CALCULATION OF THE ABSOLUTE DETECTION EFFICIENCY
OF A MODERATED ^{235}U NEUTRON DETECTOR ON THE
TOKAMAK FUSION TEST REACTOR

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Calculation of the Absolute Detection Efficiency of a Moderated ^{235}U Neutron Detector on the Tokamak Fusion Test Reactor

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Abstract

Neutron transport simulations have been carried out to calculate the absolute detection efficiency of a moderated ^{235}U neutron detector which is used on the TFTR as part of the primary fission detector diagnostic system for measuring fusion power yields. Transport simulations provide a means by which the effects of variations in various shielding and geometrical parameters can be explored. These effects are difficult to study in calibration experiments. The calculational model, benchmarked against measurements, can be used to complement future detector calibrations, when the high level of radioactivity resulting from machine operation may severely restrict access to the tokamak.

We present a coupled forward-adjoint algorithm, employing both the deterministic and Monte Carlo sampling methods, to model the neutron transport in the complex tokamak and detector geometries. Sensitivities of the detector response to the major and minor radii, and angular anisotropy of the neutron emission are discussed. A semi-empirical model based on matching the calculational results with a small set of experiments produces good agreement ($\pm 15\%$) for a wide range of source energies and geometries.

I. Introduction

Moderated ^{235}U neutron detectors have been used on the Tokamak Fusion Test Reactor (TFTR) as the primary diagnostic for the measurement of fusion neutron source strength, i.e., fusion power produced.¹ The ^{235}U fission counter offers fast response (high count rates), good γ -ray discrimination, and high relative efficiency. Since the average energy of neutrons in the vicinity of the tokamak is above the major capture resonances for ^{235}U fission, a moderator and neutron slowing down medium made of polyethylene and lead has been provided for the fission counter. Lead is effective in slowing down fast neutrons. It also helps to further reduce the count rates due to fission events that might result from high energy X-rays produced by runaway electrons in the plasma (< 50 MeV) and from high energy γ -rays produced by neutrons interacting with machine components and count rates resulting from pulse pile up. Three ^{235}U detectors are used on the TFTR, two with 1.3 g of ^{235}U loading and one with 18 g of fissionable material. These detectors have been used to measure the time evolution of the neutron emission rate since 1985.² They were also used to obtain the average ion temperature of the plasma during the early stage of the TFTR operation when the neutrons were mainly produced by thermonuclear reactions,³ and not mainly by beam-target reactions as is the case generally with Neutral Beam Injection Heating.¹

The fission detectors, however, lack energy resolution. Neutrons of all energies may be detected. Accurate measurements of neutron yield from fusion reactions in the plasma can only be obtained if the detectors are calibrated in-situ, for which the scatterer and source geometry are identical to those in the measurement, and by using the correct neutron source energy spectrum. Such absolute calibrations on TFTR have been accomplished by using either a ^{252}Cf spontaneous fission neutron source or a D-D or D-T neutron generator placed inside the vacuum chamber at a set of discrete locations.⁴ The total detector response was obtained by summing up the responses from the sources at each location.⁵ A similar procedure was used to measure the neutron and γ -ray fluence spectral distributions in the TFTR test cell using a NE-213 detector system.⁶

Calibration experiments using a point source to simulate the extended plasma neutron source are difficult to carry out, and are time consuming. The limited space inside the tokamak vacuum vessel restricts the freedom of approximating the plasma geometry with a neutron generator. The number of discrete points that can be selected and the accuracy of the counting statistics that can be achieved may also be limited because of the amount of time required to carry out the experiments. Activation of the vacuum vessel is now approaching a few mrem per hour even after weeks of cooling,⁷ and with the induced radioactivities of the machine increasing, extensive

calibration experiments will be more difficult to carry out in the future. Developing a simulation model to complement in-situ calibration experiments is evidently needed. Using neutron transport simulations to calculate neutron source strength of the plasma from the fluence measured by activation foils was the method chosen by the Joint European Torus (JET) for their neutron activation diagnostic system for determining the absolute fusion neutron yield.⁸

