and a less intense but longer-lived one. Further discussion of this point will be deferred until later in this report.

To summarize the above, the present situation is that the necessity of the calculation of the radioactivity of fusion reactors has been recognized and most de-

signs brought forward include a calculation of the radioactive inventory of these reactors and make an effort to make judgements with respect to the consequences of this activity. The data and methods of calculations are adequate for the present state of development of fusion technology.

To make the previous remarks more meaningful, the results of a calculation for one such reactor design will be presented. The reactor chosen is UWMAK-I[11]. This reactor, which is one of a series of D-T Tokamak designs from the University of Wisconsin group, has a thermal output of of 5000 Mw. The major radius is 13 meters, the nominal plasma radius is 5 meters, and the neutron wall loading is 1.25 MW/m. The base design uses liquid lithium as a coolant with the first wall and blanket structure being 316 stainless steel. Figure 2 shows the representation of the blanket and shield regions used in the neutronics and afterheat calculation. While the base design material for the blanket structure is 316 stainless steel, the calculations have been made for alternate blanket materials specifically, TZM (0.45% Ti, 0.1% Zr, remainder Mo), Nb-1Zr, V-20 Ti and 2024 series aluminum alloy. In all cases, the substitutions were made on a straight volume for volume basis. In an actual reactor design, adjustments in relative volumes might well have to be made to allow for the different mechanical properties of the materials, but the substitution used above is useful for relative comparisons here. Also, the neutron flux was not recalculated for each material. Past experience is that this introduces no more than a few percent error.

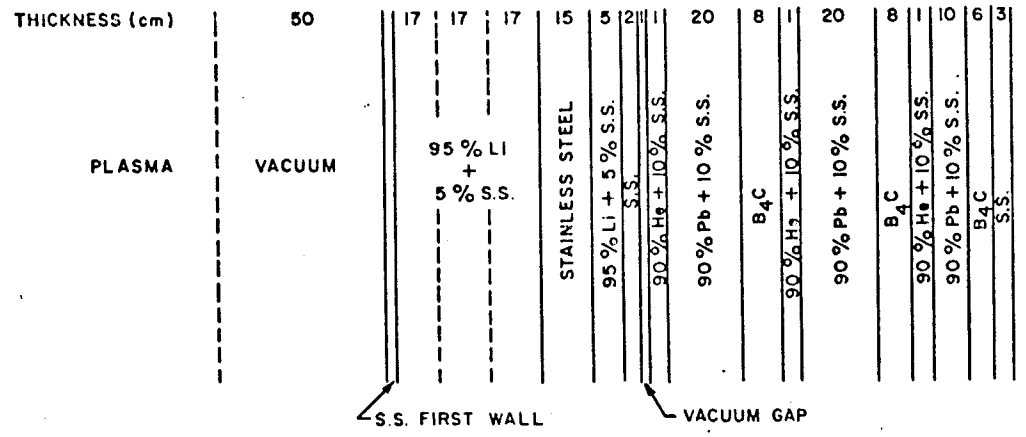

FIGURE 2. UWMAK-1 Blanket model used for neutronics and radioactivity calculations

The first item that may be considered is the buildup of the activity as the reactor operates. This is shown in Figure 3. There are three main points of interest in this figure. First, for most of the materials, the activity levels build up quite rapidly. After only 1000 seconds, TZM has 10% of the activity expected after 30 years of operation, while Nb-1Zr has 93% of the expected 30 year activity. After one day all materials have at least 50% of 30 year activity and after 10 days, it is over 90%

