NUCLEAR ANALYSIS FOR ITER

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Abstract

This paper summarizes the main results of nuclear analysis calculations performed during the International Thermonuclear Experimental Reactor (ITER) Engineering Design Activity (EDA). Major efforts were devoted to fulfilling the General Design Requirements to minimize the nuclear heating rate in the superconducting magnets and ensuring that radiation conditions at the cryostat are suitable for hands-on-maintenance after reactor shut-down.

1. Introduction

Radiation transport calculations to estimate neutronics parameters are an important part of the ITER design process. The nuclear performance and shielding requirements for the in- and ex-vessel components and dose limits for hands-on activities to support remote maintenance operations for ITER are specified in the General Design Requirements Document (GDRD) [1]. In particular, these include

Nuclear heating rate in the toroidal field coils.	17 kW
Dose rate at the cryostat at two weeks after shutdown.	100 • Sv/h

The results reported here are for the Basic Performance Phase (BPP) of ITER operation (first-wall neutron fluence, 0.3 MWa/ m^2; average neutron wall loading, 1 MW/m^2). The BPP is envisioned to last for about ten years and involve a few thousand hours of DT operation. Blanket, vacuum vessel and other reactor component shield design are carried out in a logical progression. Initial results are obtained using one-dimensional scoping and parametric analyses and optimum shield compositions are recommended. In the BPP, the blanket and vacuum vessel are water-cooled, stainless steel assemblies having average inboard and outboard thicknesses of 1.0 and 1.25 m, respectively, with bulk shielding capabilities that provide adequate shielding to meet GDRD specifications. ITER, however, contains 60 major ports (vertical, equatorial including three neutral beam injection ports, and divertor), gaps between blanket modules, and other penetrations that degrade the blanket-vacuum vessel shielding performance.

Detailed two- and three-dimensional radiation transport calculations are then used to characterize radiation streaming and account for the geometric complexity of the tokamak including experimental, diagnostic, and plasma heating systems that penetrate the shielding. Three-dimensional models of ITER were developed for use with the Monte Carlo code MCNP [2]. Both 9° (half-sector) and 18° (full sector), models were widely used to characterize neutron and gamma-ray flux distributions, heating, radiation damage and other responses in in-vessel components (blanket, vacuum vessel, divertor, test-modules, diagnostic and heating systems) and outside the reactor (superconducting magnets and cryogenic systems). Estimates of the effects of radiation streaming through the major ports, gaps, channels and other small penetrations were carried out to provide data for determining shield disposition and composition to realize heating and shutdown dose rate limits. Multidimensional calculations were also performed to estimate ^{16}N-decay gamma ray heating in cryogenic components from activated water in the outlet coolant pipes.

Residual dose rate maps and nuclear heating rates in the toroidal field coils from streaming through the fully open Neutral Beam Injection ports were analyzed using 36° and 90° calculational models. A half torus (180°) model that included a tritium breeding blanket test module in one equatorial port was used to investigate the radiation environment inside the cryostat volume and to estimate port-to-port radiation streaming interference effects.

2. Nuclear-Heating in the Superconducting Magnet System

The integrated nuclear heating rate in the toroidal field coils and intercoil structure from radiation leaking through the reactor ports and from ^{16}N-decay gamma rays for 1.5 GW of fusion power is summarized in Table 1. Also given in the table are the calculational models used to obtain these data.

Table 1. Integrated Nuclear Heating Rates in the Toroidal Field Coils and Intercoil Structure