A simulation model must be able to carry out numerical calibration experiments as close to the real geometry as possible, and should be sufficiently flexible to allow the study of various parametric variations. At present, the Monte Carlo method is the only established technique that can be used to solve neutron transport in TFTR with reasonably detailed modeling of the complex geometries. To obtain the detector fission rate in the simulation, neutrons incident on the detector assembly would have to be further transported through the moderator. Since the surface area of the detector assembly is only a fraction of the total surface area at the major radius where the detector is located, the chances that neutrons would cross the detector surface are small. Clearly, a direct calculation would be very inefficient. Thus, we chose to derive a "surface" response function for the detector-moderator assembly instead. The total detector response is obtained by integrating the incident neutron fluxes times the detector surface response function over the surface of the detector assembly. Since the particle transport through the moderator is not directly simulated, the efficiency of Monte Carlo calculations for neutron transport can be increased by the use of a pseudo-collision method (or so-called last flight estimator) for the flux calculation at the detector.⁹

To determine the surface response of the detector-moderator assembly, transport calculations in the adjoint mode were carried out. Because of the relatively simple geometry, the discrete-ordinates method of solving the Boltzmann equation can be used. As shown in Section IV, the surface response function can be represented by the adjoint function on the surface of the moderator-detector assembly. Hence, the total fission rate is simply the surface integration of the product of the forward neutron fluxes calculated by the Monte Carlo method and the adjoint function calculated for the detector. This coupled forward-adjoint model for the numerical simulation is a powerful tool for the study of the fission detector response on the TFTR.

A brief description of the fission detector system used on the TFTR for neutron monitoring is given in Section II. In Section III we review the theory of the response calculation using surface coupling of the forward and adjoint fluxes. The calculation of the detector-moderator surface response is presented in Section IV, where comparisons are made with results from a set of experiments using a point source in a simple geometry. Section V summarizes the numerical simulations of the in-situ calibration experiments on TFTR, and compares results of calculations and measurements car-

ried out between 1983 and 1987. Studies of the effects of the source major and minor radii, as well as the anisotropy of the neutron emission, on the detector response, are given in Section VI.

II. ^{235}U Fission Detector System on the TFTR

The ^{235}U fission detector used on the TFTR is the Reuter-Stokes type RS-C3-2510-114, which contains 1.3 g of ^{235}U distributed over a sensitive area of 7 cm in diameter and 23.5 cm in length. It has a dynamic range of 10^3 to $5 \cdot 10^5$ cps in the count mode, and 10^5 to 10^9 equivalent cps in the current and mean-square-voltage modes, using Gammametrics¹⁰ electronics. The detector is surrounded by an electrostatic shield made of 305 stainless steel, 0.35 cm in thickness. The moderator assembly wall thickness consists of 9.95 cm of lead and 5.05 cm of polyethylene. The outer surface is covered with boron panels. The detector assembly has a die-away time of $\leq 100\mu\text{s}$. A schematic diagram of the detector assembly is shown in Fig. 1.

There are two such detectors used, designated as NE-1 and NE-2. NE-1 is located in Bay C, 54° from due north, on the machine substructure behind the lower shear compression panel, 5.4 m from the center of the machine and 1.85 m below the midplane. The vacuum pumping duct for the TFTR vessel passes right above the detector. The shear compression panel has an equivalent steel thickness of ~ 0.25 m. Thus, there is virtually no uncollided neutron reaching the detector assembly. The other detector, NE-2, is located in the opposite bay. In this paper, we shall only discuss the simulation of NE-1 responses.

III. Forward-Adjoint Formulation for Calculating the Detector Response

Since there is no time-dependence, and the number of fission events is small, neutron transport in the detector assembly can be described by the homogeneous linear Boltzmann equation

$$\mathcal{L}\phi(\vec{\Omega}, E, \vec{r}) = 0, \quad (1)$$

with the boundary source

$$\phi(\vec{\Omega}, E, \vec{r}) = \phi_b(\vec{\Omega}, E, \vec{r}) \text{ for } \vec{n} \cdot \vec{\Omega} < 0 \\ 0 \text{ otherwise} \quad (2)$$

defined at the outer surface of the assembly. Here, $\mathcal{L}$ is the neutron transport operator defined as

