Superconducting Tokamak T-15 Upgrade

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Abstract. The plans of upgrading the superconducting tokamak T-15 and the program of physics and technology research for the period from 2008 to 2022 are outlined. The technical modernization of the T-15 is aimed at formation of the elongated divertor configuration, increase of the plasma discharge duration up to 1000 s and the total heating power to 20 MW. Upgrade will allow fusion oriented research on the T-15 supporting the ITER and DEMO projects.

1. Introduction

Experimental magnetic fusion devices of the next generation as well as industrial thermonuclear plants will operate in quasi-stationary regime. This requires superconducting magnetic systems. Development of the basic technological systems of a tokamak-reactor such as cryogenics, plasma heating and current drive, plasma shape and position control, divertor is needed in steady state scenarios.

Up to now only a few superconducting fusion devices (T-7, T-15, Tore Supra, TRIAM1-M, LHD, HT-7) have been in operation. Design, construction and operation of these devices gave substantial experience of using the cryogenic techniques and technology for quasi-stationary regime maintenance, which determined the ITER technological solutions [1]. The next generation of the superconducting tokamaks (EAST, KSTAR, SST-1, JT60-SC) and upgraded T-15 will contribute to R&D reactor oriented fusion programs.

Technical upgrade of T-15 is aimed at production of 10 keV plasma, formation of the poloidal divertor configuration, increase of the plasma discharge duration up to 1000 s and the total heating power to 20 MW.

Modernization of the tokamak T-15 with the purpose of creation of the ITER-like magnetic configuration is considered as a step to increasing participation of RF in the experimental collaborative investigations on the tokamaks.

2. Status of the T-15 upgrade

The T-15 tokamak (Fig.1), major radius 2.40 m and minor radius 0.78 m, was in operation from 1988 to 1995 and then it was shut down. The experiments were performed with circular limiter configuration and had demonstrated reliable operation of the superconducting toroidal magnetic system from Nb_3Sn. Regimes with 1 MA current and heating power of MW level and 1 second pulse duration had been obtained (see Table 1) [2].

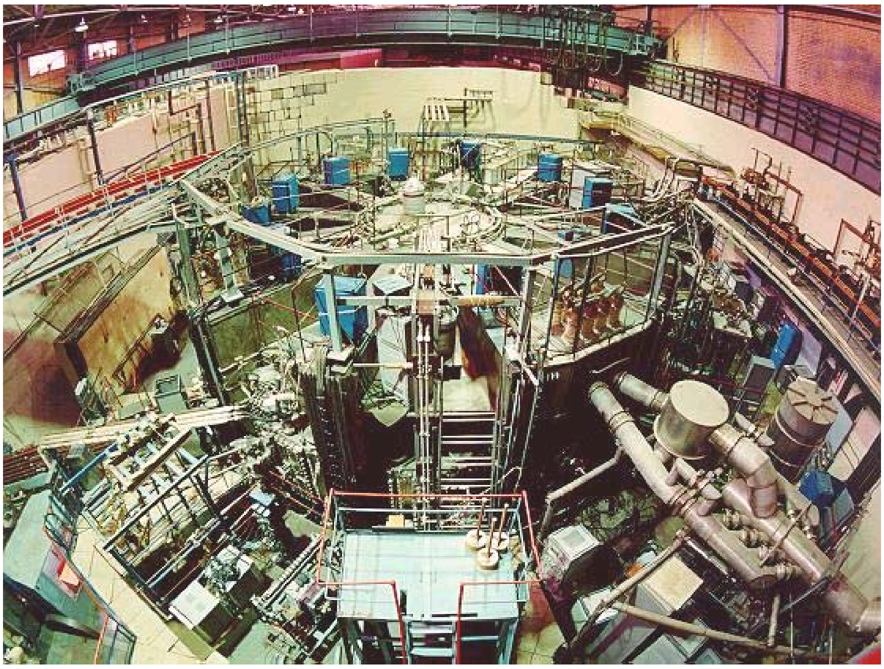

FIG. 1. Tokamak T-15.

TABLE 1. DESIGN AND ACHIEVED PARAMETERS OF THE TOKAMAK T-15 (LIMITER CONFIGURATION).