Toroidal Field Coil Parts and Locations		kW	Calculational Method
Inboard Toroidal Field Coil Legs		1.2	1- and 2-D
	Vertical Ports (20):		
Blanket Cooling Pipe Vertical Port (10)		1.3	3-D (9° Model)
Vertical Diagnostic Port (10)		0.2	3-D (9° Model)
Inter-Coil Structure		0.3	3-D (9° Model)
	Mid-Plane Ports (20):		
ICRF Ports (3)		<0.3	3-D (9° Model)
NBI Ports (3)		0.14	3-D (36° /90° Models)
ECH Ports (2)		<0.2	3-D (9° Model)
Remote Handling Ports (4)		<0.1	2-D (R-Z Model)
Test Blanket Ports (4)		0.05	
Diagnostic Ports (4)		0.07	3-D (9° /18° Models)
	Divertor Ports (20):		
Toroidal Field Coil		0.5	3-D (18° Model)
Inter-Coil Structure		0.4	3-D (18° Model)
	^{16}N Decay Gamma Rays*		
Upper Ports	with Guard Pipe	0.3	3-D (9° Model)
Upper Ports	without Guard Pipe	1.1	
Equatorial Ports		1.2	3-D (9° Model)
Divertor Ports		<0.1	3-D (9° Model)
Total Nuclear Heating		7.3	30% accuracy**

*)Includes heating in the poloidal field coil clamps, cryogenic lines, and break boxes.

**)Based on uncertainties in the reactor models and C/E ratios from integral experiments on ITER blanket-vacuum vessel mockups.

Other magnet responses based on a first wall neutron fluence of 3 MWa/m^2 specified for the lifetime performance of permanent components [1] (vacuum vessel, toroidal field coils, etc.), for example, the damage to the toroidal field coil copper stabilizer (1.2 x 10^{-5} dpa) and the total insulator dose (1.5 x 10^5 Gy), are below GDRD specifications. The total nuclear energy deposition in the poloidal field coils varies between 10 and 130 W and the total heating rate in all PF coils is <0.4 kW. The specific nuclear heating in the poloidal field coils is ~ 0.01 mW/cm^3. The estimated nuclear heating in the poloidal coils and coil clamps from ^{16}N-decay photons is <0.02 mW/cm^3.

3. Shutdown Dose Rates at the Cryostat

Residual dose rates at locations where hands-on maintenance is proposed are governed by activation of the outer steel layers of the vacuum vessel, intercoil structures and the cryostat. During maintenance periods (usually 10-14 days after reactor shutdown), the residual dose rates from the decay of ^{58}Co and followed by ^{60}Co from the Ni-component and Co-impurities in stainless steel are the main contributors to the. Table 2 summarizes the maximum dose rates at two weeks after shutdown at the end of the BPP at locations near the vertical, equatorial, and divertor ports where hands-on maintenance will be performed. Residual gamma ray dose rate distributions in cryogenic elements for different port configurations are characterized by uncertainties of factor of ~1.5 to 2 due mainly to the uncertainties in reactor operation scenarios.

Table 2.Shutdown Dose Rates at Maintenance Locations
(Cooling Time = 10^6 s, Neutron Fluence = 0.3 MWa/m^2)

Port and Location	Dose Rate (μSv/h)	Design Actions
		Vertical Ports
Cooling Pipe Ports:		
•Around the Port	50 - 120	If the port sidewall thickness is increased from 13 to 18 cm.
•Correction Coil Break Box	90	Local dose effect provided the intercoil structure adjacent to the port is modified.
Diagnostic Ports:		
•Upper port	240	Appropriate shielding will be recommended for each port.
•At the Port Walls	20 - 40	Diagnostics are still being designed.
		Equatorial Ports
ICRF Ports:		
•At the Closure Plate	40 - 70	
•At the Cryostat	~ 20	
NBI Ports:		NBI port wall thickness must be ~60 cm thick.
•TF Coil Break Box	150	Port wall thickness and composition was modified based on
•At the Cryostat	50-180	results given in. Ref. 3, Sec.7.2.2
ECH Ports:	60	Modified dogleg wave-guides and intermediate plug are proposed.
Remote Handling Ports.		
•Upper Port Wall Surface	~ 180	
•Outer Port Plug Surface	30 - 40	
•Cryostat/Bio-Shield Gap	20 - 40	
Test Blanket Ports	~ 100	The steel frame is sufficient to reduce the dose rate at the cryostat to ~100 μSv/h. Inclusion of test blanket modules assures dose rate will be acceptable.
Diagnostic Ports:		Appropriate shielding must be developed for each diagnostic port.
•LIDAR System	< 50	
•Integrated Diagnostic Port (#16)	< 50	
•Electron Cyclotron Emission Port (#19)	< 50	
		Divertor Ports
Pumping Ports:	20 - 30	Additional shielding around the regeneration pump
Remote Handling Ports:		
•At Cryostat	40 - 50	Modified port shielding structures added. See Ref. 3.
•At Suppression Tank Pipe	50	Resize suppression pipe thickness and length.

