Neutronic Analysis of ITER Neutral Beam Test Bed

M.J.Loughlin

UKAEA Fusion, Culham Science Centre, Abingdon, Oxfordshire, OX14 1PR, UK
E-mail: michael.loughlin@ukaea.org.uk

ABSTRACT

The neutron dose arising during the operation of the ITER neutral beam test bed was calculated using the Monte-Carlo code MCNP. The activation of the injector components was determined using the inventory code FISPACT and the resultant gamma ray doses at the end of the test programme were determined. It is found that there would be a neutron dose during beam operations sufficient to entail restricted and monitored access to areas outside the shield wall. A careful assessment of the shielding is needed. The individual components of the injector are generally not activated excessively, i.e. their activation levels fall below the hands on limit within one year. However, the total gamma dose close to the vessel is substantially higher than any individual contact dose. This must be taken into consideration in maintenance planning.

1. INTRODUCTION

The commissioning programme of the ITER neutral beam injector will involve the production of a significant number of neutrons. This can lead to a potential radiation exposure to workers either directly, during beam operation, or as a result of neutron activation of the injector and the subsequent γ-ray emission. It is important that excessive activation of the injector is avoided so that there are no difficulties in subsequently transporting and deploying the injector at ITER. This paper describes calculations of neutron transport, activation and γ-ray transport used to estimate the direct neutron dose and the γ-ray doses at the end of the commissioning period.

The ITER neutral beam injector will accelerate deuterons to 1MeV. During commissioning the deuterons will be fired into a calorimeter which will therefore become loaded with deuterium. Further incoming beam particles will undergo the reactions $d(d,n)^3He$ and $d(d,p)^3H$. The neutrons are emitted with an energy of approximately 2.5MeV and this reaction is the most important means of neutron production from the test bed. The production of tritium (^3H) is itself a radiological concern but it also implies the further production of 14MeV neutrons by the tritons reacting with the deuterium in the calorimeter or with incoming beam deuterons. Calculations [1] indicate that the latter reaction is ~350 more likely than the former but five orders of magnitude less than the d(d,n) reaction. The d(d,n) reaction rate as a function of beam density was estimated to be 3.78×10^{12} $C^{-1}cm^{-2}$ while the $^3H(d,n)$ rate is 5.2×10^{7} $C^{-1}cm^{-2}$. This equates to 1.512×10^{18} n/s for a 40A beam.

The neutron transport and activation were calculated using the Monte-Carlo code MCNP[2] and the inventory code FISPACT[3]. Gamma-ray transport calculations were also carried out to determine the dose which results from the decay of activation products. The next section describes these calculations, section 3 describes the results and the conclusions are briefly discussed in the final section.

2 CALCULATIONS

2.1 THE NEUTRAL BEAM INJECTOR

The major components of the injector are the high voltage(HV) bushing, the beam source, the neutraliser, the residual ion dump and the calorimeter. These are all contained in a steel vacuum vessel. Each of these components was included in the MCNP model (figure 1). However some approximations were made since all the details are not required and the creation of the model is very time consuming.

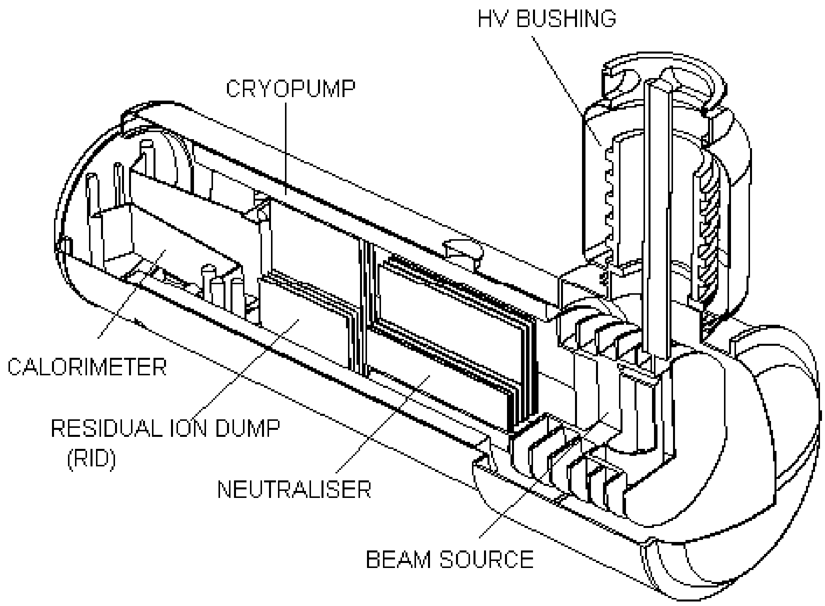

Figure 1: Cutaway iso-view through MCNP model of ITER neutral beam injector.