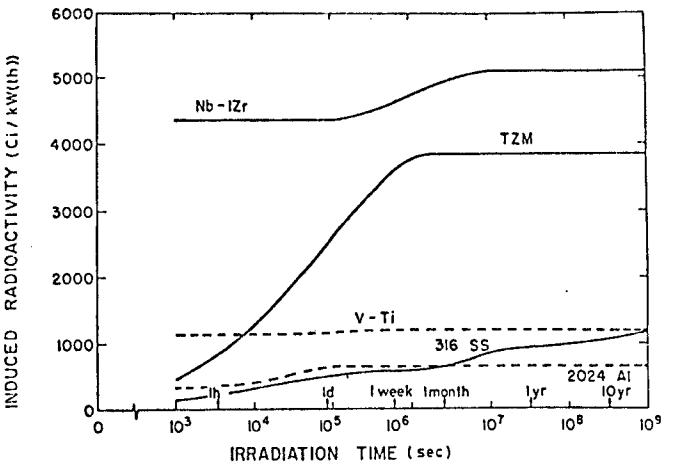

FIGURE 3. Build-up of Blanket Radioactivity following Start-up

for all materials except 316SS. This means that the radioactivity builds up quite rapidly and as far as short time effects, i.e., loss of coolant events, shutdown maintenance, etc. the problem is practically as bad as it is going to be after only a few days of operation and is quite significant almost from the start. The second point is that the levels of activity are quite significant. For example, for V-20 Ti, the activity is about 1100 curies/kw. Since the plant thermal rating is 5000Mw (5×10^6Kw), the radioactive inventory is ~5.5×10^9 curies. The third feature is that regardless of the material, the radioactivity levels after two years, for example, do not differ greatly. The activity from the Nb-1Zr case is ~7 times as large as that from 2024 aluminum.

Following operation for some period, it is of interest to consider the way the activity decays away. In such a consideration, it is useful to consider three different time spans: short term, 0 to a few hours; intermediate times, 1 day to a few months or a year; and long term, longer than a few tens of years. The reason for this distinction is that each one of these intervals is important for a different reason. In the short time interval after shutdown, the designer is interested in the activity from the point of view of how it might affect abnormal or accident conditions. In the intermediate time span, the interest focuses on what are the problems associated with major maintenance work, e.g., the replacement of whole sections of the first wall or whole blanket modules. The interest in the long term is that of evaluating the problems of long term storage of radioactive wastes. Of course, these may be situations which bracket these time spans. If, for example, there is interest in reprocessing blanket material, then both the intermediate and long times must be considered.

Figure 4 shows the activity following shutdown after two years of operation. In the short time range ~10^{4-5}seconds, the activities differ by a factor of 7 with 2024 Al, the lowest, and Nb-1Zr the highest. In the intermediate range, around 10^6 seconds, V-20 Ti and Nb-1Zr have the lowest activity being a factor of 20 lower than

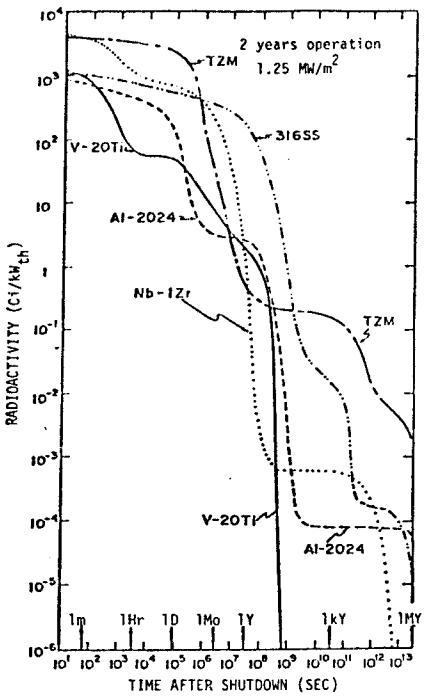

FIGURE 4. UWMAK-1 Blanket Radioactivity after Shutdown

Nb-1Zr and 316SS, and a factor of 60 lower than TZM. For long times (greater than 100 years) the vanadium alloy is best having no long-lived isotopes. (Note that in a more realistic treatment, impurities in the alloy might well yield a noticeable activity level.) The presence of ^{93}Mo in TZM ($t_{1/2}$ ~1,000 years) becomes the dominant activity after about 100 years at an activity level of 0.2 - 0.3 Ci/kw_{th}. This same isotope is present in 316SS but at a much lower level appearing at about 10^{-2} Ci/kw_{th}. Note that in 316SS after about 10,000 years, the dominant activity is ^{59}Ni ($t_{1/2}$ = 80,000 years) at an activity level of 10^{-4} Ci/kw_{th}. In Nb-1Zr, the dominant long-lived activity is ^{94}Nb ($t_{1/2}$ = 20,000 years) at an activity level of $6x10^{-4}$ Ci/kw_{th}. The aluminum alloy has a long lived activity of $9x10^{-5}$ Ci/kw_{th} due to the presence of 26 Al ($t_{1/2}$ = 740,000 years). Table 1 presents the times required for the activity of the various materials to decay to lower radiation levels.