$$\begin{aligned}\mathcal{L}\phi(\vec{\Omega}, E, \vec{r}) &= \vec{\Omega} \cdot \nabla \phi(\vec{\Omega}, E, \vec{r}) - \Sigma(E, \vec{r})\phi(\vec{\Omega}, E, \vec{r}) \\ &+ \int \int \Sigma(E', \vec{r})f(\vec{\Omega}', E' \rightarrow \vec{\Omega}, E; \vec{r})\phi(\vec{\Omega}', E', \vec{r})d\vec{\Omega}'dE',\end{aligned}\quad (3)$$

where, $\phi(\vec{\Omega}, E, \vec{r})$ is the neutron angular flux at $\vec{r}$, with direction $\vec{\Omega}$ and energy E . Σ is the macroscopic total collision cross section, f is the probability of scattering from $(\vec{\Omega}', E')$ to $(\vec{\Omega}, E)$, and $\hat{n}$ is the unit vector outward normal to the surface.

Now consider the inhomogeneous adjoint equation for neutron transport in the detector assembly, with the fission response, $\Sigma_f(E, \vec{r})$, as the source, where $\Sigma_f(E, \vec{r})$ is the macroscopic fission cross section,

$$\mathcal{L}^*\phi^*(\vec{\Omega}, E, \vec{r}) = -\Sigma_f(E, \vec{r}).\quad (4)$$

Here, $\mathcal{L}^*$ is the transport operator adjoint to $\mathcal{L}$ and $\phi^*(\vec{\Omega}, E, \vec{r})$ is the adjoint function satisfying the equation

$$\begin{aligned}\mathcal{L}^*\phi^*(\vec{\Omega}, E, \vec{r}) &= \vec{\Omega} \cdot \nabla \phi^*(\vec{\Omega}, E, \vec{r}) - \Sigma(E, \vec{r})\phi^*(\vec{\Omega}, E, \vec{r}) \\ &+ \int \int \Sigma(E', \vec{r})f(\vec{\Omega}, E \rightarrow \vec{\Omega}', E'; \vec{r})\phi^*(\vec{\Omega}', E', \vec{r})d\vec{\Omega}'dE',\end{aligned}\quad (5)$$

under the standard free-surface boundary condition.¹¹ From Eqs. (1) and (4), we have

$$\int \int \int \Sigma_f(E, \vec{r})\phi(\vec{\Omega}, E, \vec{r})d\vec{\Omega}dEdV = \int \int \int (\hat{n} \cdot \vec{\Omega})\phi_b(\vec{\Omega}, E, \vec{r})\phi^*(\vec{\Omega}, E, \vec{r})d\vec{\Omega}dEdA.\quad (6)$$

Thus, the fission rate in the detector given by the left-hand side of Eq. (6) can also be obtained by the inner product of the forward angular flux and the adjoint function on the boundary surfaces shown on the right-hand side of Eq. (6). Note that ϕ^* depends only on the property and configuration of the detector and moderator assembly, whereas ϕ depends on the neutron source energy as well as the source-detector geometry. Once ϕ^* is determined, only the forward neutron flux incident on the surface of the detector assembly needs to be calculated in order to determine the detector efficiency.

IV. Surface Response of the Moderator-Detector Assembly

A. Detector surface response function

We define the surface response function, $\epsilon_s(E)$, as the count rate per unit flux incident on the surface of the detector.

$$\epsilon_s(E) = \epsilon_i \cdot \int \int_{\vec{n} \cdot \vec{\Omega} < 0} \phi(\vec{\Omega}, E, \vec{r}) \phi^*(\vec{\Omega}, E, \vec{r}) d\vec{\Omega} dA / \int \phi(\vec{\Omega}, E, \vec{r}) d\vec{\Omega}, \quad (7)$$

where ϵ_i is the detector intrinsic efficiency (counts per fission event), $\phi(\vec{\Omega}, E, \vec{r})$ is the forward neutron angular flux, and $\phi^*(\vec{\Omega}, E, \vec{r})$ is the adjoint function on the outer surface of the detector and moderator assembly. The integration in the numerator is carried out over the surface of the detector assembly in the domain $\vec{n} \cdot \vec{\Omega} \leq 0$. The absolute detector efficiency, ϵ , defined as the count per source neutron, in a particular experimental setup, is then

$$\epsilon = \int \epsilon_s(E) \phi(E, \vec{r}) dE, \quad (8)$$

where $\phi(E, \vec{r})$ is the scalar neutron flux at $\vec{r}$.

