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SEVENTH MEETING OF THE ITER PHYSICS EXPERT GROUP ON ENERGETIC PARTICLES, HEATING AND STEADY STATE OPERATIONS

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The seventh meeting of the ITER Physics Group on Energetic Particles, Heating and Steady State Operation was held at CEN/Cadarache from 14 to 18 September 1999. It was the first one following the redefinition of the Expert Group structure at the ITER Physics Committee in October 1998. It was also the first one without participation of US physicists. The meeting was well attended, with a large participation from all the ITER EDA Parties. The success of the meeting was due to the strong involvement of the Co-Chair, Professor Miyamoto, to the help of the Central Team and of Dr Rimini (JET) as well as of Dr Nguyen (CEN/Cadarache) regarding the local organization. The objectives of the Meeting were:

1. Review and implications for ITER of the results achieved since last meeting, including APS, EPS conferences on: fast particles confinement and fast particles excited modes; heating and current drive system (NBI, ICRH, LHCD, ECRH); and steady state aspects.

2. Re-evaluation of the physics of energetic particles, heating and steady state operation in RTO/RC ITER such as central current drive scenarios, heating and current drive efficiency of high energy negative NBI, MHD effects of energetic particles in reversed shear configuration, RF stabilization of the neoclassical tearing mode (NTM), toroidal and poloidal rotation induced by ICRH.

3. Consequences of new objectives/design of RTO/RC ITER including high Q operation, steady state and fast particle behaviour on proposed heating systems.

The Meeting started with a review by the Chairman of the Changes to the Expert Group and of the Urgent Physics Research Areas as defined by the ITER Physics Committee. Then the new design and physics issues of Reduced Technical Objectives/Reduced Cost (RTO/RC) ITER were reviewed. For the most part of the first two days of the Meeting, a series of presentations were made. Subsequently, the participants split into smaller groups aiming at summarizing the discussions and identifying priority actions. The topics of these groups were: Energetic Particles, Ion Cyclotron Resonance Heating (ICRH), Lower Hybrid Current Drive (LHCD) including stabilization of neoclassical tearing modes, Electron Cyclotron Resonance Heating and Current Drive, Neutral Beam Injection (NBI) and Steady State Aspects. The findings of these groups, briefly summarized below, were presented to all participants on the third and final day of the Meeting, with a broad consensus on the priority actions.

Energetic Particles

The conclusions from modelling studies of Alfvén eigenmodes were sometimes somewhat contradictory. A high $q_0 = 4.5$ reversed shear configuration is strongly unstable to low frequency drift KAE, suggesting that operation at q_0 closer to $q = 2$ will be safer. It was also found that when q_{min} is slightly below an integral, for instance $2.5 < q_{min} < 3$, a slight change of q_{min} causes a considerable change of AE frequency and stability. Negative shear configurations are not necessarily more stable than positive shear. Other analysis indicates that high n TAE modes are less dangerous in reversed shear than in monotonic shear. For monotonic shear, it has been found that TAEs are more stable in the LAM proposal than in the IAM proposal for a given Q, but at Q = 10 destabilizing effects are still not very important.

New results on TAEs modes driven by the Negative Ion Beam system on JT-60U (360keV, 4MW) were presented with various b_{fast} (0.1 to 0.6%), various velocities (v_b/v_a = 0.4–0.9), various current profiles, including reversed shear configurations. The results agree qualitatively very well with the results of the Nova-K code. Comparison between the Penn code and JET experiments indicates the stabilizing effects of high edge magnetic shear and of a weak central shear.

Calculations have indicated that alpha ripple losses of RTO/RC ITER are of the same level as in FDR-ITER. In particular, up to 10% of alpha power can be lost in IAM for a reversed shear configuration. They are reduced to less than 1% if the ripple amplitude can be reduced by <1.5. It is to be noted that NBI ripple losses are small both for DT and for non-active operation.

Experimental data from JFT2-M have shown that TF ripple can be reduced from 2.2% to 1.1% by Ferritic Steel insertion. The corresponding surface ripple losses of NBI are reduced from 10% to 5%. These encouraging results indicate that ripple losses will be acceptably low in a 13MA positive shear pulse operation. A ferritic steel insert could significantly reduce the 16% losses predicted for a 9MA steady state RS configuration.