Parameter	**Design**	**Achieved**
Magnetic field, T	3.5	3.6
Plasma current, MA	1.4	1
Pulse duration, s	5	1.5
NBI, MW	6	0.6
ECRH, MW	5	1.5

The design parameters of the tokamak T-15 upgrade with divertor magnetic configuration are presented in Table 2. The plasma elongation and triangularity are lower than those in ITER, while the aspect ratio is higher.

TABLE 2. DESIGNED PARAMETERS OF THE TOKAMAK T-15 (DIVERTOR CONFIGURATION).

Parameter	**Value**
Magnetic field, T	3.5
Plasma current, MA	1
Pulse duration, s	5(25)(1000)
Minor radius, m	0.42
Major radius, m	2.43
Aspect ratio	5.8
Elongation	1.47
Triangularity (average)	0.25
Divertor	Single-null
NBI, MW	9
ECRH, MW	7
LCRH, MW	4

Main technical objectives of the T-15 upgrade are as follows:

- substantial modernization of all technological systems with increase of their reliability;
- creation of the elongated separatrix magnetic configuration in the existing vacuum chamber;
- creation of equilibrium control system for elongated plasma;
- design, manufacturing and installation of the divertor in the existing vacuum chamber;
- modernization of the heating and current drive systems, increase of the heating power up to about 20 MW and pulse duration to 1000 s;
- design and manufacturing of the system of integrated control of stability, equilibrium, heating and confinement of high temperature plasma in real time using feedback control.

Physical program of T-15 will be oriented on studies of:

- quasi-stationary plasma discharge with high level of plasma heating and current drive;
- real-time control of MHD activity and current and pressure profiles;
- problems of divertor and plasma periphery;
- plasma interaction with different materials including graphite, tungsten and lithium;
- technology of the first wall maintenance;
- energy and particle transport and turbulence.

The experimental investigations on the tokamak T-15 upgrade will provide educational opportunities for ITER and later for DEMO. Testing the diagnostic systems manufactured for the tokamak ITER will be made.

3. Milestones of T-15 tokamak upgrade and experimental program

Three stages of the project are considered in the period from 2008 to 2022.

At the first stage (2008-2012) we plan to re-equip the engineering and diagnostic systems, to perform integrated tests of the facility with circular limiter configuration, and to install new in-vessel elements of the divertor, equilibrium and control systems. Plasma heating will be provided by neutral beam injectors (NBI) and gyrotrons (ECRH). Auxiliary heating power will be 11 MW with pulse duration 5 s (NBI 9 MW, ECRH 2 MW) at this stage (see Table 3).

TABLE 3. DESIGN PARAMETERS OF THE HEATING SYSTEMS.

Method of heating	**2008-2012**	**2013-2017**	**2018-2022**
NBI, MW	9	9	9
ECRH, MW	7	7	7
LHRH, MW		4	4
Duration, s.	5	25	1000

Experiments with elongated configurations and divertor at pulse duration up to 25 seconds will start at the next stage (2013-2017). Heating power will be increased up to 20 MW. This will allow us to operate the tokamak in basic regimes.

Modifications of the magnetic, heating and current drive systems essential for operation at 1000 s pulse duration are planned from 2018 to 2022 simultaneously with experimental program. Superconducting coils of poloidal field and thermal shield elements will be designed and installed within this period.

The existing system of poloidal magnetic fields will be modified to provide the plasma elongation K ~ 1.5 and triangularity δ ~ 0.3.

The additional poloidal field coils will be installed inside the vacuum chamber to update the PC-system. An example of new magnetic T-15 configuration provided by such modification is shown in Fig.2. Three coils (shown by squares) at the bottom of the chamber form the divertor zone. The coil at the top will allow the triangularity control.

Passive copper plates and active correction coils will be installed in the vacuum chamber at the outboard side of torus for VDE-stabilization of the plasma column and for active MHD experiments.