By themselves, these numbers are not too meaningful. They can be put into some perspective by noting that they represent quite low activities per unit volume. Table 2 shows the activity per unit volume of the highest activity region i.e., the first wall. The activity at shutdown ranges from 27 Ci/cm^3 for V-20Ti to 158 Ci/cm^3 for Nb-1Zr. After 100 years of storage, however, the activities range from very low values for V-20Ti to $10^{-5}Ci/cm^3$ for Nb-1Zr and 2024 Al to $4x10^{-4}$ Ci/cm^3 for 316SS and $7x10^{-3}Ci/cm^3$ for TZM. These values may be compared with the activity of natural uranium which has a value of $6x10^{-6}$ Ci/cm^3, which is higher than V-20Ti but about the same as Nb-1Zr or 2024 Al and a factor of 270 lower than 316SS or a factor of 1000 lower than TZM.

A slightly different representation is obtained if, rather than looking at the results in terms of the number of disintegration, attention is focused on the energy released in the radioactive decay. The afterheat following two years of operation is shown in Figure 5. Here we see that whereas the initial activities differ by a factor of ~7, the initial afterheat differs by only a factor of 3.5. V-20Ti has the highest afterheat, being ~1.5% of the operating power with TZM being only slightly lower at 1.4%. Nb-1Zr, on the other hand, which showed the highest activity at shutdown, has the lowest afterheat at 0.45%. 316SS and 2024 Al are intermediate at ~1%. After a year of decay, however, 316SS has only decayed by a factor of 9 to 0.1% while all the other alloys show significantly greater decay

TABLE 1. Time required to reach specific levels of radioactivity inventories in the metals of various CTR reactor designs - 2 year operation

	Approximate Time Required to Achieve Radiation Level				
	316 SS	TZM	Nb-Zr	V-Ti	2024-Al
Initial Level					
Ci/kW	*1200*	*3800*	*5100*	*1200*	*750*
1000 Ci/kW	200 sec	3 d	3 h	1 min	-
100 Ci/kW	3 yr	3 weeks	2 weeks	20 min	1 d
10 Ci/kW	15 yr	3 months	1 month	10 d	3 d
1 Ci/kW	30 yr	1 yr	3 months	3 yr	4 yr
10^{-1} Ci/kW	40 yr	10,000 yr	6 months	3.3 yr	20 yr
10^{-2} Ci/kW	50 yr	50,000 yr	2 yr	5 yr	25 yr
10^{-3} Ci/kW	25,000 yr	700,000 yr	20 yr	10 yr	30 yr

TABLE 2. Maximum radioactivity density in the first wall of various CTR blankets

	Ci/cm^3[(a)]		after given decay time	
Alloy	t=0	1 day	1 year	100 years
316 SS	100	68	29	$4 \cdot 10^{-4}$
TZM	125	83	$4 \cdot 10^{-2}$	$7 \cdot 10^{-3}$
V-Ti	27	$2 \cdot 10^{-3}$	10^{-4}	$<10^{-37}$
2024 Al	44	8.7	0.3	$1 \cdot 10^{-5}$
Nb-Zr	158	94	$6 \cdot 10^{-4}$	$1 \cdot 10^{-5}$
Natural Uranium				$6 \cdot 10^{-6}$

(a) 2 year operating time, 1.25 MW/m^2

with 2024 Al being the next highest at only $1x10^{-3}$%. In the long term case(t>100 years) V-20Ti again is the lowest because of the absence of long lived isotopes, while both TZM and 316SS are a few x 10^{-6}%. At even longer times when ^{93}Mo dominates the decay of both TZM and 316SS, the 316SS afterheat is ~$3x10^{-7}$%, i.e., about one decade lower than TZM. Again, the residual afterheat due to the long lived isotopes covers a much narrower range than does the activity ranging from ~$1.5x10^{-7}$% for 2024 Al to $3x10^{-6}$% for TZM. Thus, the net amount of heat generated in long term storage is not great being less than 150 watts for all of the blanket material at times greater than 100 years. In terms of power density, these levels represent a power density of less than $7x10^{-5}$ watts/cm^3 for these long times.