Ion Cyclotron Resonance Heating

The design of the antenna has been re-assessed, leading to higher voltage handling capability (42kV for 20MW coupled power). Variations of the Resonant Double Loop antenna are being investigated. The benefit of a better coupling is compensated by the more stringent technical requirements.

A comparison of ELMs between NBI and ICRH (JET) has indicated that for an identical gas flow and with a power well above the L to H power threshold, type I ELMs were obtained with a similar pedestal height. This is in agreement with Asdex results although sawteeth activity is very different for the two heating methods. Mode conversion H-^{3}He and minority ^{3}He, scenarios relevant for the non-active phase of ITER, have been successfully tested on Asdex. Alfvénic activity driven by ICRH has been stabilised by 300keV NBI on JT-60U. Internal Transport Barriers with similar ICRH and NBI heating have been obtained in JET (similar electron and ion temperatures).

Bulk ion heating scenarios are available at $2f_{cr}$, especially with the addition of ^{3}He minority. An efficient current drive efficiency is obtained at higher frequency (60MHz). Off-axis mode conversion current drive might be sufficient for neo-classical tearing mode stabilization. "Heavy minority" heating scenarios are also available for the non-active ITER phase.

Lower Hybrid Current Drive

The main design concepts produced for ITER EDA can be kept for RTO/RC ITER, with the advantage that coupling and thermal load of the LHCD launcher will be improved.

Stabilization and control of NTM with LHCD have been demonstrated on Compass-D, leading to operation at high beta values.

Advanced scenario discharges with ITBs with full current drive were produced in JT-60U with LHCD, and steady discharges with b_N in the range 1.5–1.7 have been achieved.

Electron Cyclotron Resonance Heating and Current Drive

110GHz gyrotron (1MW/2s, 0.3MW/5s) with diamond window (steady-state temperature reached) and depressed collector has been operated in JT60-U. Commissioning of a 118GHz gyrotron (~ 0.4MW/~ 16s) with a record energy of 6.2MJ has just been done at CEN/Cadarache.

The latest generation of experiments with poloidal steering and control of deposition/CD profiles has started. First results of off-axis CD efficiency (TCV, DIII-D) have been reported, which somewhat exceeds predictions of existing codes. Full current drive has been achieved on TCV for 1.9s (1.5MW/125kA) at $1 \times 10^{19} m^{-3}$. Promising NTM stabilization experiments in ASDEX-U (d.c. scheme) have been reported. Very notably, transport barriers with $T_e > T_i$ using counter-ECCD have been observed in Asdex U.

Extensive calculations confirm performance in RTO/RC ITER comparable to that achievable in ITER-FDR. Initial calculations for the non-active phase have been carried out. The potential performance in this case may be limited by $3w_e$ damping. The performance with poloidal steering is investigated. It is important to incorporate real launcher geometry.

Detailed calculations required for NTM stabilisation in ITER have been carried out. Results suggest that a power lower than 30MW is sufficient but there are still some uncertainty in the exact requirements.

Experiment and theory suggest that d.c. schemes are sufficient but an accurate control of the location is required. NTM avoidance by profile control (ECRH) is also demonstrated (COMPASS-D, TCV). 'Top' launch in ITER offers certain advantages for NTM stabilisation schemes, eg. flux expansion, deposition localisation, reduced sensitivity to launch angles etc.

Neutral Beam Injection

The NBI energy of 1MeV is considered to be a good optimum for RTO/RC ITER and gives good beam penetration and current drive efficiency without excessive shinethrough at lower densities. A large beam tangency radius is desirable for high current drive efficiency. In practice this is limited by engineering constraints and this is considered satisfactory in present IAM design. The vertical beam power footprint, both the power profile and the vertical displacement from the plasma magnetic axis, has a strong influence on the heating and current drive in the plasma core. The optimization of this profile is highly sensitive to plasma transport and MHD assumptions and further simulation work is required in this area. The choice of vertical beam power footprint influences the requirements for port access. It may, however, be difficult to rigidly define the optimum vertical power profile for all modes of operation and some capability to vary the shape of the power footprint could be incorporated in the design.

