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R&D ITPA ACTIVITIES IN SUPPORT OF OPTIMIZING ITER DIAGNOSTIC PERFORMANCE

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SPECIALIST WORKING GROUPS on DIAGNOSTICS

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GENERAL ATOMICS

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ABSTRACT

The mission of the International Tokamak Physics Activity Topical Group (TG) on Diagnostics is to identify and develop solutions for diagnostic techniques, which are necessary for the fulfilment of ITER scientific goals while remaining compatible with its predicted harsh environment. While much progress has been reported in the detailed design and integration of these diagnostics, some generic issues related to ITER environment are still outstanding. These issues are the focus of the high priority items for the topical group's activities and are highlighted in this paper.

1. HIGH PRIORITY ITPA R&D ITEMS

1.1. DEVELOPMENT OF METHODS OF MEASURING THE ENERGY AND DENSITY DISTRIBUTION OF ESCAPING ALPHA PARTICLES

A key mission element of ITER is the demonstration of plasma self-heating by fusion-created α-particles and the achievement of high gain. Understanding both confinement and the loss of α-particles is therefore an important scientific goal for ITER, and requires the measurement of the escaping α-particle flux. Due to their high energy, these particles experience large excursion from flux surfaces and their Larmor radius can reach up to 10 cm. Although the overall confinement is expected to be very good, some losses to the first wall (FW) are foreseen. A small but localized loss of energetic particles (e.g. alphas) may also lead to excessive heat loads on the first wall, and lead to erosion or other damage, and can cause an undesirable influx of impurities sputtered from the first wall. Therefore it is very important to understand, and mitigate these losses.

Many diagnostic tools have been developed and used on existing and previous devices to help in those studies. For example, such diagnostics have shown a strong connection between measured lost fast ions and MHD activities [1,2]; correlation in frequency and phase between measured energy-, pitch angle- and time-resolved fast-ion losses and observed Alfvén eigenmodes, and allowed to further understand the loss mechanisms [3].

However, implementation of a lost alpha diagnostic on ITER faces significant challenges due to an expected harsh environment encountered during plasma operation. Presently, the standard pitch-angle and energy-resolved loss detector technique does not extrapolate to an ITER-like environment. The necessity of detecting these energetic particles close to or at the first wall (within a Larmor radius distance) would make this detector the most directly exposed diagnostic on ITER. As such, one needs to evaluate the signal to noise ratio in these circumstances with the proper geometry and within the expected radiation field. Some of the elements coming into that evaluation have been obtained but many are still lacking. There still remain a number of open questions that may preclude the feasibility of lost alpha measurements on ITER and are described as follows:

1. To evaluate the expected signal levels, the following need to be calculated or included:
    a. Levels of losses expected at the first wall, as a function of pitch angle ($v_{||}/v$).
    b. Detection efficiency of such detector (for a given flux to the wall), i.e. the equivalent of the sustained solid angle for curved orbits.
        - Including a full description of the aperture (gyro phase selection).
        - Including a full description of the overall probe/detector (self-shadowing).
        - Including a full 3D description of the first wall around the tokamak (probe shadowing), and especially near the detector.

   c. Optical efficiency (or equivalent for a non-optical version) of the full system to verify that expected signal can be detected by remote camera or other recorder.

      - Including scintillation efficiency (photons/ion).
      - Including scintillation degradation with temperature and radiation damage.

2. To evaluate the noise levels (likely to be background related), the following need to be calculated, measured and included:

   a. Detector response to neutrons, gammas and possible secondary electrons.

   b. Calculate background levels due to the expected radiation levels using a Monte-Carlo Neutron Particle (MCNP) calculation.

      - Include full angular dependence of the radiation in the expected scintillation efficiency calculation.

3. To evaluate its technical feasibility, one must calculate the expected heat flux to the probe and its aperture. An arbitrary increase in shielding and cooling capability is likely to result in a reduction of the measured energetic ion flux due to self-shadowing effects (e.g. a very thick and robust aperture would also block some of the alphas from penetrating the body of the probe and reach the detector/scintillator). An evaluation of the required opening in the first wall for the particles to reach the detector is also needed.