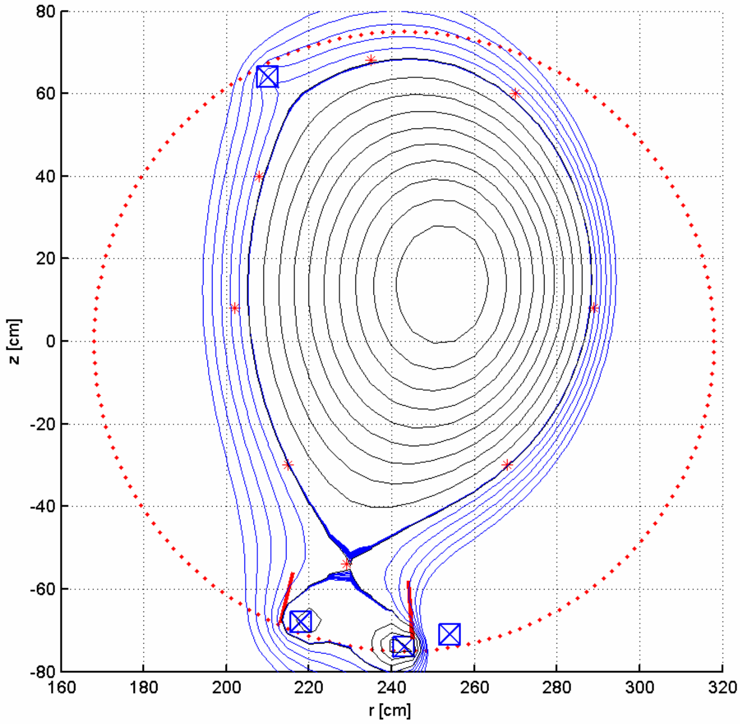

FIG. 2. Elongated magnetic configuration of the T-15.

The divertor consists of graphite target plates and coating tiles protecting the wall. The divertor plates are designed to receive the heat power density up to 20 MW/m^2 without loss of efficiency. To operate at such heat loads the water cooling system of the divertor plates will be used.

At present the conceptual design of the divertor and the poloidal magnetic field coils have been started. The design and manufacturing will be completed in 3 years. The magnetic configuration, divertor coil parameters and the target plate geometry will be optimized to provide flexibility of plasma shaping and divertor flow control. Also the specifications of divertor regimes (attached/detached) will be defined: divertor layout, X-point position, divertor leg length, pumping slots geometry, pumping speed, cryopanels allocation. Simulations of the SOL and divertor regions will be produced using two-dimensional divertor code B2SOLPS [3]. Preliminary estimations show that the detachment control is possible at the pumping speed higher than 100 m^3/s and location of the cryogenic pumps close to the divertor volume.

Analysis of the equilibrium divertor configurations with warm poloidal field coils in the tokamak T-15 shows that the maximal discharge duration at the plasma current 1 MA cannot exceed 25 s. Therefore, maintenance of the equilibrium divertor configuration during 1000 s requires installation of the superconducting poloidal field coils and the water cooled thermal shield.

The evolution of heating systems during the planned period is presented in Table 3. Combination of Neutral beams, electron cyclotron and low hybrid waves injection with total power up to 20 MW will be used for plasma heating and current drive. The heating complex will consist of 3x3MW NBI, 7x1MW gyrotrons and 2x2MW klystrons developed for long pulse duration: 5, 25, 1000 seconds.

4. Basic scenarios of the tokamak T-15 upgrade and MHD stability

4.1. Basic scenarios

Calculations of basic plasma scenarios and parameters in the tokamak T-15 were performed using the transport code ASTRA [4]. H-mode confinement scaling was used for plasma transport description. Three different scenarios described below reflect opportunities of T-15 operation in high performance modes: high plasma density (regime I); low density with fully non-inductive current (regime II); high β_N with fully non-inductive current (regime III). The efficiency of the current drive by ECR waves was evaluated by the code OGRAY [5]. The heating power was assumed to be equal 16 MW for all cases (9MW NBI and 7 MW ECR). The results of simulation are shown in Table 4 and Figures 3-5.

TABLE 4. CALCULATED PLASMA PARAMETERS FOR THREE DIFFERENT REGIMES OF THE T-15.

Regime	**I**	**II**	**III**
I_p, MA	1	1	0.5
B_t, T	3.15	3.15	2.63
n, 10^{19} m^{-3}	14.4 (0.8 n_{Gr})	3.56 (0.2 n_{Gr})	7.2 (0.8 n_{Gr})
β_T	1.82	1.35	1.58
β_N	2.41	1.78	3.49
β_p	1.17	0.87	2.83
l_i	0.76	0.65	0.65
I_{OH} (%)	75.3	0.0	0.0
I_{bs} (%)	14.7	20.4	44.9
I_{CD} (%)	10.0	79.6	55.1
$(I_{bs}+ I_{CD})/ I_{\Sigma}$	0.147	1.0	1.0

The plasma parameters (density, temperature, beta, non-inductive currents) were simulated near standard operational limits.