These values will be achieved if the design actions recommended from a neutronics point-of-view are implemented. The dose rates are generally low enough (<50-200 μSv/h) to allow controlled access for repair work activities.

Detailed 3-D Monte Carlo calculations were performed to determine the shielding for the virtually „open“ NBI ports using a 90° calculational model. Calculations showed that a wall thickness of 60 cm is needed to reduce the dose rate at the cryostat surface to levels between 50 and 180 μSv/h. However, in the space between the vacuum vessel and toroidal field coils, the NBI shield is thinner to avoid interference with the toroidal field coils. This leads to higher dose rates at nearby break boxes and at the cryostat. The divertor remote-handling port model did not initially include divertor diagnostics equipment with slots in the diagnostic cassettes and viewing system channels. Consequently, dose rates were too high and required insertion of shield plates in the duct and improved shielding around the suppression tank pipe.

4. Summary

Comprehensive neutron and photon calculation were performed using state-of-the-art radiation transport codes and cross-section data to assess the shielding performance of the reactor and structural assemblies to identify adverse radiation conditions and nuclear responses inside and outside cryostat. Analyses focused on global and local shield optimization/disposition, neutron streaming through major penetrations, operational neutron and gamma-flux distributions and energy release in reactor components, and radiation conditions in plasma heating and diagnostic systems. Detailed three-dimensional calculations were carried out to estimate radiation environments in and around the divertor including nuclear responses in the divertor itself. Estimates of the neutron and gamma ray flux, power and dose distributions were provided to designers to optimize shielding and reduce helium-production and radiation damage in stainless steel. Gas production at locations where rewelding is required for a fluence of 1 MWa/m^2 is generally acceptable (<1 appm) at most locations but marginal (~3 appm) where radiation streaming effects dominate. The main neutronics responses are given in Table 3.

Table 3 Calculated Nuclear Responses

Total Power	~ 2100 MW
Nuclear Power	~ 1800 MW
Average DT-neutron Wall Loading	~ 1 MW/m^2
TFC Nuclear Heating	~ 7.3 kW
Dose Rate at the Cryostat 14 days after shut down at end the of the BPP	~100 - 200 μSv/h

The estimated total nuclear heating in the toroidal field coils and intercoil structure is ~7.3 kW, a factor of two lower than the GDRD value which gives a reasonable safety margin. The dose rates at two weeks after shutdown allow controlled personnel access at locations where hands-on maintenance is required. Some port configurations, particularly those being used for diagnostics may, however, require additional study. Steel-water ratios in the vacuum vessel and port walls were assessed for cost and shielding efficiency. With the exception of the NBI and some diagnostic ports, port shielding is adequate for reducing the nuclear heating in the superconducting coils and, simultaneously, reducing the dose rates at the cryostat to levels that permit personnel access.

Material optimization studies showed that stainless steel with high boron content (2 wt% in non-structural shielding elements) reduces the nuclear heating in the toroidal magnets by as much as a factor of two. Co, B, and Nb levels in stainless steel were recommended for reducing He production and long term activity. Borating the biological shield concrete (~ 0.1 g/cm^3 B) reduces the dose rates outside the cryostat by factors of 2-3 which is important for maintenance considerations.

Radiation transport methods, nuclear data and calculational models combined with 14.1-MeV benchmark measurements and supporting analyses established the reliability of design calculations.

References

[1] [1] Technical Basis for the ITER Final Design Report, Cost Review and Safety Analysis, FDR0, IAEA/ITER EDA/Documentation Series (1998, to be published).

[2] J. Briesmeister (Editor), „MCNP-4A Monte Carlo N-Particle Transport Code System“, CCC-200 revised, February 1994; Version „MCNP4B“, CCC-660, Radiation Shielding Information Center, Oak Ridge National Laboratory, (April 97).