Activation Analyses for the Different Options Considered in the U.S. ITER Blanket Trade-Off Study

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Abstract

Detailed activation analyses have been performed for the different blanket design options considered in the ITER blanket option trade-off study. The options considered included a self-cooled Li/V option, a helium cooled Li/V option, and a water cooled 316 SS non-breeding shield option. A vacuum vessel made of double wall inconel 625 and water cooled 316 SS balls is used with all options. The He-cooled blanket activity is higher than that of the self-cooled blanket due to the larger structure content. Meanwhile, the vacuum vessel activity is lower for the He-cooled blanket option due to the larger neutron attenuation in the blanket. Shield activity and decay heat of the 316 SS/H_2O option are higher than those for the Li/V blankets due to the large amount (80%) of 316 SS used. In both Li/V options the blanket qualifies as Class C low level waste. On the other hand, the 316 SS/H_2O shield does not qualify for disposal as low level waste. The 316 SS/H_2O option produces the highest off-site doses in case of accidental release of 100% of its radioactive inventory. Only remote maintenance would be allowed for all options.

1. Introduction

Since the beginning of the Engineering Design Activity (EDA) of ITER, a variety of blanket design options have been proposed. A blanket option trade-off study has been conducted by the U.S. ITER home team to examine some of these options. As part of this study, detailed activation analyses have been performed for the following blanket options:

(1) A self-cooled Li/V option.

(2) A helium cooled Li/V option.

(3) A water cooled 316 SS non-breeding shield option.

The first two options utilize V-5Cr-5Ti as structural material and liquid lithium as breeder. In both designs, beryllium is used as neutron multiplier. While the first option was proposed originally by the U.S. home team, the second option is based on an early helium cooled blanket design proposed by the ITER Joint Central Team (JCT). Option 3 represents a non-breeding shield option utilizing water cooled austenitic steel (316 SS). All options utilize a vacuum vessel (VV) made of double wall inconel 625 and water cooled 316 SS balls.

Several activation-related issues for the reactor structure have been examined. The activity, decay heat and integrated decay heat have been calculated for up to 1000 years following shutdown. Evaluation of the structure activity is needed to calculate the potential effects of radioactive inventory release in the event of an accident. In addition, results of the decay heat calculation are essential to examine the thermal response of the reactor shield following a loss of coolant accident (LOCA). Another issue that has been examined in this analysis is the waste disposal rating (WDR) of the reactor structure at the end of its lifetime. The waste disposal rating is needed to determine if the structure would satisfy U.S. regulatory criteria for shallow land burial as low level waste (LLW). The contact dose rates have been calculated for different locations inside the reactor containment. Finally, off-site doses caused by the release of 100% of the radioactive inventory have also been calculated.

2. Calculational Procedure

Neutron transport calculations have been performed using a one-dimensional toroidal cylindrical geometry model. The one-dimensional discrete ordinates code ONEDANT [1] has been used along with cross section data based on ENDF/B-V to calculate the neutron flux spatial distribution. The analysis uses a P_3 approximation for the scattering cross sections and S_8 angular quadrature set. The inboard and outboard regions are modeled simultaneously to account for toroidal effects and mutual neutronic interactions between the two regions. The analysis has been performed for the nominal case with 3 GW fusion power and average neutron wall loading of 2 MW/m^2. The corresponding peak inboard and outboard wall loadings are 2.64 and 2.78 MW/m^2, respectively. The inboard and outboard regions are assumed to extend over a height of 12.5 m. The calculations assumed continuous operation for 1.5 full power years (FPY). The radial builds used in the analysis are given in Figures 1-3.