Steady-State Aspects

High Performance plasmas with Internal Transport Barriers (ITBs) were sustained in Optimized Shear scenarios for as long as the additional heating power can be applied (JET). The key technique was to utilize Argon puff to modify edge current to optimize plasma current profile against pressure profile: $b_N = 2.5$ together with $H_{89} = 3$ were maintained for 4 s. This supports JT-60U results. High performance ELMy plasmas with $H_{89x}b_N = 9$ were maintained for the maximum duration time of the NBI: $16t_E$ (2s) (DIII-D). Probably ITBs were formed. ECCD can drive all the plasma current, and even more in quasi steady-state conditions (TCV). Efficiencies are consistent with predictions. Counter-ECCD has prevented central q to fall below q = 2 (ASDEX-U). As a result, good core confinement was maintained with $T_e > T_i$. It was shown that the b limit can be increased by increasing triangularity. High performance ELMy Reversed Shear plasmas were maintained in full current drive conditions: $f_{BS} = 75\%$ and $f_{NBCD} = 25\%$ (tentative evaluation), $I_p = 0.8$MA, $B_t = 3.4$T, $b_N = 2$, H = 3.6, $P_{NB} \sim 5$MW (JT-60U). It has also been attempted to raise sustainable b_N in LHCD reversed shear discharges. So far, b_N values in the range of $b_N = 1.5$–1.7 with have been achieved.

After a wide discussion on the findings of the groups, a broad consensus was reached on the following research priorities:

- Results of ferromagnetic insertion in JFT-2M experiments should be examined in more detail. Realistic ferromagnetic insertion (allowing port existence) should be studied.

- Extend understanding of AE damping and drive by comparisons of fluid and gyrokinetic models with experiments. Improvement of predictive capability of theory is needed, such as the drift kinetic Alfvén eigenmode. Transport of energetic particles related to AE shall be studied.

- More comparisons between ICRF and NBI in ELMy H-modes shall be made (compare edge pedestal core confinement). Modelling and experimental simulation of ICRH scenarios for the non-active phase, in particular at reduced B_T, shall be studied.

- Experiments and modelling for stabilization of NTM's using Mode Conversion CD. Destabilization of sawteeth in discharges with high fast-particle content and NTM stabilisation with minority ICRH CD.

- More assessment of ICRH toroidal rotation is needed. Extrapolation of present heating results mainly rests on positive NBI database with high ratio of i/e heating, significant fuelling and rotation drive. More data with ICRH and ECRH should be included in the database.

- Study tools to improve the coupling capability of LHCD in relevant scenarios: gas feed, edge ionization, gap control. Active control should be a goal. The LHCD antenna design has to be completed in order to take into account the port constraints in RTO/RC ITER and the present Passive-Active Multijunction concept has to be validated on present tokamaks.

- The effect of ion versus electron heating in ITB formation shall be studied. Investigations of edge conditions in advanced scenarios with ITBs in order to assess LHCD and ICRH coupling properties of such plasma.

- Investigate NTM stabilization with LHCD on present tokamaks with ITER relevant plasma scenarios. Compare ECCD and LHCD NTM stabilization. Further evaluation of requirements for NTM stabilization current drive efficiency at rational surfaces JCD(r) required for stabilization, supported by experimental data where possible and further analysis of stabilization terms. 'Robust' schemes are required.

- Further evaluation of poloidal vs. toroidal steering (2D steering very difficult) for both outside and top ECRH launch taking into account engineering constraints and operating flexibility is urgently required. Bearing in mind the strong desire to utilize a single frequency for all applications, an urgent assessment of the optimum frequency is required.

- Current Profile Control issues include: b limit, alignment of j_{driven} (p) with jb_N(p) with $fb_N > 50\%$ in steady-state, control of ITB controllable with j(p) and demonstration with real time control of the influence of current profile on confinement.

- Integration of issues in a consistent steady state scenario (Integrated Scenario), including: the consistency of the CD/Heating systems from the ITB formation phase to the steady-state phase, the importance of rotation control and the amount of spare power needed to suppress MHD modes. The robustness of ITB in conditions of low fuelling, smaller torque input and dominant electron heating is another issue. The relationship between wide ITBs, ELM penetration and the control of edge pressure/current is also very important to establish.

- Study steady-state scenarios without ITBs.