The vessel, which encloses all the other components, is 2.5cm thick stainless steel. The HV bushing is represented as alumina and water cooling and copper feed-through are included. The six accelerator stages of the beam source and the arc chamber are modelled but the grids are modelled as homogenous cells filled with reduced density copper. The insulators, electromagnetic shields, Sm-Co permanent magnets, support frame and other details are not included.

The Neutraliser has four vertical channels of rectangular cross-section, each 3 m long and 1.7 m high. The channels walls are formed by panels made of OFHC copper. The residual ion dump consists of five panels between two stainless steel side plates. There are inlet water pipes beneath the dump which are not modelled in detail. They are represented by a rectangular block containing water at a reduced density. The inlet and outlet manifolds are ignored.

The Calorimeter consists of two panels constructed of swirl tubes which carry cooling water. The coolant is fed in through a series of pipes. These are approximately modelled; the exact details of the pipe work is not considered to be critical but the inclusion of water is because it is a powerful moderator of neutrons.

The injector was housed in a test cell with interior dimensions of 30m x 30 x 20m and with 1.5m thick borated concrete walls.

2.2 THE NEUTRON SOURCE

The principle source of neutrons is the calorimeter, where the beam deposits deuterons creating a deuterium loaded target. Deuterium ions not neutralised in the neutraliser are deposited on the residual ion dump (RID). This therefore represents a second neutron source. Calculations in [1] indicate that 39% of the neutrons emanate from the RID. The deposition on the calorimeter and RID panels is not uniform and so the neutron production is distributed is a complex manner across these components. This non-uniformity was reproduced in the source representation in MCNP (see figure 2).

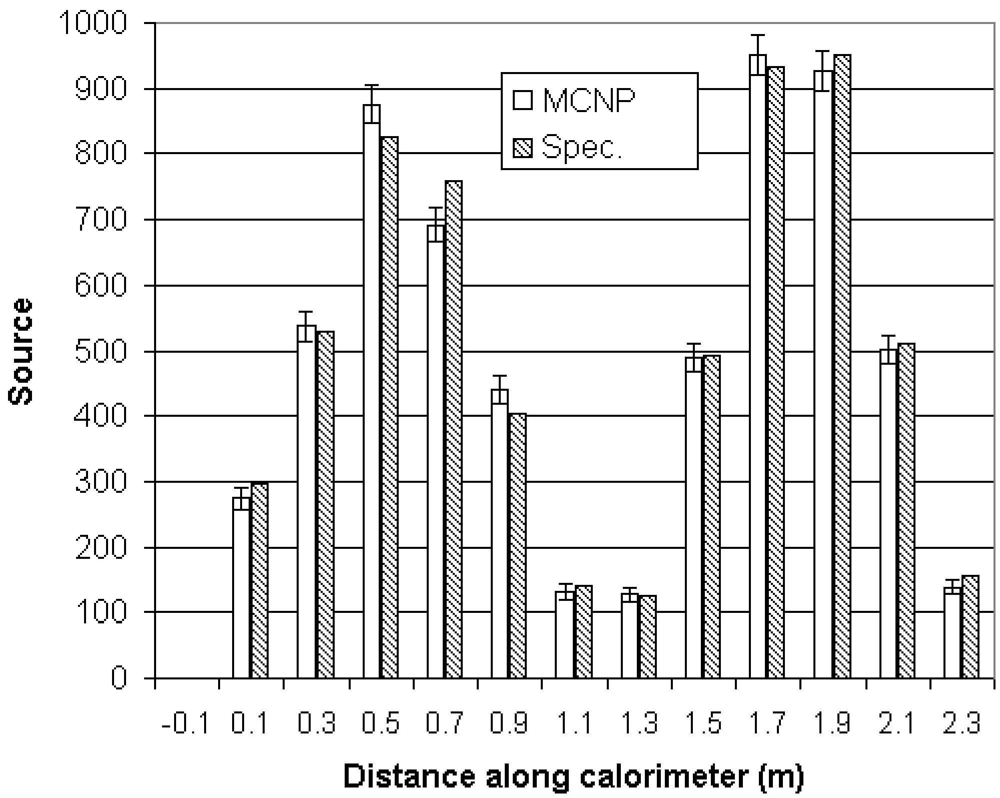

Figure 2: Comparison of the frequency of neutron emission along the calorimeter from MCNP and the specification (SPEC) from [1].