The neutron flux obtained from the neutron transport calculations has been used in the activation calculations. The calculations have been performed using the computer code DKR-ICF [2] with the ACTL [3] activation cross section library. The neutron transmutation data used is in a 46 group structure format. The decay and gamma source data is taken from the table of isotopes [4] with the gamma source data being in 21 group structure format. Total activity, decay heat and integrated decay heat have been determined for the blanket and vacuum vessel. The decay gamma source produced by the DKR-ICF code is used to calculate the biological dose rate after shutdown using the DOSE [2] code. The dose rate calculations have been performed at different locations inside the reactor containment. The activation results have also been utilized in the radwaste classification and off-site dose calculations. The waste disposal ratings (WDR) for the blanket and vacuum vessel have been determined using the calculated specific activities and the US NRC 10CFR61 [5] waste disposal concentration limits. The off-site dose calculations were calculated by the FUSCRAC3 [6] code. The off-site doses are produced by the accidental release of the radioactive inventory from the reactor containment building assuming the worst case weather conditions and 100% release.

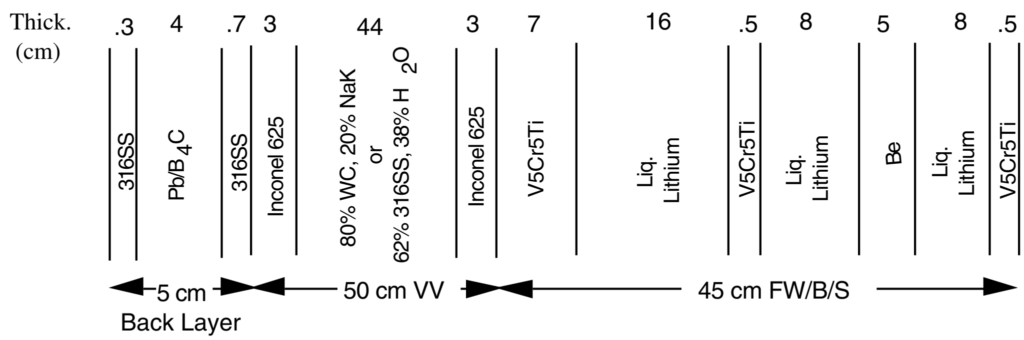

Fig. 1. Radial build of inboard FW/B/S/VV for the self-cooled Li/V design option.

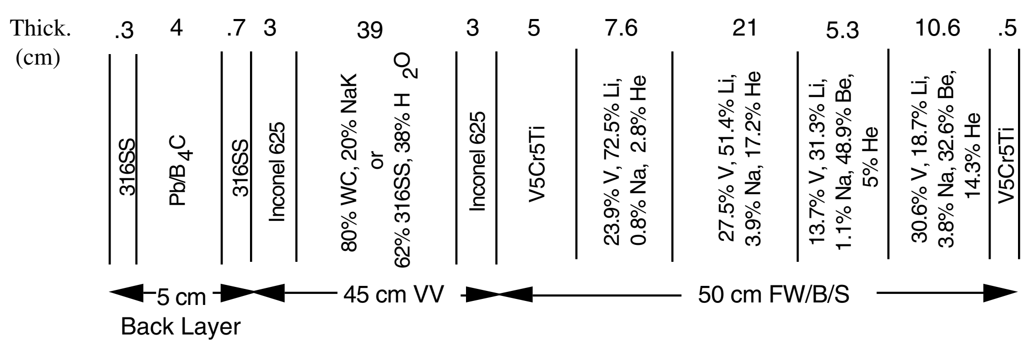

Fig. 2. Radial build of inboard FW/B/S/VV for the helium cooled Li/V design option.