LIST OF PARTICIPANTS

EU: Becoulet, Bergeaud, Campbell, Challis, Froissard, Giruzzi, Gormezano (Chair), Jaun, Koch, Lloyd, Moreau, Nguyen, Notredaeme, Righi, Rimini, Sauter, Tuccillo, Vlad

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SUMMARY OF THE FIFTH INTERNATIONAL SYMPOSIUM ON FUSION NUCLEAR TECHNOLOGY (ISFNT-5)

by Dr. J.E. Vetter, Member, International Steering Committee

The ISFNT-5 was held in Rome at the Augustianum Conference Centre on 19–24 September 1999. It has displayed valuable and detailed information on the topics around the central theme "Fusion Nuclear Technology".

The Symposium was attended by about 350 participants from 17 countries. About 350 contributed papers were presented. In the four plenary sessions, distinguished speakers presented review papers as well as papers of general interest; 48 oral presentations on specific subjects related to the conference topics were given in the parallel sessions. About 300 posters divided into 6 sessions were displayed during the symposium. It should be emphasized that more than 50 % of the contributions to this conference were a result of, or inspired by, ITER. The progress in all fields is remarkable. Database and modelling tools are steadily improving, the output from the experiments is impressive.

The ITER project has greatly helped in achieving a common understanding of what a next step shall aim at, and what can realistically be built. Through its large projects, industry has impressively contributed to a demonstration of technical feasibility and capability of larger and more complex components. Careful engineering, designing and modelling have preceded these fabrication processes.

The result, as far as already available on ITER R&D, fully corroborates the design assumptions: ITER is ready to be built.

The Reduced Technical Objectives / Reduced Cost (RTO / RC) ITER, as it is more compact than the previous design (ITER FDR), will imply re-designing of the magnet systems, the shield blanket, the vacuum vessel, the divertor and some handling equipment. Re-designing gives, however, the opportunity for improvements. Considerable improvements could be attained in the building layout and fabrication methods for the components. The R&D basis for ITER remains fully relevant, even though engineering of the divertor and remote maintenance equipment have to be modified.

Much progress was made in the fabrication and qualification of actively cooled components for the divertor. Results from fatigue testing with heat loads of 10 to 20 MW/m^2 and up to many thousand cycles are promising, and the realization of larger modules is also in progress. The discussion on the type of plasma facing material is continuing. Strong erosion and excessive tritium hold-up tends to move the swing in favour of high Z materials rather than carbon or Be.

After many years of preparation and development it can now be stated that the out-of-machine fuel cycle systems are well in hand. The purification system for plasma off-gas has demonstrated its performance by using combined gas separation and catalytic decomposition methods. Prototypical cryopumps are undergoing their first test. However, tritium behaviour inside a Tokamak is an issue for further work, as is detritiation of components on a large scale. High tritium hold-up and migration via dust, as experienced in JET, is a matter that needs better understanding and has an impact on the choice of wall materials in the future.

When mentioning JET, one has to remember that this is still the only machine that has the capability of tritium operation. Therefore, it is not only suited to giving a valuable contribution to the plasma physics program, but also fulfils an important technology mission during the next 3 years' program, especially with its tritium operation campaign. The remote exchange of the complete set of divertor elements in JET has greatly increased the confidence in the capabilities of "hot" intervention. Improvements in computer software, but also in methodology and in development of more sophisticated sensors, give by far better means of plasma control today than a decade ago. This is being impressively shown by the ITER large project on divertor remote handling development.

The continuous work on improving the database and the computational tools was a pre-requisite for the ITER design and will be of benefit for any later hardware project. Finite element programs were developed to model loads, stresses and temperatures, quantities which act as the backbone for the choice of materials, fabrication methods and the overall machine design. The tools for neutronics have been refined to an extent that it is possible today to determine neutron multiplication factors within a few per cent or perform shielding calculations over many orders of magnitude attenuation with an accuracy of 30 %.

It was the amount and quality of detailed work on ITER that makes it possible today to make clear statements on the environmental impact of a large fusion device. From the study of several beyond-design-dose-accidents, it could be concluded that evacuation of the population in the vicinity of the plant needs not to be foreseen for any hypothetical event. This important feature is transferable to a future fusion power plant, provided that proper engineering of the containment is applied.