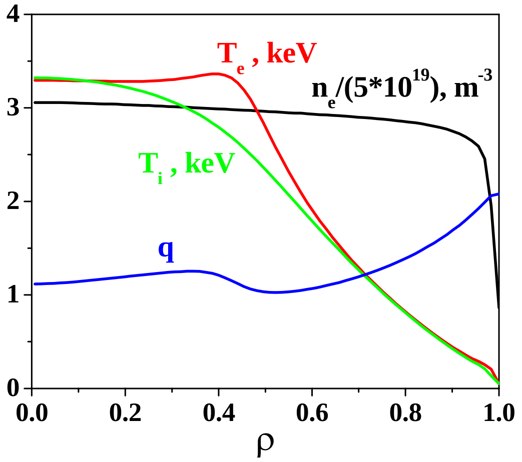

FIG. 3. Radial profiles of the density, electron and ion temperature and safety factor in regime I.

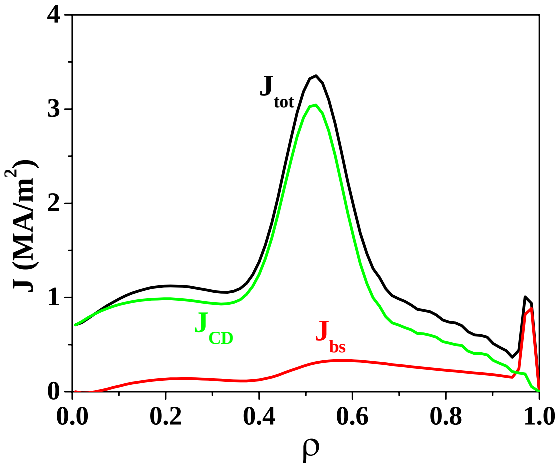

FIG. 4. Radial profiles of the total current (J_{tot}), non-inductive current (J_{CD}) and bootstrap current (J_{bs}) in regime II.

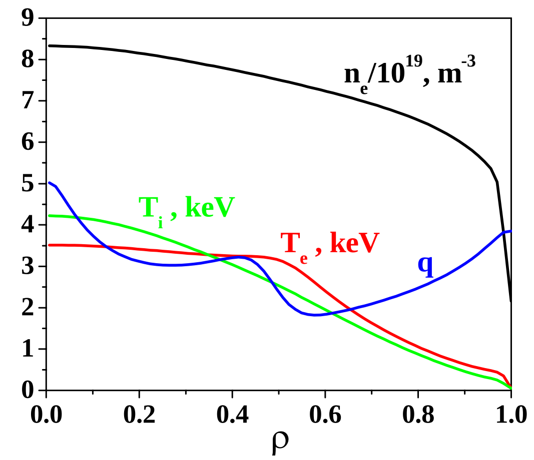

FIG. 5. Radial profiles of the density, electron and ion temperature and safety factor in regime III.

1. In high density regime (Fig. 3) with plasma current 1 MA, the density n_e=1.44x10^{20}M^{-3} (0.8 of the Greenwald density n_{Gr}) and the heating power 16 MW it is possible to reach core plasma temperatures T_e, T_i > 3 keV. Normalized beta is rather high (2.4) in this regime and it is close to that in the ITER inductive scenario. Reducing the plasma density to n_e=0.5x10^{20}M^{-3} results in the temperature rise above 6 keV.

2. Availability of fully non-inductive regimes is significant for long pulse operation of T-15. The radial profiles of currents in the non-inductive scenario are shown in Fig. 4. The nominal 1 MA total current can be maintained using 9 MW NBICD and 7 MW ECCD for the density 3.6x10^{19} m^{-3}. The fraction of bootstrap current amounts to f_b=0.2 in this regime.