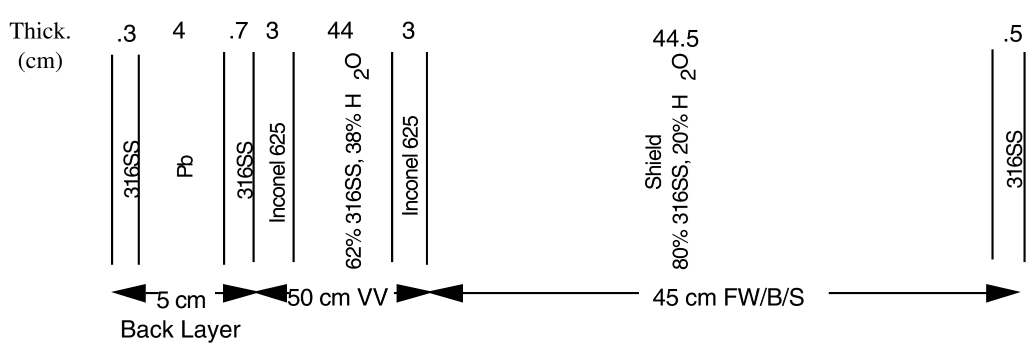

Fig. 3. Radial build of inboard FW/B/S/VV for the non-breeding 316 SS/H_2O design option.

Table 1. Total Activity (MCi) as a Function of Time Following Shutdown.

Time	Self-Cooled Li/V Blanket	Self-Cooled Li/V VV	He-Cooled Li/V Blanket	He-Cooled Li/V VV	316 SS/H_2O Blanket	316 SS/H_2O VV
0	3421	1719	9613	965	11642	51.7
1 min	1863	1704	7042	956	11517	51.5
1 hour	321	1496	919	835	10050	46.3
1 day	254	1067	723	598	6740	35.8
1 week	113	877	298	488	6031	31.3
1 month	73	677	171	373	4979	25.2
1 year	35	150	56	78	2256	5
10 year	17	9.92	22.5	5.3	176	0.27
100 year	0.1	0.69	0.15	0.44	2.13	0.023

Table 2. Total Decay Heat (MW) as a Function of Time Following Shutdown.

Time	Self-Cooled Li/V Blanket	Self-Cooled Li/V VV	He-Cooled Li/V Blanket	He-Cooled Li/V VV	316 SS/H_2O Blanket	316 SS/H_2O VV
0	41	11.7	128	6.7	76	0.338
1 min	25	11.5	98	6.6	74	0.335
1 hour	4	9.4	11.9	5.4	55	0.285
1 day	2.8	4.6	8.3	2.7	13	0.175
1 week	0.4	3.9	1.15	2.3	11.7	0.163
1 month	0.088	3.1	0.24	1.9	10	0.138
1 year	0.013	0.52	0.031	0.31	3.1	0.021
10 year	0.001	0.044	1.5×10^{-3}	0.025	0.25	1.2×10^{-3}
100 year	5.4×10^{-6}	1.9×10^{-4}	7.3×10^{-6}	1.2×10^{-4}	3.1×10^{-4}	5.2×10^{-6}

3. Activity and Decay Heat

Radioactivities produced in the three options have been calculated using all possible impurities of V-5Cr-5Ti, 316 SS, Be, Pb and inconel. Tables 1 and 2 give the total activities and decay heat in the blanket and vacuum vessel (VV) as a function of time following shutdown. The decay heat and activity results exhibit similar trends. The He-cooled blanket activity is higher than that for the self-cooled blanket due to the larger structure content. On the other hand, the vacuum vessel activity is lower for the He-cooled blanket option due to the larger neutron attenuation in the blanket. Since the activity in the 316 SS and inconel used in the vacuum vessel does not fall as rapidly with time as the activity in the vanadium and beryllium used in the blanket, the total blanket and vacuum vessel activity is higher for the self-cooled option at times greater than an hour following shutdown. Shield activity and decay heat of the 316 SS/H_2O option are higher than those for the Li/V blankets due to the large amount (80%) of 316 SS used. On the other hand, the vacuum vessel activity is much lower than that with the Li/V blanket due to the better shielding performance obtained without a breeding blanket. In general the fluxes in the vacuum vessel of the Li/V options are more than an order of magnitude higher than the 316 SS/H_2O.