In a complex geometry it is very difficult to calculate $\phi(\vec{\Omega}, E, \vec{r})$ in Eq. (7) to within a satisfactory degree of accuracy. Normally, the scalar neutron flux is calculated instead. By defining $\phi^*(\vec{\Omega}, E, \vec{r}) = \vec{n} \cdot \vec{\Omega}$ as the adjoint current in Eq. (7), however, we may regard ϵ_s as the weighted adjoint current integrated over the surface of the detector assembly. The weighting function is proportional to the forward neutron flux incident on the detector. This concept would allow us to make some approximation to the form of the weighting function, and to decouple the forward neutron transport calculation for a specific geometry from the determination of the detector surface response given by Eq. (7). There are two special weighting functions that are of particular interest:

$$\phi(\vec{\Omega}, E, \vec{r}) = \phi(E, \vec{r}) \cdot \delta(\vec{\Omega} - \vec{\Omega}_0), \quad (9)$$

and

$$\phi(\vec{\Omega}, E, \vec{r}) = \text{const} \cdot \phi(E, \vec{r}). \quad (10)$$

Equation (9) is applicable to the calibration of the fission detector in the neutron laboratory where the incident flux on the detector is almost monodirectional due to uncollided neutrons from a point source. Equation (10) is applicable to the calibration of the fission detector in the actual machine where the neutron flux is approximately isotropic due to multiple scatterings.

B. Numerical calculation of $\epsilon_s(E)$

To derive the numerical values of $\epsilon_s(E)$, we carried out neutron transport calculations using a one-dimensional representation of the detector-moderator assembly described in Section II. In this model all the components were represented as infinite slabs, as shown in Fig. 2. The leakage due to the finite size in the axial and traverse directions can be approximately accounted for by diffusion buckling corrections.¹² Since the axial dimension is much larger than the flux relaxation length, the one-dimensional model is a reasonable approximation of the detector geometry.

The transport calculation was carried out using the ANISN code,¹³ which solves the Boltzmann transport equation in one-dimensional geometry, using the multi-group, discrete-ordinates (S_n) method. The energy space is divided into groups, with the average properties represented or calculated for each group. The angular directions of neutron transport are represented by a set of discrete "rays." We used the standard 25 group neutron cross sections in the PPPL nuclear data library,¹⁴ and P_3 Legendre expansions for the anisotropy in nuclear cross sections and angular flux.

The computed ϵ_s for 14 MeV neutrons are $\sim 6.7\%$ and 1.5% for the monodirectional (with $n \cdot \Omega \approx -1.0$) and isotropic fluxes, respectively. The efficiencies at other neutron energies relative to the efficiency at 14 MeV, assuming isotropic flux, are illustrated in Fig. 3. In deriving the $\epsilon_s(E)$, we have taken the intrinsic efficiency to be 0.25. This intrinsic efficiency is that at the minimum discriminator voltage setting to reject electrical and background α and γ noises.¹⁵ Note that ϵ_i depends on the burnup of the fissionable material in the detector and the setting of the detector electronics, and may vary as a function of time. $\epsilon_s(E)$ should be renormalized if ϵ_i in subsequent recalibrations is different from that used here.

We observe that the detector response is remarkably insensitive to the energies of the neutrons incident on the moderator, varying by only 20% over a range of 14 MeV to ~ 10 eV. This attractive feature provides many advantages in both the neutron measurement on TFTR and the numerical simulation work, as will be seen in the following sections.