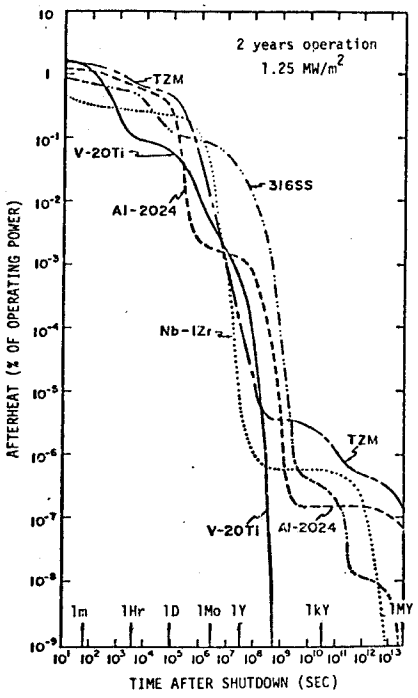

FIGURE 5. UWMAK-1 Blanket Afterheat after Shutdown

As mentioned previously, while the energy released by the radioactive products is certainly more meaningful than activity, it still falls short of taking into account the biological effects of the activity and the use of BHP as an attempt to take these biological effects into account. As stated earlier, BHP is defined as the activity/kilowatt divided by the MPC value for air or water. The MPC values and BHP at shutdown for the five materials are shown in Table 3. The MPC values are taken from 10 CFR 20 or the Los Alamos calculated values reported by Dudziak[3]. The excep-

TABLE 3. Summary of Biological Hazard Potentials (BHP) of Various CTR Structural Materials at Shutdown

Isotope	MPC (Ci/km^3)	BHP (km^3 of air/kW(th)) 316 SS	TZM	V-20Ti	2024 Al	Nb-1Zr
Na24	5				51.2	
Mg27	30	2-04			7	
Al26	0.1				9-04	
Al28	30	0.64			3.7	
Ca45	1.0	0.01	2.6-01	6.5		
Sc46	0.8	0.01	5.9-01	14.9		
Sc47	20	5-04	2.4-02	0.62		
Sc48	5	4-03	2.0-01	10.7		
Sc49	1300		6.0-05	1.0-03		
Ti45	340		0.2	2.0-03		
Ti51	850	2-04		0.13		
V49	250	6-03		9.0-03	2.2-02	
V52	350	0.14		3.0	1.2-02	
Cr51	80	1.6				
Mn53	0.1	7-05			1.8-07	
Mn54	1	75.6			4.4	
Mn56	20	19.4			1.3	
Mn57	30	0.12				
Fe55	30	8.6			2.1-02	
Fe59	2	0.1				
Co57	1	26				
Co58	2	65.5				
Co60m	30	0.44				
Co60	0.3	20.7			1.0	
Ni57	1	4.5				
Ni59	20	3-06				
Ni63	0.1	0.5				
Cu64	40				3.5	
Zn65	2				6.0-03	
Sr89	0.3	7-03				6.0-02
Sr90	0.03	1-04				1.2-03
Y90	3		5.9-03			3.9
Y91	1.0					0.6
Zr89	0.1		89.7			27.4
Zr95	1.0	1.8	2.2			
Nb92m	370	0.04	5.7-02			2.0
Nb94m	200,000	1-03				2.0-02
Nb94	2		2.1-07			9.0-05
Nb95m	300		1.2-02			5.6-02
Nb95	3	2-04	2.3			5.5
Nb96	0.1	5-02	45.9			
Nb97	200	0.9	2.3-02			
Mo91	30	5-02	2.5			
Mo93	370	1.4-05	6.5-04			
Mo99	7	4.1	172			
Mo101	30	0.28	13.8			
Tc99m	500	6-02	2.9			
Tc99	2	1-04	5.0-03			
Tc101	30	0.3	13.8			
Totals		231	348	36	71	40

tion to this is ^{93}Mo which is assigned the same MPC value as ^{92m}Nb based on similarities of half life and chemistry even though the x-ray energy of ^{92m}Nb is 30 times that of ^{93}Mo.