The short-term activities (≤1 day after shutdown) generated in the two Li/V blanket options are dominated by ^{48}Sc ($T_{1/2}$ = 43.7 hr), ^{51}Cr ($T_{1/2}$ = 27.7 day), ^{47}Sc ($T_{1/2}$ = 3.349 day) and ^{45}Ca ($T_{1/2}$ = 162.7 day). In the period between 1 day and 1 year after shutdown, ^{49}V ($T_{1/2}$ = 337 day), ^{45}Ca and ^{46}Sc ($T_{1/2}$ = 83.81 day) dominate the activity induced in the blankets. ^{14}C ($T_{1/2}$ = 5730 yr), ^{93m}Nb ($T_{1/2}$ = 16.1 yr), ^{94}Nb ($T_{1/2}$ = 20,000 yr), and ^{63}Ni ($T_{1/2}$ = 100 yr) dominate the long-term activity in each of the three blankets. On the other hand, in the 316 SS/H_2O option, the short-term activity is dominated by ^{55}Fe ($T_{1/2}$ = 2.73 yr), ^{56}Mn ($T_{1/2}$ = 2.6 hr), ^{51}Cr, and ^{58}Co ($T_{1/2}$ = 70.88 day). The intermediate-activity is produced by the decay of ^{55}Fe, ^{51}Cr, ^{54}Mn ($T_{1/2}$ = 312.2 day), ^{57}Co ($T_{1/2}$ = 271.8 day), ^{58}Co, and ^{60}Co ($T_{1/2}$ = 5.27 yr). ^{63}Ni, ^{93m}Nb, ^{93}Mo ($T_{1/2}$ = 3,500 yr), and ^{59}Ni ($T_{1/2}$ = 7.6 x 10^4 yr) are the major contributors to the long-term activity.

The short-term activities induced in the VV of all of the three options are due to ^{99}Mo ($T_{1/2}$ = 2.75 day), ^{99m}Tc ($T_{1/2}$ = 6.01 hr), ^{182}Ta ($T_{1/2}$ = 114.43 day), and ^{183}Ta ($T_{1/2}$ = 5.1 day) contributed by inconel, as well as ^{56}Mn, ^{55}Fe, and ^{51}Cr contributed by the 316 SS portion of the vacuum vessels. The intermediate and long-term activities are dominated by the 316 SS constituents. However, the large amount of Nb included in the inconel (3.06w%) results in ^{94}Nb being the largest contributor to the long-term activity following shutdown.

Table 3. Integrated Decay Heat (MJ) as a Function of Time Following Shutdown.

	Self-Cooled Li/V		He-Cooled Li/V		316 SS/H_2O	
Time	Blanket	VV	Blanket	VV	Blanket	VV
0	41	11.7	128	6.7	76	0.338
1 hour	2.8×10^4	3.7×10^4	9.4×10^4	2.1×10^4	2.3×10^5	1104
1 day	3.1×10^5	5.1×10^5	9.3×10^5	3×10^5	2.1×10^6	1.8×10^4
1 week	9.4×10^5	2.8×10^6	2.8×10^6	1.6×10^6	8.5×10^6	1×10^5
1 month	1.3×10^6	9×10^6	3.8×10^6	5.3×10^6	2.8×10^7	3.8×10^5
1 year	2.4×10^5	5.2×10^7	6.8×10^6	3.1×10^7	2×10^8	2.1×10^6
10 year	3.7×10^6	1.1×10^8	9.6×10^6	6.3×10^7	5.2×10^8	4.2×10^6
100 year	4.2×10^6	1.3×10^8	1×10^7	7.6×10^7	6.3×10^8	4.8×10^6

In Table 3 a comparison between the the integrated decay heat values of the different blanket options showed that the 316 SS/H_2O option produces the highest values at all times following shutdown. Since the amount of structure in the self-cooled Li/V blanket is smaller than the amount used in the helium cooled Li/V blanket, the integrated decay heat of the self-cooled option is about a factor of 3 lower at all times following shutdown. On the other hand, comparing the integrated decay heat values of the VV of the different options showed that the self-cooled Li/V option produces the highest values. While decreasing the amount of structure in the blanket reduces its decay heat, at the same time more neutrons will interact with the VV resulting in an increase of its decay heat. This is a preferable trade-off as radioactivity generated in the first wall and blanket is usually more of a safety concern than that generated in the VV. Finally, comparing the total integrated decay heat (blanket + VV) for the three options showed that the 316 SS/H_2O design produces about 30% more decay heat than the other options during the first month following shutdown.