C. Comparison of calculated $\epsilon_s(E)$ with measurement

Calibration experiments with simple geometries were conducted in 1982. The experiments involved a ^{252}Cf spontaneous fission source in a large concrete room, with the detector assembly placed 2.1 and 4.2 m from the source. In this case, the neutron

fluxes at the detector can be calculated analytically if the scattered component is negligibly small. Using the spectrum evaluated by the National Bureau of Standards for the ^{252}Cf fission neutrons¹⁶ and the surface response function derived above, we obtained a surface efficiency of 6.6%. The measurement gave a value of $(7.3 \pm 0.23)\%$. The discrepancy is attributable to the added contribution from neutrons scattered from the floor. This is evidenced from the observation that the measured count rates, when normalized to the uncollided flux, tend to increase at larger distances. For example, at 4.2 m the surface efficiency was increased to 8.8%. It can be deduced from these comparisons that the scattered fluxes accounted for about 10% of the direct flux at the detector 2.1 m from the source.

V. Transport Simulations of Calibration Experiments on TFTR

A. In-situ calibration experiments

In-situ calibration experiments were carried out in 1983 and 1985, using three point sources, ^{252}Cf ($2 \cdot 10^8$ n/s), a DD neutron generator ($\leq 5 \cdot 10^8$ n/s) and a DT neutron generator ($\sim 5 \cdot 10^{10}$ n/s), to derive the absolute detection efficiency, taking into account the scattering and shielding effects of all the TFTR machine and diagnostic components. Another calibration was performed in 1987, but only with a DD neutron generator to check possible changes due to modifications of components close to the tokamak (which is an on-going activity), and due to drifts in the electronics or variations of other components. These point sources were placed inside the TFTR vacuum vessel at many locations to simulate the extended plasma. One example is given in Fig. 4 which shows the arrangement of the 1983 calibration with a DD neutron generator. The arrows indicate the direction of the ion beam ($\vec{v}_i$) in the neutron generator. The ion beam direction was reversed in the 1987 experiment. Since the neutron emission from a generator source is not isotropic (peaking in the forward direction), the orientation of the beam direction will change the results, as we shall discuss later. The mean energies of the ^{252}Cf , DD, and DT sources are 2.0 MeV, 2.15 MeV, and 14.07 MeV, respectively.

The neutron emission from the DD and DT generator was monitored by a small ^{238}U fission chamber mounted on the generator head. This ^{238}U detector was fabricated and calibrated at the National Bureau of Standards.¹⁷ The uncertainties in the ^{252}Cf source calibration and the monitoring detector were $\sim \pm 3\%$, and the overall errors of the experiment were $\sim \pm 15\%$.¹⁸

A typical result is given in Fig. 5, which shows the detector efficiency as a function of toroidal angle of the source position measured relative to the location of the detector. We observe that the curve is almost symmetric with respect to the 0° toroidal angle, although the machine components are not distributed symmetrically, and that the curve drops off rapidly, almost exponentially, away from the 0° angle. The symmetry in the efficiency curve can be attributed in part to the insensitivity of the detector response to the energy of the incident neutrons. Also, since most neutrons contributing to the detector response suffer many collisions, with average energy loss per collision of only $\sim 7\%$, the effect due to nonuniform distribution of the materials in the TFTR will be smoothed out. The rapid drop-off of the response curve can be understood from the fact that the average material density around the machine is relatively low so that there are many leakage pathways for neutrons to escape from the local environment, and that more neutrons will be attenuated as the distance between the source and detector is increased due to the increased effective material thickness. Thus, mainly those neutrons produced in the plasma regions near the detector are likely to be detected.

B. Transport simulations of in-situ experiments

To provide the modeling capability for a point source in the TFTR vacuum vessel, a three-dimensional computer model suitable for the Monte Carlo calculation was generated.¹⁹ A cross section of the model is shown in Fig. 6. The vacuum vessel, center column assembly, toroidal field (TF) coils, shear compression panels between the TF coils, as well as the floor and test cell walls are explicitly modeled. In addition, the stainless steel columns supporting the upper steel umbrella structure, which in turn supports the upper poloidal field coils, are also explicitly modeled in the bay where the detector is located (bay C). The diagnostic instruments, and other minor components, which are made of mostly stainless steel, are modeled as homogeneous materials around the machine, with density factors appropriate for different regions, as discussed in the next subsection. Homogenization is a good approximation because neutrons reaching the detector after suffering many collisions with various components of the machine will not "remember" the details of the events encountered in their life histories and because the detector response is insensitive to the energies of the incoming neutrons. Homogenization not only avoids the difficult problem of describing the components in detail but also speeds up the Monte Carlo computation by simplifying the geometrical tracking of particles.