Studies reported at the ISFNT-5 have also shown that waste, qualified for geological permanent depositories, can be reduced to almost negligible quantities, provided that appropriate materials are chosen and recycling is used.

As a consequence of the debate on an economically attractive reactor, the number of studied options of blankets increases rapidly. Improvements of already well-developed blanket concepts to obtain, inter alia, high power densities could be seen side by side with rather ingenious solutions, whose practicability needs to be proven. In fact, designing, manufacturing and qualifying a blanket is a result of many considerations that have to be simultaneously taken into account. The more traditional blankets — either based on liquid metal coolant breeder concepts or on optimized arrangements of Be and lithium compounds, mostly in form of pebble beds — entrain specific problems that are being tackled by modelling and experiments. Many contributions have addressed these issues. For the liquid metal blankets MHD effects continue to play an important role as does the prevention of tritium permeation. The pebble bed systems have to prove that thermomechanical stresses and irradiation effects do not negatively affect their functions. Progress could be seen in all the issues mentioned, but much needs to be done to obtain a reliable component.

Structural materials play a meaningful role in blanket design. Though a number of general con-tributions in this field were hosted and invited by this conference, the lack of a fusion-relevant neutron source to study irradiation effects, may jeopardize any progress in this essential part of research effort in a while from now. If the fusion community will not be able to build new facilities for testing components and modules, the field of fusion nuclear technology will dry out. Dry out in results and human resources.

In conclusion, my judgement is that the message from the papers presented at the symposium is that the fusion program is now technically ready to take an experimental step such as ITER (i.e. to enter in a regime of long-burning D-T plasma, with a-particle heating being the dominant factor), on the basis of the remarkable results obtained in six years of joint work under the ITER EDA Agreement both in design and R&D activities. For DEMO, however, nuclear technology needs a new dimension of research for which decisions on an appropriate testing facility are imminent.

IAEA TECHNICAL COMMITTEE MEETING ON ELECTRON CYCLOTRON RESONANCE HEATING (ECRH) PHYSICS AND TECHNOLOGY FOR FUSION DEVICES

by Dr. T. Dolan, Head, Physics Section, IAEA

The Meeting of the IAEA Technical Committee "Electron Cyclotron Resonance Heating (ECRH) Physics and Technology for Fusion Devices" was held in Oh-arai, Japan, 4–8 October 1999. About 96 people from 13 countries attended this Technical Committee Meeting (TCM), which was combined with the 11th Joint Workshop on Electron Cyclotron Emission (ECE) and ECRH. There was a substantial ITER participation in the Meeting, including Members of the ITER Joint Central Team (A. Costley, K. Ebisawa, and G. Vayakis, all from Naka JWS, and N. Kobayashi from Garching JWS.) and scientists working for the ITER Home Teams at their national laboratories.

Drs. T. Imai and K. Sakamoto of the Japan Atomic Energy Research Institute (JAERI) hosted the meeting. The following topics were discussed:

- Theoretical studies of wave-plasma interactions
- Plasma diagnostics using ECRH frequency waves

- Experiments for plasma heating, current drive, and electron temperature profile control
- ECRH technology.

The results of the Meeting are briefly summarized as follows:

Theory

Dr. A. Costly reviewed the status of the new "Reduced Technical Objectives/Reduced Cost (RTO/RC)" design for ITER. The new device is being designed to achieve Q = (fusion power)/(input power) ~ 10. About 5 MW of ECRH will be used to assist plasma startup. At higher power levels (~ 20 MW), ECRH could also be used to suppress neoclassical tearing modes, which would help prevent plasma disruptions.

Several papers described theoretical predictions of microwave beam trajectories, including diffraction, absorption, and change of polarization in the inhomogeneous plasmas of tokamaks and stellarators, with complex helical magnetic fields. Other authors studied the change of the plasma electron velocity distribution caused by absorption of the microwaves, and possible mechanisms by which ECRH could suppress tearing mode instabilities.