The Monte-Carlo code MCNP was used to determine the energy dependent neutron flux in all material cells. These results were then used in the subsequent activation calculations. MCNP was also used to estimate the neutron dose during beam operation at a location outside the test cell.

2.3 ACTIVATION

To compute the activation of the components, the energy dependent neutron flux and the irradiation history must be specified. The neutron spectra were available in Vitamin J energy group structure from the MCNP calculations described above. The irradiation, or neutron production history was based on a probable commissioning programme. This was specified as 300days with 100 pulses per day with each pulse lasting 20sec. This is followed by 14 days of a full power programme with 6 one hour pulses per day. Only the pulses in the full power programme were explicitly described in the activation calculations. The 300 day programme was described as a continuous irradiation at a level which conserved the total neutron production.

The calculations were carried out using FISPACT. This provided the activity, and contact dose rates for all components after cooling times of 1, 5, 10, 30, 60, 120 and 365 days. It also provided coarse descriptions of the gamma-ray spectra emitted from each cell. These were used as input to gamma dose calculations.

2.4 GAMMA DOSE CALCULATIONS

For maintenance purposes it will be necessary to know the gamma dose close to the injector which results from the gamma emission from all components and not just the contact dose for each component. The dose is estimated using a version of MCNP which takes the results of the FISPACT calculations i.e. the energy dependent gamma production in each cell, and carries out the gamma-ray transport calculations.

The transfer of results from MCNP to FISPACT and to MCNP is achieved automatically using a set of computer programmes described in [4].

3 RESULTS

3.1 DIRECT NEUTRON DOSE

The neutron dose during operation was calculated for a location outside a 1.5m thick borated concrete wall. The MCNP modelling shows that the dose is 3.724×10^{-19} Sv/hr/source neutron. For a source rate of 1.512×10^{14} n/s this would correspond to a dose of 56μSv/hr. However, the experimental programme assumes that the test bed operates with 100 pulses/day and each pulse lasts 20s. The dose averaged over a year assuming an eight hour day, a 5 day week, 52 week year is 2.7mSv. In the United Kingdom, a dose in excess of 6mSv/yr requires that the area be designated a controlled area. A lesser dose, but one which exceeds 1mSv/yr, requires designation of a supervised area. The results imply that the area outside the wall would have to be a supervised, but not controlled: but during the higher power final test the average dose rate is ten times higher, making this zone a controlled area.

3.2 COMPONENT ACTIVATION

The table lists those components which have activities above the IAEA limit of 7×10^4 Bq/kg (i.e. they would be classified as low level waste (LLW)) at one year after the end of test bed operation) and those which would give a contact dose in excess of 10^{-5} Sv/hr (the "hands on" limit). The numbers in brackets indicate the number of components above the limit, when this number is greater than 1.

Only the steel components are activated above LLW and hands on limits. The dominant nuclides are ^{51}Cr at two months, then ^{58}Co at 120 days and ^{55}Fe at one year. The main production pathways are ^{50}Cr(n,γ)^{51}Cr, ^{58}Ni(n,p)^{58}Co and ^{54}Fe(n,γ)^{55}Fe. A low activation steel such as Eurofer with less chromium and no nickel could be considered to reduce the activation.

Activation without final test programme - The neutron production in the last two weeks of the programme produces approximately half of the number of neutrons that are produced during the previous 300 day campaign. If this final section of the commissioning is eliminated the activity of the steel is reduced by just over a factor of two after 120 days, this has reduced to a factor of 1.65 at one year (see figure 3).

However, this would not greatly reduce the cooling period required before the activity drops below the LLW limit.