The neutron transport calculations were carried out using the MCNP code.²⁰ The last flight estimator was used to calculate the flux at the detector location in an energy group structure consistent with that used to derive the detector efficiency.

Various variance reduction techniques were invoked to speed up the calculations, and most calculations were terminated when the fractional standard deviation fell below 8%. The angular distribution of the neutrons emitted by the ^{252}Cf was assumed to be isotropic, and those of the DD and DT neutron sources from the generator were taken from unpublished measurements performed together with the 1983 calibration.

C. Comparisons between calculations and measurements

A comparison of results for a DD generator source located at toroidal angles of 0, 30, and 60 degrees with respect to the detector location is given in Table 1. The beam direction of the generator was taken to be pointing away from the detector (i.e., $\vec{e}_s \cdot \vec{r}_d < 0$, where $\vec{r}_d$ is the radial vector for the detector). The calculated efficiencies for various material densities in the homogenized zones are shown. We see that the responses are not sensitive to the steel fractions in the range assumed, and that the results are generally in reasonably good agreement with the measurements. The best agreement is obtained for 5% of steel in the region between the vacuum vessel and the TF coils and 20% of steel plus 2% of water equivalent hydrogen in the region outside the TF coils but inside the pedestal on which the machine and supporting structures stand (case 5). Since for a toroidally uniform source those regions in the forward 60° contribute to ~80% of the detector response, we choose the material densities corresponding to case 5 as the baseline model for further comparison and analysis.

The comparison of results due to a toroidal ring source is given in Table 2. The results are given for three different source spectra: ^{252}Cf , DD, and DT. The measured data were those from the calibration experiments in 1983 and 1985, with the generator pointing away from the detector. The measured count rates for the source at 8 toroidal locations were graphed, and the absolute efficiencies obtained by integrating the curves using log-linear interpolation. These measured values have been the basis of deriving neutron source intensities on the TFTR in the past few years. It is seen that within the statistical errors the calculations using the baseline model are in good agreement ($\pm 5\%$) with measurements. If comparisons are made for each individual location, the differences are typically $< 15\%$. There is no significant difference in the detector absolute efficiency found between DD and ^{252}Cf sources. The detector efficiency is ~20% higher for a DT source. This higher response is due to interactions involving neutron multiplicity, such as the (n,2n) reactions, possible only for neutrons with energies greater than 10 MeV, and to the increase in slowing down length as the effective source energy is increased.

The effect of generator orientation has also been examined for various source

toroidal angles. The calculations showed that, for example, there is a 20% difference in the detector absolute efficiency between two opposite orientations of the DD generator located at the 30° angle ($1.18 \cdot 10^{-8} \pm 5\%$ vs. $1.00 \cdot 10^{-8} \pm 6\%$), in agreement with the experimental results from in-situ calibrations.²⁰

VI. Results Derived from Other Source Geometries

Since neutrons produced in a fusion plasma invariably have a radial intensity distribution, it is important to know the change in detector efficiency for different plasma major and minor radii. Table 3 gives some results for an isotropic DD source. The results indicate that the detector efficiency is higher when the center of plasma is nearer to the detector, but the size of the plasma at a given position has lesser effects on the detector response.

The effects of angular anisotropy in the neutron emission for a DD toroidal ring source are shown in Table 4. The angular distribution used was that measured for the DD neutron generator. It is seen that the count rate is lowest when the direction of forward neutron emission is oriented away from the detector, although the difference is less than 10% compared to the isotropic case. No difference from the isotropic case is observed when the ion beam direction of the generator is oriented toward the detector in one half of the torus and away from it in the other half of the torus (single beam direction for the entire torus).