4. Contact Dose

Contact dose rates have been calculated for both maintenance and radwaste evaluation. The contact doses have been calculated at four different locations. Results labeled "Maintenance" include the dose contributed from all the surroundings at each location. On the other hand, results labeled "Radwaste" include the dose contributed by the contacted zone only. For example, for maintenance purposes, the front of vacuum vessel values show the doses contributed by the first wall, blanket, back plate and vacuum vessel. In the meantime, for radwaste purposes, results at the same location show the doses contributed by the vacuum vessel only. The European Community criteria for waste management depend on specific decay heat (W/cm^3) and contact dose rate (mSv/h).

The results in Tables 4-6 show that assuming a 25 μSv/h limit for hands-on maintenance, only remote maintenance would be allowed at any of the considered locations and for all options. The vacuum vessel dose is dominated by Ta isotopes within the first year, Co isotopes up to 10 years and Nb isotopes at times more than 10 years following shutdown.

5. Waste Disposal Ratings

The waste disposal ratings (WDR) for the blanket and vacuum vessel of each of the three options have been determined using the calculated specific activities and the U.S. NRC 10CFR61 waste disposal concentration limits. The different radionuclide specific activities calculated by the DKR-ICF code were used in the analysis. As shown in Table 7, after a one year cooling period, the blankets of the two Li/V options qualify as Class C low level waste. In the meantime, the results indicate that the 316 SS/H_2O shield does not qualify for disposal as low level waste. The WDR for the vacuum vessel is higher than that for the blanket due to the lower long term activity produced in V and Be compared to 316 SS and inconel.

^{94}Nb, ^{59}Ni, ^{63}Ni, and ^{14}C are the major contributors to the waste disposal rating in each of the Li/V blankets. ^{63}Ni, ^{94}Nb, and ^{59}Ni dominate the WDR of the 316 SS/H_2O shield. On the other hand, the WDR of each of the vacuum vessels used are dominated by ^{94}Nb and ^{63}Ni. It is important to keep in mind that the vacuum vessel waste disposal ratings are obtained by averaging over the total volume of the vacuum vessel. Examining the WDR of the individual layers of inconel leads to a different conclusion. For example, the WDR of the front 3 cm of inconel wall in

Table 4. Contact Dose Rates (mSv/h) for the Self-Cooled Li/V Option.

Li/Li/V	Back FW		Back Blanket		Front VV	Back VV
Time	Maintenance	Radwaste	Maintenance	Radwaste	Maintenance and Radwaste	
shutdown	1.27E+08	1.05E+08	7.93E+07	2.52E+07	6.26E+07	5.88E+03
1 min	1.16E+08	9.64E+07	6.94E+07	2.28E+07	5.65E+07	5.84E+03
10 min	7.05E+07	6.27E+07	2.98E+07	1.30E+07	3.22E+07	5.65E+03
1 hour	5.80E+07	5.31E+07	1.98E+07	1.04E+07	2.51E+07	5.32E+03
6 hour	5.37E+07	4.93E+07	1.82E+07	9.60E+06	2.32E+07	4.64E+03
1 day	4.09E+07	3.75E+07	1.41E+07	7.29E+06	2.13E+07	4.37E+03
1 week	7.03E+06	6.05E+06	3.38E+06	1.19E+06	1.69E+07	4.11E+03
1 month	2.74E+06	2.13E+06	1.85E+06	4.41E+05	1.38E+07	3.47E+03
1 year	7.64E+05	6.60E+05	3.50E+05	1.35E+05	1.80E+06	4.43E+02
10 year	8.58E+03	2.17E+03	1.36E+04	1.06E+03	1.27E+05	2.69E+01
100 year	1.25E+01	2.63E+00	2.93E+01	6.92E-01	4.20E+02	3.97E-02
1000 year	1.21E+01	2.55E+00	2.83E+01	6.67E-01	4.07E+02	3.81E-02

Table 5. Contact Dose Rates (mSv/h) for the He-Cooled Li/V Option.