As can be seen from Table 3. the BHP at shutdown after two years of operation, varies from a low of 63 km^3 per kw (th) in V-20Ti to a high of 348 for TZM. The way the BHP changes with time following shut-

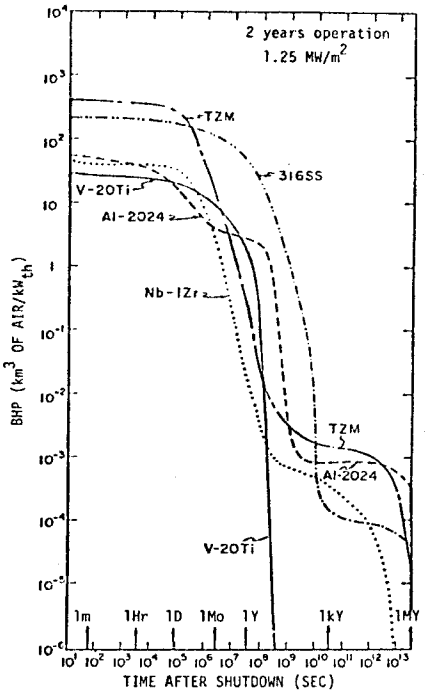

FIGURE 6. UWMAK-1 Blanket BHP after Shutdown

down is shown in Figure 6 for the case of dispersion in air. For the first day the BHP stays roughly constant. The 2024 aluminum alloy is the first to show a drop due to the decay of ^{24}Na($t_{1/2}$ = 15 hrs) so that after the first week it has the lowest BHP value (by about a factor of two). After one year of decay, the situation has changed. Nb-1Zr has the lowest BHP at this time with a value of 10^{-2} km^3/kw (th) compared to the shutdown value of 40. 316SS, however, has undergone relatively little decay, having a value of 40km^3/kw (th) compared to the shutdown value at 230 km^3/kw (th). By 10 years after shutdown, V-20Ti has decayed away to very low BHP values. For the other alloys, however, the long lives isotopes remain to present noticeable BHP values. The presence of 1000 year ^{93}Mo in TZM dominates the BHP for times greater than 10 years at a value of ~$10^{-3}$$km^3$/kw (th). The decay of 316SS, at times greater than 10 years, is dominated first by ^{63}Ni ($t_{1/2}$ = 100 years) then ^{93}Mo ($t_{1/2}$ ~1000 years) resulting in values that level off after a few thousand years at about 10^{-4} km^3/kw (th). At extremely long times first ^{59}Ni ($t_{1/2}$ = 8×10^4 years) then ^{99}Tc ($t_{1/2}$ = 2.14×10^5 years) and ^{53}Mn ($t_{1/2}$ ~3.7×10^6 years) contribute. The presence of ^{26}Al ($t_{1/2}$ = 7.5×10^5 years) results in a long term activity of 8×10^{-4} km^3/kw (th) in 2024 Al somewhat lower than that of TZM. Nb-1Zr is somewhat lower than 2024 Al at long times with a value of 3×10^{-4} km^3/kw (th) with the activity being dominated by ^{94}Nb ($t_{1/2}$ = 2×10^4 years).

The previous kind of calculation of activity or afterheat or BHP really represents only the first step in assessing the problems associated with the induced activity in a fusion reactor. All they represent is a source term for subsequent calculations or a means for making a relative comparison of the potential hazard of similar type plants. The next step is to evaluate the effects of the radioactivity on various real or postulated plant conditions. Some progress has been made in this direction, specifically in estimating the dose behind the shielding of the blanket after shutdown. This is important for evaluating what can be done in the way of maintenance in the reactor room. As representative of dose rates which may be expected from the near term machines, Jedruch[12] has reported the dose criteria and consequent dose rates for the Tokamak Fusion Test Reactor (TFTR). Their design criteria was that two hours after shutdown, the dose inside the test cell would be no greater than 100 mrem/hr and the bulk shielding was sized to meet this criteria. However, they recognized the possible

existence of hot spots and calculated a contact dose rate of ~$3x10^3$ mrem/hr at the source end of one of the neutral beam injectors indicating that additional shielding will be needed for special components. TFTR is a near term machine and the design is based on a limited number of pulses which produce neutrons and contribute to the activity.