VII. Summary and Conclusions

We have developed a model to simulate the absolute responses of a moderated ^{235}U neutron detector used on the TFTR for neutron yield measurements. The neutron scalar flux at the surface of the detector assembly is calculated using Monte Carlo simulations, and the surface response of the detector assembly is calculated using the discrete ordinates method in the adjoint mode, with the macroscopic fission cross section as the source. The detector absolute efficiency is the inner product of these two quantities. The minor machine components and diagnostic instruments surrounding the tokamak are represented as two homogeneous regions. By adjusting the densities in these regions to match the calculational results with a small set of DD experiments, we were able to develop a simulation model that gave good results for a wide range of source energies and geometries ($\pm 15\%$). Neutron transport simulation using this "semi-empirical" model promises to be a useful tool to complement future re-calibration experiments which will gradually become more difficult because of the

increased level of residual radioactivity in the machine.

The study confirms that the detector and moderator arrangement is such that the fission rate of the detector is remarkably insensitive to the energies of the neutrons incident on the detector. Thus the perturbation due to nonuniform distribution of the objects around the tokamak will be smoothed out so long as the neutrons reaching the detector suffer a significantly large number of scattering events and there is no strong shadow/shielding effects from local objects. Although the TFTR has gone through many modifications with some hardware being installed and removed every year, the flatness of the detector response alleviates the necessity of calibrating the detector every time a modification is done. However, recalibration or renormalization, not in-situ but with a neutron source adjacent to the detector (but outside the vacuum vessel), may be of interest to detect electronics drifts.

The studies showed that the detector efficiency is sensitive to the position of the source. Thus, more data points concerning the effects of plasma position variation should be taken during the next major re-calibration, which should include complete toroidal traverses of the point source at different major radii and elevations.

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Case No.	x_1^*	x_2^*	$\phi = 0^\circ$	$\phi = 30^\circ$	$\phi = 60^\circ$
measured	-	-	$1.7 \cdot 10^{-8}$	$9.0 \cdot 10^{-9}$	$4.0 \cdot 10^{-9}$
1 (a)	0.00	0.00	$1.2 \cdot 10^{-8} \pm 5\%$	$8.6 \cdot 10^{-9} \pm 5\%$	$4.8 \cdot 10^{-9} \pm 8\%$
2 (b)	0.05	0.05	$1.8 \cdot 10^{-8} \pm 7\%$	$9.9 \cdot 10^{-9} \pm 11\%$	$5.8 \cdot 10^{-9} \pm 14\%$
3 (c)	0.05	0.05	$1.9 \cdot 10^{-8} \pm 5\%$	$1.3 \cdot 10^{-8} \pm 8\%$	$5.9 \cdot 10^{-9} \pm 7\%$
4 (c)	0.05	0.05-0.02(H ₂ O)	$1.7 \cdot 10^{-8} \pm 5\%$	$1.1 \cdot 10^{-8} \pm 8\%$	$4.9 \cdot 10^{-9} \pm 9\%$
5 (c)	0.05	0.20-0.02(H ₂ O)	$1.7 \cdot 10^{-8} \pm 6\%$	$1.0 \cdot 10^{-8} \pm 6\%$	$4.8 \cdot 10^{-9} \pm 7\%$
6 (c)	0.10	0.20-0.02(H ₂ O)	$1.3 \cdot 10^{-8} \pm 6\%$	$8.7 \cdot 10^{-9} \pm 8\%$	$4.8 \cdot 10^{-9} \pm 14\%$

Note:

- (*) x_1 is the steel volume fraction between vacuum vessel and toroidal field coils.
- (*) x_2 is the steel volume fraction between toroidal field coils and R=6 m.
- (a) The global model used for general shielding calculation.
- (b) A 30 cm radius sphere at the detector location is excluded from homogenization.
- (c) Bay C is excluded from homogenization in x_2 , instead columns are modeled.

Table 1: The calculated absolute efficiencies (counts per source neutron) for the ^{235}U neutron detector in TFTR due to neutrons from a DD generator. The direction of the generator beam ion is tangential to the major radius at R=2.55 m, pointing away from the detector.