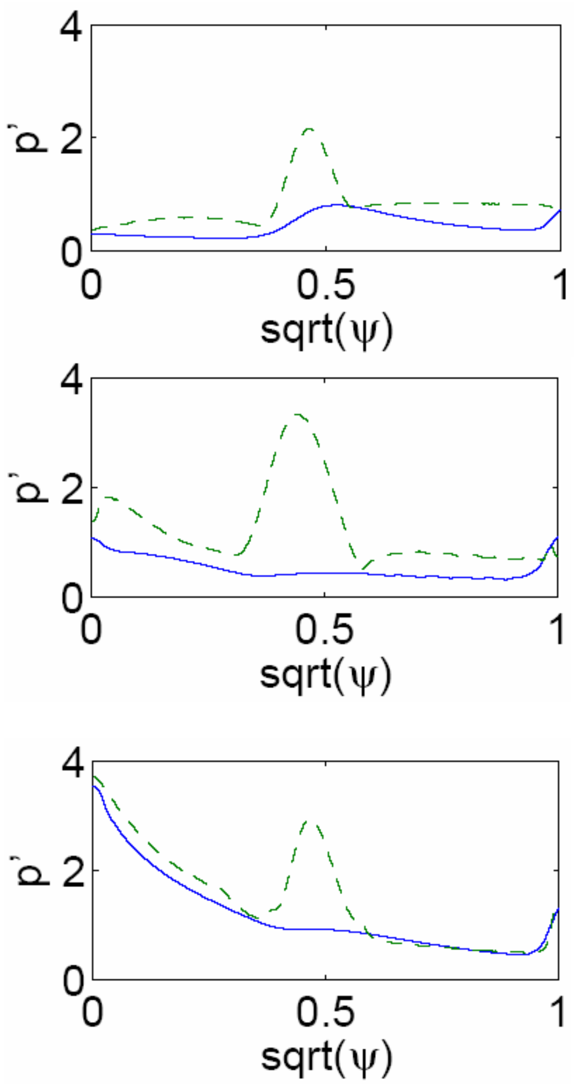

FIG. 6. Radial profiles of the pressure gradient for regimes I, II and III.

Non-inductive currents I_{cd} generated by ECCD and NBI make possible the control of plasma current profiles in a wide range. In particular, the profiles with reversed shear can be created and the safety factor q_{min} can be varied in a wide range (q_{min}=1-2).

3. Figure 5 demonstrates possibilities of plasma operation with fully non-inductive current (0.5 MA) at high plasma density (0.8 n_G) and high normalized beta β_N = 3.46. Such regime where β_N exceeds the ideal no-wall limit will be used for stability studies. In order to sustain high-β steady state operation in such regimes we plan to use ECCD for neoclassical tearing mode control and in-vessel coils for suppression of resistive wall modes.

4.2. MHD stability

For regimes I, II and III the MHD plasma stability with respect to no-wall external kink, ballooning and Mercier instabilities was simulated by the code KINX [6]. This code allows investigation of the plasma stability in axially symmetric systems (including configurations with a separatrix) within the linear ideal magnetohydrodynamics. Results of the simulations are shown in Figures 6 and 7.

Figure 6 presents the radial profiles of pressure gradient p' for three regimes. Expected profiles of the plasma pressure are marked by blue solid lines. Green dashed lines denote ballooning limits of the pressure gradients. The plasma pressure gradient for regimes I and II are below the ballooning limits. In regime III with high β_N the pressure profile is close to marginal stability of the ballooning modes.

Regime I is Mercier unstable because of high pressure gradient p' near q_{min} when $q_{min} = 1.02$. In this case the m = 1 and m = 2 harmonics are coupled with the n = 1 external mode. Our calculations show that $q_{min} > 1.04$ is needed for stability in this regime.

Regime II is characterized by very large bootstrap current in the pedestal region that drives n=5 ideal mode. Strong type I ELMs extending up to q_{min} location may be expected.

Regime III is strongly unstable against n = 1 external mode. Figure 7 shows the normal displacement $\xi\nabla\psi$ of this mode versus square root of the poloidal magnetic flux. The mode n = 1 couples to m = 2 and 3 "infernal" harmonics. Therefore excitation of the external kink modes could be expected and experiments on the stability control and RWM suppression can be carried out.

