Extensive remote handling and conservative plasma conditions to enable fusion nuclear science R&D using a component testing facility

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Abstract. The use of a fusion component testing facility to study and establish during the ITER era the remaining scientific and technical knowledge needed by fusion Demo is considered and described in this paper. This use aims to test components in an integrated fusion nuclear environment to discover and understand the underpinning physical properties, and to develop improved components for further testing, in a time-efficient manner. It requires a design with extensive modularization and remote handling of activated components, and flexible hot-cell laboratories. It further requires reliable plasma conditions to avoid disruptions and minimize impact, and designs to reduce the divertor heat flux. As the plasma duration is extended through the ITER level ($\sim 10^3$ s) and beyond, physical properties with increasing time constants would become accessible for testing and R&D. The longest time constants of these are expected to be many days ($\sim 10^5 - 10^6$ s). Progressive stages of research operation are envisioned in deuterium, deuterium-tritium at the ITER-level, and deuterium-tritium with increasingly longer plasma durations.

1. Introduction

As the world fusion energy programs enter into the ITER/Burning Plasma era, the U.S. fusion energy sciences community has begun the process of preparing a plan that will guide its efforts during the next 15-20 years. The remaining areas of scientific and technical questions that must be resolved to proceed to Demo were identified in a recent FESAC report [1]. Two of these areas address issues anticipated for the Demo fusion nuclear environment:

Taming the Plasma Material Interface – deals with *material components that interface the hot plasma in the presence of very high neutron fluences*; and
Harnessing Fusion Power – deals with *systems that can covert fusion products to useful forms of energy in a reactor environment, including self-sufficient supply of tritium fuel*.

It is therefore timely to develop an understanding of how a Component Test Facility (CTF), previously proposed [2] to test and demonstrate Demo component technologies [3], could be reconsidered for addressing the fusion nuclear science issues contained in these two areas. In particular, the stages of testing envisioned [3] for technology testing and demonstration would include Stage 1: "*fusion 'break-in' & scientific exploration*," requiring an estimated fusion neutron fluence of ~0.3 MW-yr/m^2, Stage 2: "*engineering feasibility & performance verification*," requiring ~3 MW-yr/m^2, and Stage 3: "*component engineering development & reliability growth,"* requiring ~6 MW-yr/m^2.

It is clear that the above-mentioned fusion nuclear science issues would be addressed during the "scientific exploration" stage in an integrated fusion environment, which aims to establish the knowledge base required to design and build the Demo-capable components. As depicted in FIG. 1, this science-oriented R&D would encompass the following phases:

I: Systems shake-down in hydrogen,
II: Testing and commissioning plasma facing components, radiation shielding, and instrumentation capabilities in deuterium,
III: Testing and commissioning at the ITER-level (plasma durations ~10^3 s) in D-T, and
IV: Fusion nuclear science R&D with progressively increasing plasma durations (~10^4 – 10^6 s).

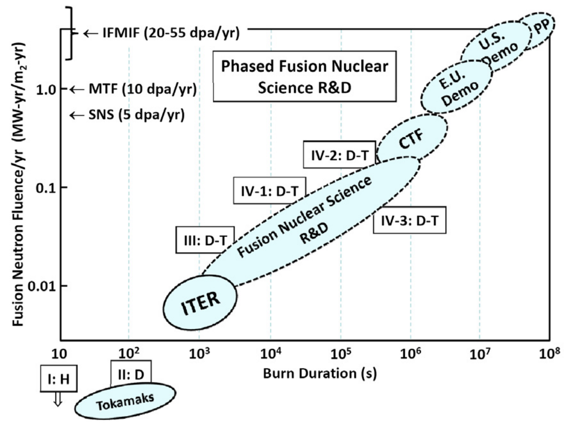


FIG. 1. Phases III and IV of fusion nuclear science R&D in D-T with increasing plasma durations, following phases I and II in H and D, respectively.

For Phase II, particle recycling and wall pumping in long duration discharges on TRIAM-1M indicated time constants of ~10^2 s [4], related to the mechanism of wall saturation in the presence of molybdenum co-deposition. Deuterium implantation and diffusion in the material bulk were estimated on Tore Supra [5] to have a combined time constant of ~10^2 s. For Phases III and IV, time constants of interest would include [3], in the presence of fusion neutron irradiation, tritium release from LiPb breeder (~10^3 s), from Li_2O breeder (~10^4 s), and from Li_4SiO_4 breeder (~10^5 s), and tritium diffusion through stainless steel (~10^5 s) and ferritic steel (~10^6 s) at high material operating temperatures (~500°C).