	^{252}Cf	DD Neutrons	DT Neutrons
Measured	$4.61 \cdot 10^{-9} \pm 5\%$	$4.80 \cdot 10^{-9} \pm 5\%$	$5.42 \cdot 10^{-9} \pm 5\%$
Calculated	$4.65 \cdot 10^{-9} \pm 6\%$	$4.60 \cdot 10^{-9} \pm 5\%$	$5.50 \cdot 10^{-9} \pm 5\%$

Table 2: Comparison of calculated versus measured ^{235}U neutron detector absolute efficiencies (counts per source neutron) for three different sources. The measured values were those obtained in the 1983 calibration. In the calculation, the source has been assumed to be a filament with a major radius of 2.55 m. The DD and DT neutrons are from a generator with the ion beam direction oriented away from the detector.

R_p (m)	r_p (m)	Z_p (m)	ϵ (c/n)
2.55	0.0	0.0	$5.1 \cdot 10^{-9} \pm 5.5\%$
2.55	0.0	0.3	$5.0 \cdot 10^{-9} \pm 5.0\%$
2.55	0.0	-0.3	$5.1 \cdot 10^{-9} \pm 6.0\%$
2.55	0.4	0.0	$4.8 \cdot 10^{-9} \pm 5.2\%$
2.85	0.0	0.0	$5.6 \cdot 10^{-9} \pm 6.0\%$
2.85	0.4	0.0	$5.6 \cdot 10^{-9} \pm 7.0\%$
2.25	0.0	0.0	$5.0 \cdot 10^{-9} \pm 6.0\%$
2.25	0.4	0.0	$4.7 \cdot 10^{-9} \pm 5.0\%$

Table 3: Comparison of the calculated detector absolute efficiency (counts per source neutron) for various plasma major radii (R_p), minor radii (r_p), and vertical positions (Z_p) for an isotropic DD source.

Neutron Generator Orientation		ϵ
$0^\circ \leq \phi < 180^\circ$	$180^\circ \leq \phi < 360^\circ$	
Isotropic	Isotropic	$5.1 \cdot 10^{-9} \pm 5.5\%$
$\vec{r}_d \cdot \vec{v}_s < 0$	$\vec{r}_d \cdot \vec{v}_s < 0$	$4.6 \cdot 10^{-9} \pm 5.6\%$
$\vec{r}_d \cdot \vec{v}_s > 0$	$\vec{r}_d \cdot \vec{v}_s > 0$	$5.6 \cdot 10^{-9} \pm 4.7\%$
$\vec{r}_d \cdot \vec{v}_s > 0$	$\vec{r}_d \cdot \vec{v}_s < 0$	$5.1 \cdot 10^{-9} \pm 3.8\%$

Table 4: Calculated detector efficiencies (counts per source neutron) for various orientations of the DD neutron generator. The efficiencies are for a toroidal source with major radius=2.55 m. The beam orientation is represented by $\vec{r}_d \cdot \vec{v}_s$, where $\vec{r}_d$ is the radial vector of the detector and $\vec{v}_s$ is the vector of the ion beam direction. ϕ is the toroidal angle relative to that of the detector.

Figure Captions

- Fig. 1** Diagram showing the ^{235}U fission detector/moderator assembly.
- Fig. 2** One-dimensional model of detector/moderator assembly used for the adjoint transport calculation.
- Fig. 3** Calculated relative efficiency of detector/moderator assembly as function of neutron energy.
- Fig. 4** Diagram showing layout of the in situ calibration experiment carried out on the TFTR in 1983. Arrows indicate direction of the ion beam in the DD neutron generator (see text for details).
- Fig. 5** Measured absolute efficiency of ^{235}U fission detector as function of the toroidal angle of the source relative to that of the detector. Note that detector efficiencies have been multiplied by 10^{-8} . Ion beam direction in the neutron generator (NG) is indicated by arrows: downward pointing for beam direction away from the detector, upward pointing for beam direction toward the detector. Also, see Fig. 4 for geometrical arrangement.
- Fig. 6** Example showing the three-dimensional MCNP model of the TFTR for the Monte Carlo calculations. (a) Vertical cut through bay C of half of the model showing major tokamak components. (b) Horizontal cut through upper shear compression panels of the model showing north-west half of TFTR. (c) Horizontal cut through the midplane of the model showing south-east half of TFTR. Note that test cell walls and roof are not shown.

Fig. 1

Fig. 2

Fig. 3

Fig. 4

Fig. 5

Fig. 6(a)

Fig. 6(b) & 6(c)

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