Much information has been gathered over the years in support of the first task (1a). The calculations require a full orbit code (with gyro-radius), inclusion of field ripples (TF and others) and a relatively complete description of the first wall geometry. Latest calculation results are shown in Fig. 1 for the escaping flux impacting the center of blanket module 15 (slightly below the outer midplane) for both Scenario 2 and Scenario 4 cases. However, these calculations are for prompt (first-orbit) losses, and do not take into account any finite detector geometry and aperture as required in task (1b).

Candidate scintillators have been studied over the last few years and the best prospect remains the Ce:YAG ceramic scintillator. Tests made in Japan [4,5] indicate that this scintillator has good scintillation efficiency for ions, a manageable temperature dependence (showing a significant drop at $T > 300°C$), tolerable scintillation efficiency to normally incident neutrons and gammas, but issues remain for radiation damage from neutrons and impacting ions. Other candidates include Faraday Cup Detectors and secondary gamma detection.

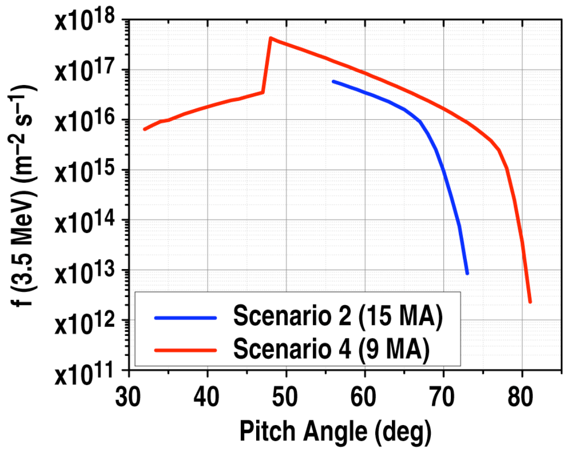

FIG. 1. Pitch-angle distribution of escaping flux of fusion alphas (3.5 MeV) for Scenario 2 and Scenario 4 at the center of blanket module 15.

Initial studies have been undertaken to evaluate the thermal handling of a probe schematically located as shown in Fig. 2. In this basic thermal analysis, the probe is passively cooled through radiation with the surrounding blanket modules. In this case, 2 MW/m^2 of parallel heat flux together with 0.5 MW/m^2 of perpendicular radiative heat flux were used; nuclear heating was omitted. Calculations show that the inner side of the probe made of copper (CuCrZr) rises in temperature to at least 600°C (or 3000°C for stainless steel), thus requiring active cooling. Further analysis will need to include transient heat loads, which can reach up to 5 MW/m^2 or during ramp-up, where they could reach up to 40 MW/m^2 with an outboard plasma initiation. The possibility of a radially scanning probe is being evaluated in order to reduce the steady-state heat flux, but with the loss of continuous monitoring of the escaping fast ion flux.

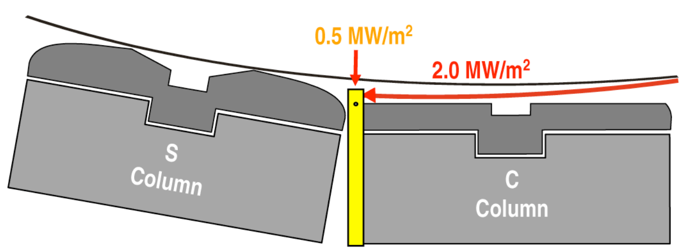

FIG. 2. Possible configuration of lost alpha probe within the outer first wall (top view).

1.2. ASSESSMENT OF THE CALIBRATION STRATEGY AND CALIBRATION SOURCE STRENGTH NEEDED FOR THE NEUTRON DIAGNOSTICS

A direct and key measurement for confirming the energy gain in ITER is the fusion power. Overall, the neutron diagnostics will be primarily used to measure the fusion yield, source profile, and the fuel ratio (n_T/n_D). In order to achieve the required 10% accuracy in total fusion yields, a functional combination of all neutron diagnostic measurements will be required [6]. Neutron diagnostics sub-systems can be grouped into four categories: flux monitors, activation systems, cameras (source profile), and spectrometers. Each parameter is measured by one or two primary systems, while being supported by secondary systems in order to reduce uncertainties. These uncertainties are principally based on the in-situ calibration, detector cross-calibration, and the establishment of reliable neutron transport calculations. In addition, since the needed dynamic range in neutron detection is rather large (up to 12 orders of magnitude i.e. 10^8–10^{20} n/s), and since limited strength calibration sources are available (<10^{10} n/s), cross-calibrations between detectors will be necessary, as it is normally performed in existing devices. This can be done with dedicated plasma discharges, and supported by appropriate MCNP calculations.