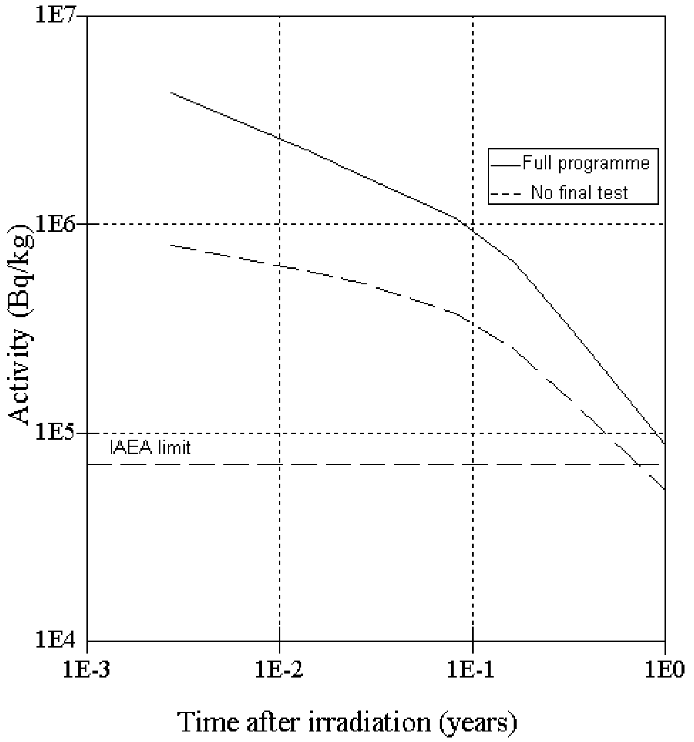

Figure 3: Activity of steel in vessel as a function of time assuming a full commissioning programme and a limited programme with no final full power test.

3.3 GAMMA DOSE

The gamma dose in the environment surrounding the test bed is the result of the gamma emission from all components. The vacuum vessel may be expected to be the dominant source of radiation on the vessel exterior because it is the closest and it will attenuate the gamma produced by internal components. However, internal components are the most highly activated so a full modelling of the radiation transport is essential.

The dose was computed in three spheres next to the vacuum vessel at locations in the injector mid-plane and next to the calorimeter, the residual ion dump (RID) and the neutraliser following 10 days of cooling time after the end of operations. Table 2 lists the results. The total dose is 3.5 to 5.5 times higher than the contact doses on the sections of the vessel next to the tally locations.

4 CONCLUSIONS

Calculations of the neutron dose during beam operations indicate levels which would requires restricted and monitored access to the area outside the shield wall if the walls were 1.5m thick borated concrete. This results implies that a careful assessment of the shielding is needed.

The individual components of the injector are generally not activated excessively, i.e. their activation levels fall below the hands on limit within one year. However, the total gamma dose close to the vessel shortly after (10days) the end of commissioning is substantially higher than the vessel contact dose. A longer cooling period may be needed to reduce the workers' exposure.

It should be noted that these calculations have assumed that the injector operates at full power throughout the commissioning period. This will not be the case. Early operations will be at reduced power so the neutron emission is over estimated. These doses and activities should therefore be regarded as the expected maxima.

ACKNOWLEDGEMENTS

This work was funded jointly by the United Kingdom Engineering and Physical Research Council and by Euratom.

REFERENCES

[1] A.Emmanouilidis and S.Cox Calculation of Neutron Yield and Isotope Retention in the ITER NB Test Bed, UKAEA/Euratom Association Report, July 2004

[2] Judith F. Briesmeister, editor, "MCNPTM -- a General Monte Carlo N-Particle Transport Code" Los Alamos National Laboratory Report, LA-13709-M (March 2000), version 4C3.

[3] R. Forrest "FISPACT 2003: User Manual" *UKAEA Fusion, UKAEA Report*, FUS 485, Dec. 2002.

[4] Y. Chen, U. Fischer "ITER-FEAT Shutdown Dose Rate Analysis by Rigorous Method" *Final report on Contract EFDA/00-564*, FZK Interner Bericht IRS-Nr 9/01-Fusion Nr 177, Sept. 2001. See also: Y.Chen, U. Fischer: "Rigorous MCNP based shutdown dose rate calculations: Computational scheme, verification calculations and applications to ITER" , Fusion Engineering and Design 63-64 (2002), 107-114.

Material	**LLW after 1 year**	**Above hands on limit after one year**
Steel	Vessel around calorimeter	
	Horizontal lip of vessel at fast shutter end	
	End plate	
	Calorimeter- Support structure (2)	Calorimeter- Support structure
	RID Wall (2)	RID Wall (2)
Water & steel	Pipework around calorimeter (12)	Calorimeter lower pipework
	Pipework around RID	
	Cooling pipe (2)	
CuCrZr	none	none
Copper	none	none
Aluminium	none	none
Alumina	none	none

Table 1: Components of neutral beam injector which exceed the low level waste (LLW) limit after one year and those which are still above the hands on limit after one year.

Location	**Dose (mSv/hr)**	**Contact dose (mSv/hr)**
Calorimeter	0.430 (±0.010)	0.123
RID	0.222 (±0.009)	0.053
Neutraliser	0.049 (±0.003)	0.009

Table 2: Total environmental dose at locations next to the vacuum vessel and contact doses on adjacent sections of the vessel.