Diagnostics and Experiments

Dr. H. Bindslev (FOM, Netherlands) et al. used ECRH waves to measure the energy spectrum of fast ions in the Joint European Torus (JET) device by collective Thomson scattering, which is a difficult measurement, due to a low signal/noise ratio. Dr. D. K.Akulina et al. (General Physics Institute RAN, Russian Federation) used modulated microwaves to measure the propagation of the heat pulses in the L-2 stellarator plasma, from which they calculated the thermal diffusivity of the plasma. Dr. T. D. Luce (General Atomics, USA) et al. measured several components of the plasma current density: total, ohmic, neutral-beam-induced, and bootstrap, from which they deduced the current density induced by ECRH. The magnitude and spatial profile of the ECRH driven current was consistent with theoretical predictions.

The ECE is used as a plasma diagnostic technique to measure the electron temperature profiles, and these measurements agree well with measurements by Thomson scattering of laser beams. Dr. R. Chatterjee et al. (University of Texas, USA) developed a radiometer to measure second harmonic ECE at 234-306 GHz in Alcator C-mod. This 32-channel system can measure profiles with 1-cm spatial resolution and a 1 MHz frequency response.

Confinement improvement

ECRH may be used in the future to reduce tearing modes, to suppress sawtooth oscillations, and to help sustain internal transport barriers in tokamaks. The standard model of ECRH interactions with plasmas does a good job of modeling microwave beam propagation, polarization, and absorption. The Dutch group (Oomen et al.) used feedback control of the gyrotron to control the plasma electron temperature according to a prescribed time profile. (This was also done at JT-60U.) They also observed stabilization of the m=2 tearing mode when the ECRH resonant layer was placed inside the q=2 surface.

The W7-AS team (IPP Garching) demonstrated operation with the "electron root" of the radial electric field equation. However, future stellarators, such as W7-X, may have difficulty achieving electron root operation, due to their low trapped-particle fractions.

ECRH-induced profile control was used on the Gamma-10 tandem mirror experiment (Tsukuba University, Japan) to improve radial plasma confinement.

Technology developments

The technology talks concerned antenna design, gyrotron development, waveguide design, beam steering systems, mirror cooling, free electron lasers, and polarizers. Laboratories in Russia and Japan are working on the development of gyrotrons that can generate about 1 MW steady state at 170 GHz (for ITER) with good efficiency (> 30 %). The Forschungszentrum Karlsruhe (FZK), in Germany, is also developing gyrotrons with variable frequency, so that they can track the cyclotron resonance as plasma conditions change. They use a

Brewster angle window to accommodate all frequencies with low power loss. They have generated short pulses of 118–162 GHz radiation with powers up to 1.5 MW and efficiency of about 50%.

Development of synthetic diamond windows is progressing very well under cooperation of JAERI, FZK and the DeBeers Company. These windows have low microwave power absorption and excellent thermal conductivity. For example, a diamond window 1–2 mm thick with a diameter of several cm can handle 1 MW of power. On this subject, please see also the article by K. Sakamoto "Diamond Window and its application to ITER Gyrotron", ITER EDA Newsletter, Vol. 8, No. 1, January 1999.

B. Plaum (University of Stuttgart) used a "genetic algorithm" to calculate the optimum curvature of a waveguide bend, in order to minimize the microwave power lost. More details on this technique, which can find multidimensional optima for a wide variety of scientific problems, are available from T. Dolan.

Conclusions

Excellent progress is being made on ECE plasma diagnostics and ECRH physics and technology. The new gyrotrons will facilitate control of the plasma current density distribution, which will enable the sustainment of "internal transport barriers" in tokamaks, thereby facilitating improved plasma confinement. This research is of value to ITER and to the whole fusion community, as it is also applicable to other types of magnetic confinement devices.

The proceedings of this meeting will be published in the journal *Fusion Engineering and Design.* The next meeting, EC-12, will be held in Cadarache, France, in 2001.

The participants enjoyed a tour of the Naka Fusion Research Establishment (including the JT-60U tokamak) and the Tokai-Mura research establishment, where they saw the ITER vacuum vessel segment and remote handling tool. The Tokai-Mura site was near the location of a widely publicized accident on 30 September. Radiation levels were back to normal, and the site was safe when we visited.

Participants in the Meeting

Items to be considered for inclusion in the ITER Newsletter should be submitted to B. Kuvshinnikov, ITER Office, IAEA, Wagramer Strasse 5, P.O. Box 100, A-1400 Vienna, Austria, or Facsimile: +43 1 2633832, or e-mail: c.basaldella@iaea.org (phone +43 1 260026392).
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