The fusion yield will be determined by a combination of multiple time-resolved neutron flux monitors distributed around the tokamak. These systems will be calibrated in-situ, but the measurements will also be supported by the profile monitors (cameras) and spectrometers. Since the sensitivity or detection efficiency of the flux monitors is likely to change over long periods of time, the fusion yield is also measured by the activation system, which is more reliable over time but without sufficient time resolution. In addition, smaller corrections due to varying plasma positions and neutron source profile will be provided by the profile monitors (cameras). Finally, the high-resolution neutron spectrometer will also provide an independent measurement of the fusion yield and ion temperature, but does not require an in-situ calibration. It, however, only samples a small line-integrated volume near the center of the tokamak.

With the importance of the accuracy of the fusion yield, and with presently no sufficiently strong neutron generators, it was recognized that a proper calibration would require a significant amount of time and/or additional tools. In order to estimate the calibration time, MCNP calculations have been performed for the micro fission chambers at their planned locations using the Alite-ITER model. In this case with a DT source of 10^{10} n/s, the required calibration time to obtain the necessary statistics would be 3 h, and 30 h, for 1000 counts (3% statistics), and 10,000 counts (1% statistics), respectively. Many additional points will be needed in order to ascertain the toroidal and poloidal (profile) dependence of the calibration. This process may require as much as 2 weeks, without accounting for the setup time. However, other flux monitors can be calibrated simultaneously, and since they are more sensitive, this may reduce the calibration time. In addition a similar calibration is needed for DD operations, but since the available generators are less efficient (by ~2 orders of magnitude), more time would be required for a similar accuracy, if desired.

For the neutron activation system, a calibration of niobium foils through the reaction ^{93}Nb(n, 2n) ^{92}Nb would take approximately 4 days using a 14 MeV neutron generator of 10^{10} n/s fixed at one location. A total of ~3 source positions is sought in order to compare with and to validate the neutron transport calculation. Although some overlap may be possible with the calibration of the micro-fission chambers, dedicated time would be required. In addition, for the DD operation calibration, the development of a DD generator of $>10^{10}$ n/s is needed, with presently no credible development options.

In the case of the source profile measurements, in order to assess the effects of the absolute observation volume, scattering effect, and cross-talk of each collimator, the cameras (radial neutron and vertical neutron cameras) also need to be calibrated in-situ, although some preliminary work can be done off-line in a separate laboratory. The exact number of required calibration source points depends on the collimator size and exact geometry, and therefore would be determined along with their final design. This calibration procedure would be carried out using the most sensitive camera detector installed. Conservatively, a 100-source point calibration, which necessitates 3 h for each point (for statistical reasons), would require an additional 2 weeks.

Prior to the full in-situ calibration, a short in-situ calibration of the neutron diagnostics is highly desirable before the commissioning of the heating systems. However, this can be most efficiently accomplished only after the installation of most (if not all) blankets, port plugs, divertor cassettes, and the major heating systems and other large components. This initial, short calibration would simply require the source to be fixed at one or few positions near the Equatorial Port 1, so that most of neutron detectors could be tested and initially calibrated. This initial procedure would contribute significantly in optimizing the full calibration, and to validate the MCNP model previously devised.

Consequently, while taking into account presently available generator yields, and the present design, it is estimated that 2 neutron calibrations will be required, of ~2 and ~8 weeks duration respectively, which does not presently include the usually significant setup time. Efforts are continuing in devising ways to optimize the number of calibration steps and to reduce the time necessary to complete it, while meeting the required accuracy. Options include additional or more sensitive detectors, additional calibration sources or slight changes to detector configuration. Additional efforts will be required to minimize the self-shadow effects of the source/generator on the actual calibration results.