As neutron fluence per year increases through Phase IV beyond 0.1 MW-yr/m^2-yr, so will the radiation effects [6] on the properties of materials and material combinations (such as a tritium permeation barrier applied to the surface of ferritic steel) increase in the test components. This would contribute new information on the material to be selected for high-dpa testing such as on the IFMIF [7] and other possible irradiation sources[1]. The previously defined [3] Stages 2 and 3 of the Demo component technology demonstration in a CTF program would begin after the knowledge base for Demo-capable designs and components are obtained. This further research in turn is expected to contribute to and benefit from high-dpa irradiation R&D in establishing the technology capabilities for Demo.

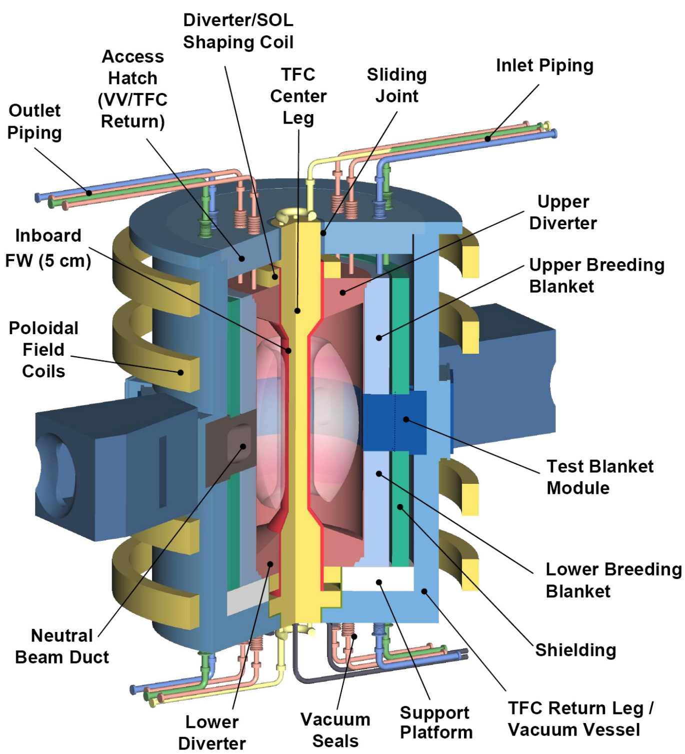
The duty factor to be achieved through this fusion nuclear science R&D will depend strongly on the progress of testing, discovery and understanding of the physical properties of interest, leading to improvements to the components. This in turn will require high maintainability of the facility to continue the R&D efficiently. Section 2 summarizes a design concept to accommodate extensive modularization and remote handling, and allow conservative plasma assumptions including an extended divertor channel to reduce divertor heat fluxes to the ITER level. Section 3 describes the concept of extensive component modularization and capability for remote handling, and estimate the replacement times of various test components. Section 4 addresses the issues relating to avoiding disruptions, limiting divertor heat fluxes, and

[1] Private communications on options under consideration, from S. Willms on Material Test Facility (MTF) and J. Haines on Spallation Neutron Source (SNS) beam dump.

minimizing their impact in such a design, to help ensure reliable plasma operation as the plasma duration is progressively increased from ~10^3 s toward ~10^6 s. Summary and discussion is provided in Section 5.

2. A test facility for fusion nuclear science R&D

The assessment results of this test facility, as an extension of earlier results [2], are presented in FIG. 2, including key parameters. It is seen that the device remains relatively compact in size (major radius, R_0 = 1.2 m), small in the plasma aspect ratio (A = 1.5), moderate in fusion power (P_{Fusion} = 75 MW), and large in fusion neutron wall power flux (W_L = 1 MW/m^2). It is assumed that the electron energy confinement time scales as 0.7 times the ITER H-mode scaling, while the ion energy confinement scales as 0.44 times the neoclassical. The resulting H-Ion H-Mode (HIHM $T_{avgi} > T_{avge}$) plasma would then have a global energy confinement time that is 1.5 times the ITER H-mode scaling. Relatively conservative plasma parameters are estimated, including β_T (=18%), β_N (=3.8), q_{cyl} (=3.7), bootstrap current fraction (~0.5), and n_e (=1.05×10^{20}/m^3), far removed from known limits of stability [8, 9].