Li/He/V	Back of FW		Back Blanket		Front VV	Back VV
Time	Maintenance	Radwaste	Maintenance	Radwaste	Maintenance and Radwaste	
shutdown	4.77E+08	2.12E+08	7.00E+07	4.15E+07	4.40E+07	4.67E+03
1 min	4.19E+08	1.87E+08	5.99E+07	3.55E+07	3.92E+07	4.64E+03
10 min	1.83E+08	8.61E+07	1.93E+07	1.13E+07	1.97E+07	4.49E+03
1 hour	1.25E+08	6.07E+07	9.45E+06	5.56E+06	1.42E+07	4.23E+03
6 hour	1.15E+08	5.63E+07	8.67E+06	5.14E+06	1.31E+07	3.68E+03
1 day	8.78E+07	4.29E+07	6.90E+06	3.91E+06	1.21E+07	3.45E+03
1 week	1.37E+07	6.74E+06	2.26E+06	6.02E+05	9.96E+06	3.25E+03
1 month	4.59E+06	2.29E+06	1.49E+06	1.98E+05	8.27E+06	2.74E+03
1 year	1.41E+06	7.27E+05	2.44E+05	5.32E+04	1.15E+06	3.51E+02
10 year	3.88E+03	1.20E+03	1.24E+04	1.24E+02	7.48E+04	2.14E+01
100 year	1.17E+00	3.92E-01	3.72E+01	9.48E-02	3.03E+02	3.13E-02
1000 year	1.12E+00	3.78E-01	3.59E+01	9.15E-02	2.93E+02	3.00E-02

Table 6. Contact Dose Rates (mSv/h) for the 316 SS/H_2O Option.

316 SS/H_2O	Back FW		Back Blanket		Front VV	Back VV
Time	Maintenance	Radwaste	Maintenance	Radwaste	Maintenance and Radwaste	
shutdown	2.62E+08	1.06E+08	3.68E+06	2.40E+06	1.90E+06	1.69E+02
1 min	2.56E+08	1.03E+08	3.64E+06	2.37E+06	1.89E+06	1.68E+02
10 min	2.32E+08	9.39E+07	3.43E+06	2.23E+06	1.82E+06	1.63E+02
1 hour	1.96E+08	7.93E+07	2.81E+06	1.82E+06	1.63E+06	1.53E+02
6 hour	1.03E+08	4.23E+07	1.13E+06	7.10E+05	1.16E+06	1.33E+02
1 day	6.74E+07	2.83E+07	5.34E+05	3.13E+05	9.85E+05	1.25E+02
1 week	5.93E+07	2.49E+07	4.80E+05	2.79E+05	9.28E+05	1.18E+02
1 month	4.99E+07	2.09E+07	4.02E+05	2.34E+05	7.91E+05	9.94E+01
1 year	1.21E+07	5.11E+06	1.09E+05	6.73E+04	1.20E+05	1.28E+01
10 year	8.67E+05	3.40E+05	1.75E+04	1.14E+04	8.42E+03	7.73E-01
100 year	2.91E+01	1.20E+01	8.23E-01	2.80E-01	9.39E+00	1.13E-03
1000 year	1.64E+01	6.96E+00	4.78E-01	6.69E-02	9.01E+00	1.09E-03

Table 7. Class C Waste Disposal Ratings for the Blanket and VV (compacted values are between brackets).