1.3. DETERMINATION OF THE LIFETIME OF PLASMA FACING MIRRORS USED IN OPTICAL SYSTEMS

All optical diagnostics in ITER will be using mirrors as plasma-viewing optical elements. The reflectivity of diagnostic mirrors will degrade mainly due to the deposition of impurities and erosion of mirror surfaces by energetic particles. The optical properties of mirrors will largely define the overall performance of the diagnostic systems; therefore the performance of mirrors in the ITER environment represents an acute issue. Many R&D efforts are presently focusing on addressing the most severe issues of mirror performance: studies of the mirror characteristics under erosion and deposition conditions in present-day tokamaks and modeling extrapolation to ITER, mirror surface recovery and protection techniques including active and corrective mitigation of deposition, in-situ mirror calibration, and an assessment of the irradiation effects on mirrors. The detailed overview of these activities, the newest results, and an assessment of their applicability to ITER are described in a companion paper [7] along with evaluation of risks and possible consequences of the mirror failure to ITER operation.

1.4. ASSESSMENT OF REQUIREMENTS AND OF TECHNIQUES FOR THE MEASUREMENT OF DUST AND EROSION

For both the safe operation and meeting the nuclear regulatory requirements, an accurate inventory of the tritium and dust inside the vacuum vessel is required. In connection with this requirement, an accounting of the divertor plate erosion will also be necessary. Recent studies indicate that the inventories for dust and tritium are expected to reach their maximum limits on a timescale comparable to the divertor target erosion lifetime. Based on this, a control strategy for dust and tritium has been formulated. Dust will be removed during the scheduled divertor replacements (approximately every 4 years). In addition, the dust will be monitored during shutdowns. During the replacement of the divertor cassettes, local measurements will be benchmarked against the tritium and dust recovered. The first benchmarking would be done in the hydrogen phase. To support this strategy additional diagnostics were recently added to the ITER baseline for measuring dust and erosion. They are shown in Table 1.

TABLE I
PROPOSED DUST AND EROSION MEASUREMENTS AND ASSOCIATED TECHNIQUES

System	Proposed Technique(s)	Main Contribution	Feasibility/Status
Divertor erosion monitor	Speckle interferometry	Non-invasive limit to dust generation rate	Proven in lab, but not in tokamaks, needs R&D
Local dust monitor	Removable samples	Benchmark dust generation rate during a shutdown	Proven in existing tokamaks, needs remote handling
Local dust monitor	Capacitance micro-balances	Non-invasive local dust measurement	Proven in lab, but not in tokamaks, needs R&D
Tritium wall retention/content	Laser-induced desorption (LID)	Non-invasive local tritium retention data	Demonstrated in tokamaks, needs further R&D and remote handling

However, an outstanding issue remains for the measurement of hot dust, for which a finalization of the measurement requirements is not yet complete. The maximum amount of hot dust is simply derived from the maximum rise in pressure (2 bar) that can be tolerated following a steam ingress incident, which could result in a possibly explosive situation due to hydrogen produced in beryllium/steam or tungsten/steam reactions, if an ingress of air is also postulated. That maximum pressure rise limits the total amount of hydrogen within the vacuum vessel to 4 kg. Knowing the reactivity of beryllium and tungsten, this corresponds to a limit of 18 kg of hot dust if it is assumed to be all beryllium. When compared to the total amount of dust (cold and hot) that could be found within the vacuum vessel, while remaining within the tritium inventory limit (as discussed above), it is estimated that up to ~40 kg of dust could be uniformly distributed on hot surfaces, which would represent a factor of 2 above the safe limit for explosion hazard. Consequently, a separate requirement is being devised for hot dust limits in line with the 18 kg derived above. Techniques to address this remaining need have not been identified as of yet, and options are being developed. Presently, the main proposal consists in measuring the chemical reactivity inside the tokamak with a controlled injection of water (steam) during a bake. This technique presents the advantage of being a global measurement, consistent with the expected measurement requirement. This technique can be complemented by qualitative study of IR emissivity of key surfaces.