W_L [MW/m^2]	0.01	1.0	2.0
R0 [m]		1.20	
A		1.50	
kappa		3.07	
qcyl	4.6	3.7	3.0
Bt [T]	1.13	2.18	
Ip [MA]	3.4	8.2	10.1
Beta_N	3.8		5.9
Beta_T	0.14	0.18	0.28
n_e [10^{20}/m^3]	0.43	1.05	1.28
f_{BS}	0.58	0.49	0.50
T_{avgi} [keV]	5.4	10.3	13.3
T_{avge} [keV]	3.1	6.8	8.1
HH98		1.5	
Q	0.50	2.5	3.5
P_{aux-CD} [MW]	15	31	43
E_{NB} [keV]	100	239	294
P_{Fusion} [MW]	7.5	75	150
T M height [m]		1.64	
T M area [m^2]		14	
Blanket A [m^2]		66	
$F_{n-capture}$		0.76	

FIG. 2. A test facility for fusion nuclear science R&D, with example parameters for deuterium operation at I_p = 3.4 MA, and deuterium-tritium operation at I_p = 8.2 MA (W_L = 1 MW/m^2) and 10.1 MA (2 MW/m^2).

The essential design features include a single-turn normal conducting toroidal field magnet center post; a relatively thin startup solenoid magnet on the center post using mineral insulated conductor (MIC); continuous energetic neutral beam injection; space for super-X divertors [10]; modular components allowing extensive remote handling; structural components hidden behind the test components; and ex-shield hands-on access to vacuum seal and services.

The parameters are chosen [11] to minimize R_0 as a function of the aspect ratio A, which in turn minimizes the Toroidal Field Coil (TFC) center post mass. This process is constrained by a set of engineering and physics [12] conditions that set the boundaries within which a best design can be found. A further requirement is that the total area for the mid-plane test module access be no less than 10 m^2, under the constraint that the height of the access is proportional to the plasma height as a result of constrained locations of the outboard poloidal field coils.

Results of this calculation are provided in FIGs. 3 and 4. It is seen that the design with the smallest R_0 is obtained at A=1.5 when the peak TFC temperature reaches the limit of 150°C, Further reduction in A would lead to increased device size. As the aspect ratio increases, R_0 is increased to maintain the mid-plane access for test modules. A minimum in the TFC center post mass is also obtained at A = 1.5, while the total electric power required by the TFCs and the auxiliary heating and current drive systems remains at 200 MW or below over a range of nearby A values.

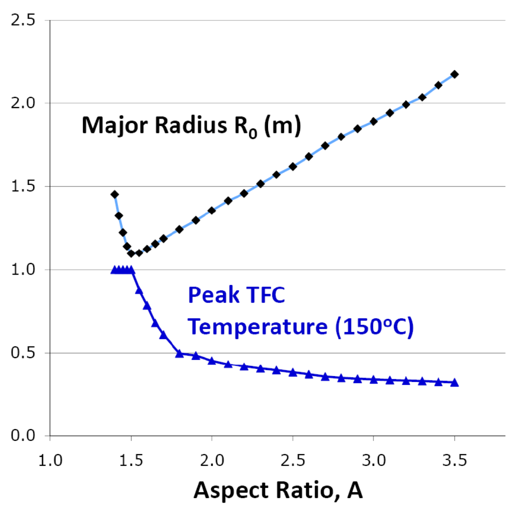


FIG. 3. Similar designs with minimum R_0 and the peak TFC temperature relative to the limit of 150°C, as a function of A

