Proposal dinklage_012 ***submitted*** *last changed on:* *April 25, 2022*

Induced plasma termination of W7-X with massive tungsten injections

Proponent

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Classification

EUROfusion work package:	WPW7X
Topic(s):	• Exploration of scenarios compatible with carbon-free operation and tungsten PFCs
Deliverable(s):	• Characterize enrichment/accumulation for low-Z and high-Z impurities • Definition of the operation limits associated with plasma-facing components containing tungsten materials

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Physics Description

Scientific background and objectives

Thermal quenches due to large amounts of material interacting with plasmas may occur in stellarators as they do in tokamaks (typically followed by current quenches not observed in stellarators). For transient loss events, overcoming a critical heat impact (in terms of power density and squareroot of impact time) leads to roughening, cracking, melting or even erruptive evaporation. The resulting risk for disintegration of plasma facing components has an impact on the device operation and could not only affect operation but also critically affect maintenance cycles. While these aspects are relevant to W7-X and the envisaged metallic wall operation, the proposed study also provides a basis for safety aspects of any reactor scale device. The specific purpose of the proposal is to assess the duration of plasma collapse and the wetted area of loads from a thermal quench in W7-X. For a quantitative study, it is proposed to inject known amounts of tungsten impurites by TESPEL and LBO and to identify the allowable amount of material maintaining plasma operation. Moreover, the specific mechanism of energy loss (radiative cooling and the role of cold fronts) is proposed to be studied with camera systems, fast bolometry and highly- sampled ECE and soft-X ray diagnostics, respectively. The proposal is addressing an objective of the EUROfusion Workpackage (WPW7X - Preparation of Metallic Wall Operation) and will be conducted as a joint experiment within the IEA Stellarator-Heliotron TCP. Experiments on LHD have been conducted.

Relevance to W7-X program

The expected findings define safety margins for the envisaged preparation of all metallic-wall operation with tungsten plasma facing components. The impact time and a documentation of the wetted area will allow to determine a minimum critical time and wetted area for a termination event in W7-X. Previous experiment on LHD revealed an amount of N=3.5E17 W atoms released in the plasma core to terminate a plasma. Complementing these experiments on LHD, the decay mechanism (identified in LHD to be a core radiation loss on fractions of the energy confinement time with inward propagating cold fronts) potential mitigation techniques (e.g. addtional heating during transient events) become possible to assess.
The proposal directly contributes to the preparation of metallic wall operation, operation limits and safe operation of W7-X. The proposal delivers quantitative information relevant to long-pulse operation and metallic wall operation. The derived infromation is also relevant to the assessment of the reactor potential of the HELIAS line and might be of use for the ITPA acitivity

Approach and Methodology

induce a termination of a stationary plasma (like 5s).
10 shots 5e19, 6MW:
N = 5, 0.5, 1, 2, 3, 4 e17 W TESPEL + contingency
10 shots 5e19, 6MW:
N =5, 0.5, 1, 2, 3, 4 e17 W LBO + contingency
For the injections of smaller amounts, both LBO and TESPEL can be injected in one program
in case the contingency is not used, programs at different density (7e19) can be repeated
Golas of the experiments:
- determine the critical N for induced plasm termination
- document the spatial distribution of radiation losses
- investigate the plasma decay w/ highly sampled ECE and soft-X measruments (to idetify the propogation of cold fronts)
- document the impurity transport also with PHA and bolomtry

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Experiment configuration

special diagnostics requirements:	fast sampled ECE, soft-X			
magnetic field				
main magnetic field configuration:	EJM000+2519 (released)			
comment:				
Beyond error field correction				
trim coil operation:	Y			
control coil operation :	no preference			
plama densitiy				
plasma density range [m^-3] :	**min:** 4e19		**max:**6e19	
divertor state:	attached			
gases and fueling				
gas fueling:	Y	H2		
gas fueling system:	main gas inlet system			
pellet fueling:	N			
seeding:	N			
diagnostic use:	Y	Ar		
plasma heating				
ECRH heating:	X2	**min [MW] :**3	**max [MW] :** 8	
ECRH off-axis heating:	N			
ECRH current drive:	N			
NBI	N	**min [MW] :**	**max [MW] :**	
NBI diagnostics blibs allowed :	Y			
ICRH :	N			
Number of dedicated programs:	20			
Number of preparatory programs :				

Remarks:

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