Beyond TFTR the nature of the machines to be built becomes more speculative. One line of thought is that of a much higher duty machine designed for engineering tests, primarily radiation damage studies. One such machine is the Tokamak Engineering Test Reactor (TETR)[13]. The dose behind the shield of TETR has been calculated and is given in Table 4 for various times after shutdown. Here the dose is appreciably higher than that of the TFTR,

TABLE 4. Calculated Dose Rate Outside the Shield in TETR for Times After Shutdown

1.10 MW/m^2 Wall Loading

Time After Shutdown	Dose Rate, mrem/hr.
0	790.0
1 min.	766.0
10 min.	675.0
1 hr.	530.0
6 hr.	153.0
1 day	21.7
1 week	17.9
1 month	15.8
1 yr.	6.8
10 yr.	1.3
100 yr.	1.3×10^{-5}
1000 yr.	2.2×10^{-6}

being 790 mrem/hrs at shutdown (2 years operation). The activity dies away fairly rapidly during the first day to 21.7 mrem/hr but after one month has only decayed to 15.8 mrem/hr. This would preclude immediate work inside the reactor area but after a day the fields are low enough that work could begin. This dose rate is that calculated behind a typical blanket section and there are other features of the design which could give higher levels. This point will be deferred until later in the discussion.

To provide some idea of the dose expected behind the shield of a power reactor design, the dose has been calculated for UWMAK-1[11]. The shield in this reactor is a mixture of lead, stainless steel and boron carbide. The thickness of the blanket and shield is 152 cm and the shield was designed on the basis of an allowable dose and nuclear heating to the toroidal field magnets during operation. The doses calculated are shown in Table 5. After shutdown, the dose rate is rather high being ~4,000 mrem/hr, clearly too high for direct maintenance. After one day it is still at about 1200 mrem/hr and even after one year has only dropped to about 170 mrem/hr. Thus, with this design as it stands, maintenance would be severely complicated. However, the calculations show the direction that can be taken to remedy the situation. The most intense region of radioactivity in the blanket-shield region is that immediately adjacent to the plasma, i.e., the first wall and the structure immediately behind it. It might be expected then that

TABLE 5. Calculated Dose Rate Outside the Shield in UWMAK-I for Times After Shutdown

1.25 MW/m^2 Wall Loading

Time After Shutdown	Dose Rate, mrem/hr.
0	4,072.0
1 hr.	3,068.0
1 day	1,165.0
1 week	1,084.0
1 month	870.2
1 yr.	167.5

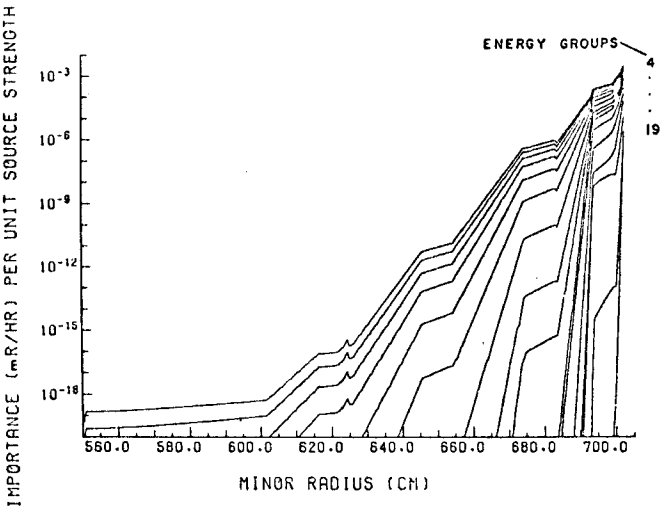

FIGURE 7. Importance of a Unit Gamma Source for Dose to Tissues at Outer Edge of Shield

the dose behind the shield is determined by this activity. However, this is not the case. Ramer[14] has calculated the importance for dose from a unit source in the blanket-shield region and the results are shown in Figure 7. Here it is seen that the importance of gammas produced in the neighborhood of the first wall is at least 16 orders of magnitude less than that of gammas produced in the regions near the edge of the shield, implying that the contribution to the dose from the first wall region is negligible. This is to be expected since the basic purpose of the shield is to attenuate the prompt radiation from this region. Furthermore, the steep slope of the importance curves indicates that only the activity from the outer layers of the shield makes a significant contribution. This is illustrated in Figure 8 which shows the contribution to the dose rate from the shield. It is seen that the major portion of the dose comes from the activitation of the outer 10 cm of the shield region. To reduce the dose at the shield, it is necessary to replace the material producing the activity in this region, i.e., the stainless steel with a material with better decay properties, e.g., 2024 Al. If this can be done and maintain