1.5. THE ASSESSMENT OF IMPACTS OF IN-VESSEL WALL REFLECTIONS ON DIAGNOSTICS

In tokamaks, many of the optical diagnostics have to function against the background of scattered (reflected, re-emitted and stray) light originating from the reflection and re-emission of plasma light by the first wall. In ITER this problem is likely to be more serious than that experienced thus far. This is largely due to its larger size, its conformal and more reflective first wall. This much larger background due to scattered radiation indicates a need for a quantitative evaluation through modeling, coupled with detailed measurements of the reflectivity of relevant materials, in the relevant conditions. Measurements are required to determine the need for in-situ checks of the evolution of the reflectance during long exposure to plasma discharges. For this analysis, there is a growing consensus requiring the development of an approach based on bidirectional reflectance distribution function (BRDF) calculations, which is widely used in other fields, for the standardization of reflection coefficients. Commercial software packages have been evaluated, and many appear suitable for the task. The modelling calculations remain to be confirmed on a tokamak with a set of completely characterised tiles. The initial results will be compared with diagnostics presently installed on the JET tokamak, following the modification of the first wall to a metallic surface (ITER-like wall). Very similar cases were encountered in the metallic-wall Tore-Supra, where significant reflection issues were encountered in the visible (MSE system) [8] and in the infrared (viewing system).

As input to the calculations of scattered light, the code requires data on the (diffuse and specular) reflectivity of beryllium tiles. This data should include the light intensity reflected at all angles (azimuthal and sagittal) for every angle of incidence. In practice, measurements have been made on inconel tiles (as a proxy for beryllium) and at certain specific incidence and reflection angles. The measurements have been extended over the whole four-dimensional domain using a heuristic model. This model for the bidirectional reflectance distribution function (BRDF) contains a limited number of free parameters which are adjusted to fit the available measurements. The software packages evaluated have the capability of importing directly 3D CATIA models, thus simplifying the needs for special handling of invessel structures.

When simulating Scenario 2 profiles (flat density, Z_{eff}=1, H-mode plasma) the results show that a chord viewing the plasma tangentially, terminating on the outer wall, would experience approximately 10% contribution of scattered light. A radially viewing chord, terminating on the inner wall, would suffer from around 30% scattered light. There are presently no plans to install viewing dumps in ITER and corrections to the Z_{eff} measurements, for example, will need to rely on estimates of the level of contribution due to scattering.

Measurements of the charge exchange emission from near the magnetic axis can also be affected by scattered light. The diagnostic neutral beam is attenuated by a factor of 100 between the edge and the centre in the high-density baseline scenario. Emission from the charge exchange line at the edge of the plasma will be scattered by reflections and will contribute to the signals seen by the core views. Since the intensity ratio between the edge and the core is so large, simulations indicate that the scattered signal will be of similar magnitude to the local emission in the core. It is already planned to modulate the ITER diagnostic beam in order to enhance the discrimination of the charge-exchange features over the background spectra, but this modulation will not remove the scattered charge-exchange component. The difference in ion temperature (spectrum) between the core and scattered edge radiation may assist in interpretation of the spectra but is likely to be difficult. Provisions are being explored to include a second set of views parallel to the lines of sight but which do not intersect the diagnostic beam. Such views would receive only the scattered/reflected signal, not the direct charge-exchange signal, hence a joint analysis of pairs of spectra is expected could yield the local ion parameters.

1.6 ASSESSMENT OF THE MEASUREMENT REQUIREMENTS FOR PLASMA INITIATION AND IDENTIFICATION OF POTENTIAL GAPS IN PLANNED MEASUREMENT TECHNIQUES

In existing devices, diagnostic systems are typically optimized for the near-stationary or flattop phases of the discharge. In ITER, the early phase of plasma formation and control may require additional or special measurements different than during the flattop phase. The phases in question include between pulses, breakdown, ramp-up and ramp-down periods. Initial assessment has been performed, and a preliminary list of needs is set out in Table II. The full evaluation of these needs, in terms of physics requirements and performance of the proposed techniques remains to be completed. In addition, a careful attention to calibration requirements will be necessary as many diagnostics are expected to be at the limit of their planned dynamic range during these periods. Coordination of these requirements with the needs of the plasma control system (PCS) has just been initiated.