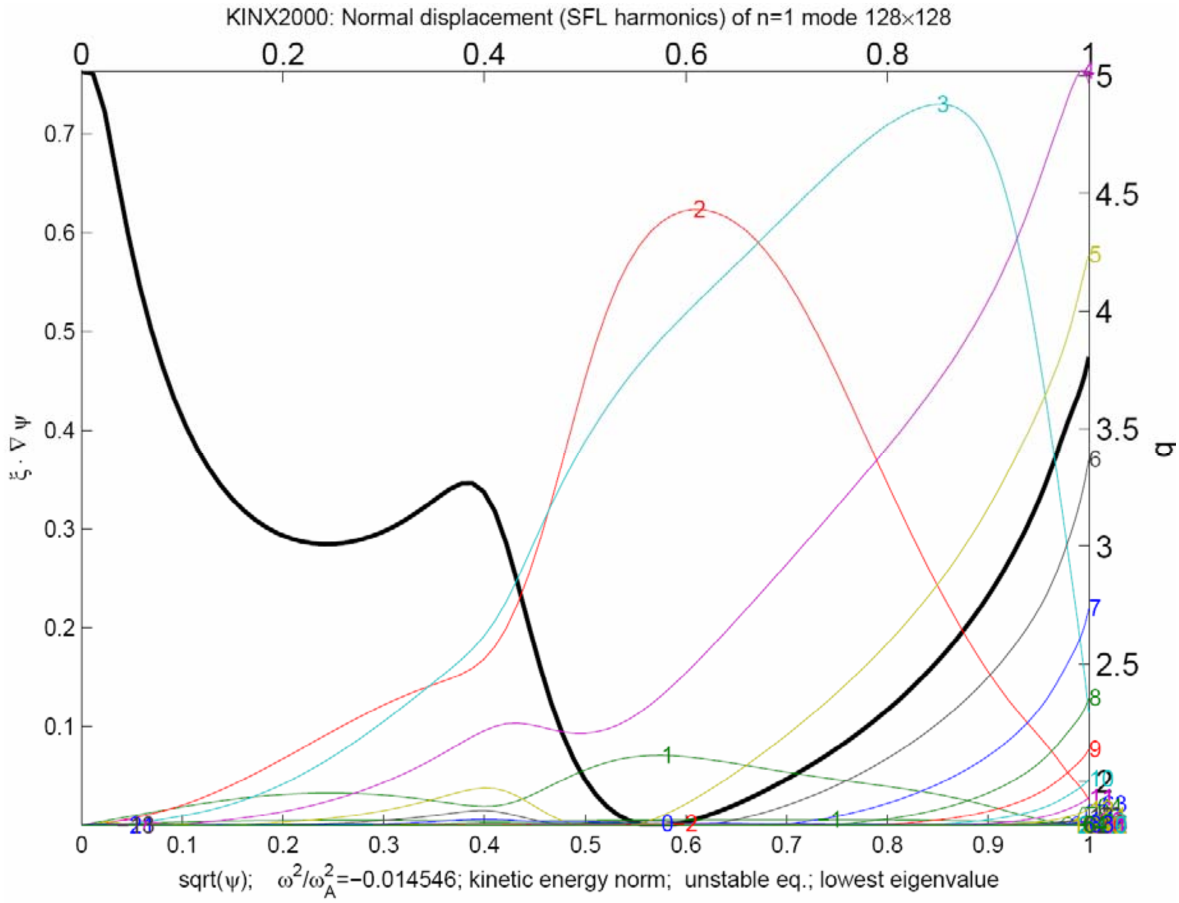

FIG. 7. Radial profiles of the normal displacement $\xi\nabla\psi$ of the n=1 mode (color lines) and the safety factor q (black line) calculated for regime III.

5. Conclusions

The upgrade of T-15 and directions of the experimental program for period of next 15 years are proposed. The successfully operated toroidal magnetic coils, vacuum vessel, basics of major technological systems will be kept as the basis of the device. The upgrade is aimed at creation of the device with noncircular cross section, divertor, plasma parameters interesting for fusion program and opportunities of long pulse operation.

Modification of the T-15 tokamak will allow operation with large aspect ratio plasma with minor radius ~0.5 m. This tokamak will extend operational domain of ITER complementary machines. Determining optimal parameters of future reactors requires.

The national fusion research center will be established in RRC Kurchatov Institute on the basis of Tokamak T-15 upgrade. It will integrate activity on tokamak research and magnetic fusion technology development in Russia and support staff training programs for ITER and DEMO.

Acknowledgements

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