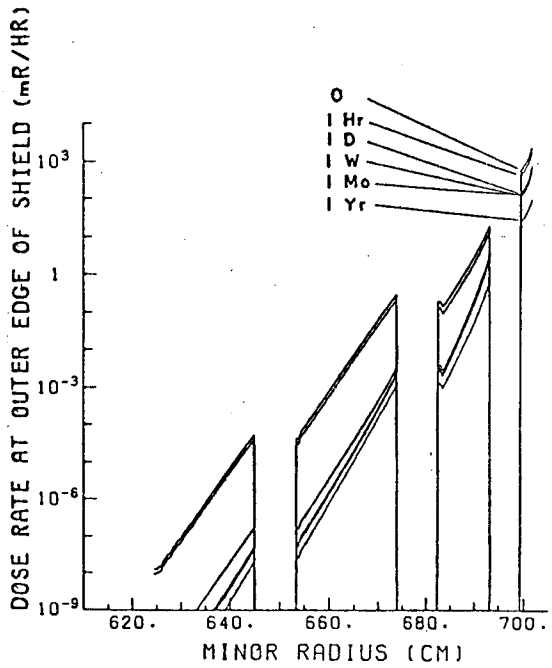

FIGURE 8. Contributions to the Dose Rate from Regions of the Shield

the desired shielding characteristics, then the dose rate behind the shield might even be lower at shutdown but will certainly decay away faster. This calculation neglects the contribution to the dose from the activation of the toroidal field magnets but similar statements may be made and advanced designs may well consider the advantages of using something other than stainless steel as structure in the magnets themselves.

The above dose calculations have usually been based on one-dimensional models treating the blanket-shield as enclosing the neutron source region. As mentioned previously, special design features exist which make the problem more complex both from the standpoint of calculating the radioactive source and the subsequent dose. As an example of this, consider the case of the TETR design. This is a driven machine and as such requires penetrations into the plasma region for the driving particle beams. In TETR, these consist of 12 holes in the first wall, each 0.23 meters by 1.5 meters, located immediately below the mid-plane of the machine. These holes form the machine end of the neutral particle beam ducts which are 9 meters long. (The ion sources are outside of the toroidal field magnets) and are at an angle of 57.5° to the normal of the first wall. Neutrons can penetrate into these beam ducts either directly from the plasma or by back scattering from the blanket opposite the beam openings. Thus, both the duct walls and shielding and the neutral beam sources are activated in the course of operation. To calculate the activity in these components and the resultant dose is clearly a much more complex task than the earlier calculations requiring two and three dimensional calculations in rather complicated geometries. Examples of special regions and areas can be found equally well in other designs. Except for the TFTR calculations reported by Jedruch[12] elsewhere in these proceedings by Abdou little has been done along these lines. However, it clearly has an important bearing on what activities may be carried on outside the reactor during shutdown.

Another area of interest in which the inventory of radioactivity is one of the input parameters involves the consequences of accidents involving melting and release of the activated reactor components. This has been recognized as an important problem for a long time in the evaluation of CTR's as a power source. The use of the BHP as a parameter characterizing the radioactivity of a plant is an attempt to make a quantitative approach to the problem. However, the use of BHP has always been regarded as an imperfect technique for this purpose for several reasons. Even if BHP was a suitable approach, there are uncertainties in the MPC values used which tend to make comparisons artificial. The MPC values used are, for the most part, taken from the appropriate section of the Federal regulations dealing with allowable exposure rates, specifically Part 20 of Title 10 of the Code of Federal Regulations. Unfortunately, MPC values are not given for all isotopes and there is a particular lack of values for isotopes produced by high energy neutrons. According to the Code, if the MPC value is not listed, then the isotope is treated as being as intolerable as the worst possible case and is correspondingly assigned a very conservative MPC value. An example of this type is ^{93}Mo. Quite often these very low MPC values result in very large BHP values for the isotopes in question. An examination of

the MPC values for other isotopes of the same element or of other element with similar chemical properties reveals that, if a proper evaluation were made, the MPC values would be much higher resulting in lower BHP values. Thus, the BHP rather than reflecting an assessment of the hazard of a material may only reflect the way a particular set of regulations have been written. Another problem which occurs in using BHP is that, as previously mentioned, it is often felt that BHP does not reflect the fact that the different isotopes decay at different rates, i.e., no adjustment is made for the fact that one isotope decays faster than another. Consequently, the concept of integrated BHP is introduced whereby the integrated BHP (IBHP) for an isotope is taken as being the integrated value of the activity divided by the MPC, i.e.,

$$\text{IBHP} = \frac{\text{Ci/kw(th)} \int_0^\infty e^{-\lambda t} dt}{\text{MPC}}$$

$$= 1/\lambda \times \text{BHP} = \frac{t_{1/2} \times \text{BHP}}{.693}$$

for the case of an isotope decaying to a stable daughter nucleus.