TABLE II
PRELIMINARY LIST OF MEASUREMENT NEEDS FOR PLASMA INITIATION AND TERMINATION

Period	Measurement Needs (Not Exhaustive)
Between pulses	IR afterglow and wall temperatures, gas composition, tritium levels, dust, erosion, window transmission, inspection
Breakdown	Plasma current (lower level), imaging (visible), electron density, null structure, ECH stray power, impurities
Ramp up	Impurities, density profile, global radiation levels, runaways, energy content, position measurement, current profile
Ramp down	Full reconstruction (shape), runaways and runaway beam position, plasma position measurement

2. OTHER ITPA ACTIVITIES

2.1. IMPACTS OF IN-VESSEL STRAY MICROWAVE RADIATION

Stray, or unabsorbed microwave radiation generated during *off-normal* tokamak operation will have a major detrimental impact on the ITER microwave based diagnostics, such as reflectometry and electron cyclotron emission (ECE). Stray radiation can arise from non or poorly absorbed microwave beams, e.g. during low plasma density breakdown assist, exotic heating schemes or fault conditions, as well as from fast/runaway electron generated Bremsstrahlung during breakdown and disruption events. Even moderate radiation levels can cause corruption of diagnostic signals and measurement misinterpretation through power loading/saturation of detectors, while more extreme levels will destroy sensitive semiconductor detectors and waveguide components through overload, cavity resonances and arcing as well as thermal heating. Other non-microwave diagnostics, such as bolometers, SXR and IR cameras, plus in-vessel components can also face potential damage.

Presently employed microwave protection techniques, such as waveguide isolators and filters, may not be suitable for ITER applications due to the expected very high levels of stray radiation. In current devices, in presence of plasmas, normal stray radiation levels for 1 MW launched microwave power are typically several milliwatts at the diagnostic detector. Under fault (no plasma) conditions this can rise by a factor of 100 to 1000, or even more for a direct hit on a diagnostic antenna by a microwave beam. The 24 MW heating system planned for ITER may easily overwhelm the protection components with sub-mm dimensions, as shown in Fig. 3 [9]. On ITER several protection systems may be required in parallel to cover different fault conditions and different diagnostic weaknesses. Such systems may include fast-acting waveguide shutters, filters, fuses or sacrificial elements as well as exotic radiation absorbing materials and gases. However, some systems still require significant R&D before deployment. A particular priority is the development of high power rejection filters. Effort is also being devoted to the development of simulation and prediction codes to better estimate individual diagnostic exposure levels. Codes, together with high power microwave test facilities & stands can be used to identify and harden diagnostic weaknesses prior to tokamak installation. Finally, a more extensive risk assessment of the frequency and degree of fault conditions will allow a better definition of the appropriate response and degree of protection to be expended. For example, extensive R&D efforts are ongoing at W7-X to quantify and study the effects of stray microwave radiations onto diagnostics and other in-vessel components.

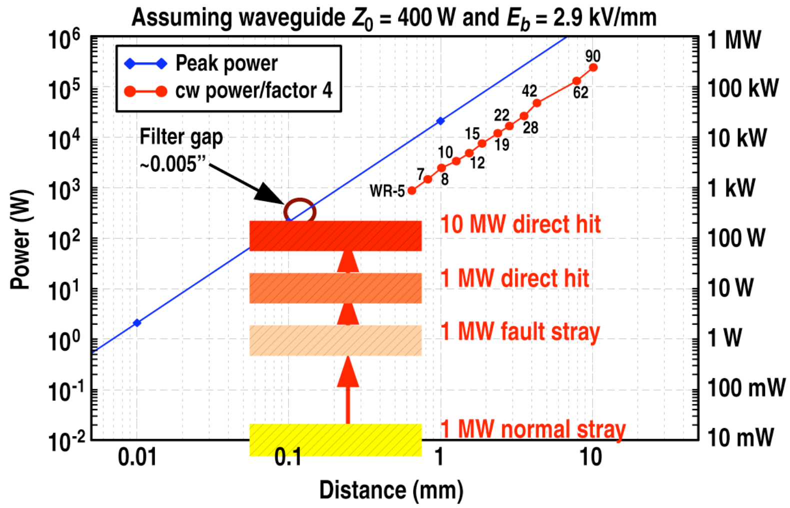

FIG. 3. Arcing threshold for power transmitted through waveguides as a function of gap distance. Blue line is ideal case, and red the empirical limit. Circle indicates a standard protection filter (gap).

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