WDR	**Blanket**	**VV**
Self-Cooled Li/V	0.036 [0.126]	52.2 [79.4]
He-Cooled Li/V	0.1 [0.29]	40.11 [60.9]
316 SS/H_2O	3.82 [4.77]	1.11 [1.67]

the inboard side of the vacuum vessel of the 316 SS/H_2O option is 25. As the neutron spectrum is softened by interaction with the 44 cm 316 SS/H_2O ball region between the 2 inconel layers, the WDR of the back 3 cm inconel wall drops to only 0.038 with the WDR of the 316 SS/H_2O layer in between at 0.0099. The WDR of the vacuum vessel on the outboard side showed a similar trend. Averaging over the total volume of the vacuum vessel (ib + ob) resulted in a WDR of only 1.11 (non-compacted) and 1.67 (compacted). Replacing inconel with another material should reduce the level of long-lived activities induced in the VV to a level at which the VV can be disposed of as low level waste.

Table 8. Whole Body Early Dose (Sv) at the Site Boundary (1 km) [The dose values are for 100% release of structure only].

Option	Blanket	VV
Self-Cooled Li/V	1088	3479
He-Cooled Li/V	3229	2073
316 SS/H_2O	10104	147

6. Off-Site Doses

The off-site doses produced by the accidental release of 100% of the radioactive inventory from the reactor containment building are shown in Table 8. The calculations used the worst release characteristics as defined by the ESECOM [7] methodology (class F wind stability, 1 m/s wind speed, etc.). However, since the existence of radioactivity does not in itself represent a safety hazard, the second step in any safety analysis should consider a set of pessimistic but rather credible accident scenarios for mobilizing and releasing the radioactive inventory. As shown in the table, the relatively high dose from the 316 SS/H_2O option is due to the much larger volume of structure used in this non-breeding blanket option. In a breeding blanket the 316 SS volume would be smaller and Li would absorb a large number of neutrons resulting in a significant reduction in the dose from 316 SS.

The off-site doses resulting from the two Li/V blanket options are dominated by the scandium isotopes, ^{48}Sc, ^{46}Sc, and ^{47}Sc. The dose from the 316 SS/H_2O shield option is dominated by the two manganese isotopes, ^{54}Mn and ^{56}Mn, and the three cobalt isotopes ^{58}Co, ^{60}Co and ^{57}Co. Finally, ^{182}Ta, ^{183}Ta, ^{54}Mn, and ^{99}Mo are the major contributors to the doses produced from the VV used with any of the three options.

7. Summary

Detailed activation analyses have been performed for the different ITER blanket design options considered in the U.S. ITER blanket option trade-off study. The options considered included a self-cooled Li/V option, a helium cooled Li/V option, and a water cooled 316 SS non-breeding shield option. Results showed that the He-cooled blanket produces a higher level of activity than that of the self-cooled blanket due to the larger structure content. At the same time, the vacuum vessel activity is lower for the He-cooled blanket option due to the larger neutron attenuation in the blanket. Since the activity in the 316 SS and inconel used in the vacuum vessel does not fall as rapidly with time as the activity in the vanadium and beryllium used in the blanket, the total blanket and vacuum vessel activity is higher for the self-cooled option, at times greater than an hour following shutdown. Shield activity and decay heat of the 316 SS/H_2O option are higher than those for the Li/V blankets due to the large amount (80%) of 316 SS used. On the other hand, the vacuum vessel activity is much lower than that with the Li/V blankets due to the better shielding performance obtained without a breeding blanket.

While both Li/V blankets qualify as Class C low-level waste, the results indicate that the 316 SS/H_2O shield does not qualify for disposal as low level waste. The waste disposal ratings of the vacuum vessels of the Li/V options are higher than those for the blankets due to the higher long-term activity produced in 316 SS and inconel compared to vanadium and beryllium. The 316 SS/H_2O option produces the highest off-site doses in case of the accidental release of 100% of its radioactive inventory. The relatively high dose from the 316 SS/H_2O option is due to the much larger volume of structure used in this non-breeding blanket option. Finally, assuming a 25 μSv/h limit for hands-on maintenance, only remote maintenance would be allowed for all options.

Acknowledgment

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