Intuitively this is an attractive suggestion. However, care must be taken in developing an interpretation for it. The MPC values are determined based on the dose to a critical organ and the effective half life of the isotope in the system. In the simplest case, this effective half life is $\frac{1}{t_{1/2e}} = \frac{1}{t_{1/2B}} + \frac{1}{t_{1/2p}}$ where $t_{1/2B}$ is the biological half life of the isotope in the critical organ and $t_{1/2p}$ is the usual physical half life. Thus, the MPC values are proportional to $(t_{1/2e})^{-1}$ and the IBHP values are proportional to $t_{1/2e}\ t_{1/2p}$. If $t_{1/2B} \ll t_{1/2p}$, then $t_{1/2e} = t_{1/2B}$ and IBHP is proportional to $t_{1/2B}\ t_{1/2p}$ while if $t_{1/2B} \gg t_{1/2p}$ the IBHP is proportional to $(t_{1/2p})^2$. Thus, the IBHP values are, in a sense, double time weighted and too much weight may be given to long half life nuclides. It is as if the IBHP values are weighted by the square of the geometric mean life between biological and physical processes.

There is a more severe criticism that has been made of any of the measures discussed above. None of them differentiate between the likelihood of an isotope being released from the reactor, that is to say that radioactive inventory has been calculated but that no accident scenarios have been followed through to determine how much of what isotope is likely to be released and in what form. The way BHP, for example, has been used makes no difference between an isotope generated in the low flux region at the back of the blanket and one generated at the first wall, no difference between the probability of release from a first wall operating in the reactor and one which has been removed and in storage for years. Thus, completely different situations are being equated. What is needed to fill the gap is a calculation involving more realistic events to try and determine what happens, how much radioactivity is involved, how much stays in the plant, and how much escapes, what isotopes are involved and what form are they released in. Then it should be possible to make a more realistic assessment of the potential hazards of a CTR. The use of quantities like BHP is merely an attempt to defer doing this sort of calculation. The kind of events which should be analyzed include in addition to those involving the reactor while it is operating, conditions which might occur during shutdown or major maintenance, during storage and during reprocessing if that is a feasible option.

Summary:

The means now exist to perform adequate calculations of the inventories of radioactivity in CTR designs. The nuclear data, which certainly will be revised and corrected between now and when the reactors will be built, is adequate for present purposes. Calculational programs have been developed to take advantage of nuclear data files and simplify the computation of the results. Calculations have been performed for various reactor designs with a selection of possible structural materials. The results indicated that at least for Tokamak reactors, the radioactivity levels are high and differ at shutdown by a surprisingly small factor ranging from 5100 Ci/kw_{th} for Nb-1Zr to 750 Ci/kw_{th} for 2024 Al. In the time following shutdown, the V-20Ti and 2024 Al alloys have a clear advantage in either radioactivity or afterheat over TZM, 316SS or Nb-1Zr. At long storage times, all of the material except V-20Ti have long half life contributions. Among those with long lived residual activity, TZM is the highest and 2024 Al the lowest differing by over three orders of magnitude at about one thousand years after shutdown. The dose calculations behind the bulk shielding after shutdown can also be high enough to limit direct maintenance unless careful attention is given to the choice of materials on the outside of the shield or provision is made for the installation of temporary shields. Even such items as not usually considered as contributing to the dose, such as the toroidal field magnets may make a significant contribution.

In an attempt to assess the biological consequences of CTR, several alternative measures of activity have been developed. These include the BHP and the IBHP. While these are useful in making preliminary comparisons, they are not a substitute for developing more realistic release models. Before many conclusions are drawn as to the relative merits of different reactor types or different energy producing schemes, this sort of more detailed analysis must be made.

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