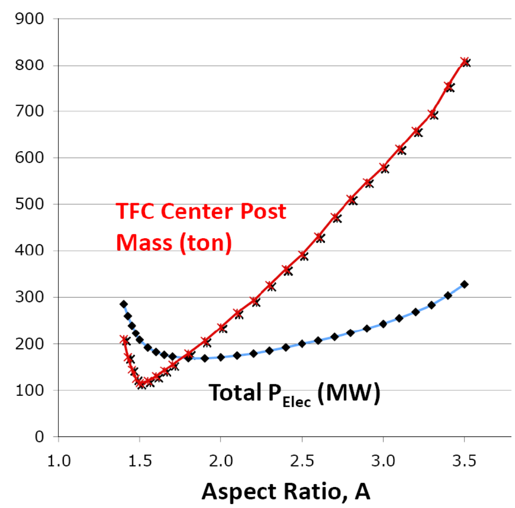


FIG. 4. TFC center post mass and the total electrical power supplied, as a function of A

3. Component modularization and remote handling

Extensive component modularization will be needed to enable remote handling of all activated components. This in turn enables time-efficient fusion nuclear science R&D in an integrated fusion nuclear environment that encompasses cycles of

1. Testing the multi-physics synergistic properties of a component to discover potentially new physical properties of interest,
2. Study these properties in the hot-cell laboratories to understand the properties and assess their consequences, and
3. Innovate and develop improved components for continued testing.

A design concept that aims for extensive modularization is depicted in FIG. 5, shown in a sequence of device disassembly, which requires only linear movement of activated components following disconnection of services and cutting of vacuum seals outside of the shield boundary. The activated components include the divertors; the divertor coils; the upper

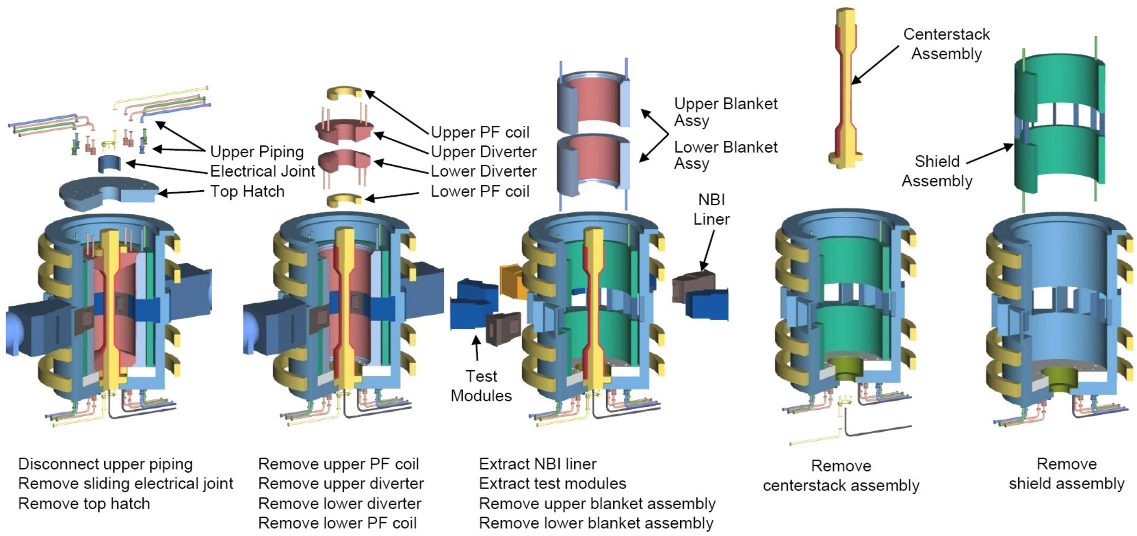


FIG. 5. A design concept that aims for extensive modularization of activated components.

and lower blanket assemblies; the mid-plane test modules, neutral beam injection systems, RF module, diagnostic module; the TFC center post; and the shield assembly. The vacuum vessel that contains the TFC return conductor and the outboard poloidal field coils are not activated and would normally remain in place.

The accompanying concept for remote handling (RH) is shown in FIG. 6. The RH systems include mid-plane port assembly handling casks, vertical port handling casks, and nearby hot-cell laboratories with extensive servomanipulators, tools, and scientific instruments. The casks are similar in concept to those designed in ITER [13] to handle port assemblies for diagnostics, RF, and the Test blanket Modules (TBMs).

The design approach minimizes interference among mid-plane casks and with the vertical port assembly casks. The time to replace an activated component can therefore be estimated assuming fully enabled RH systems and available replacement components.

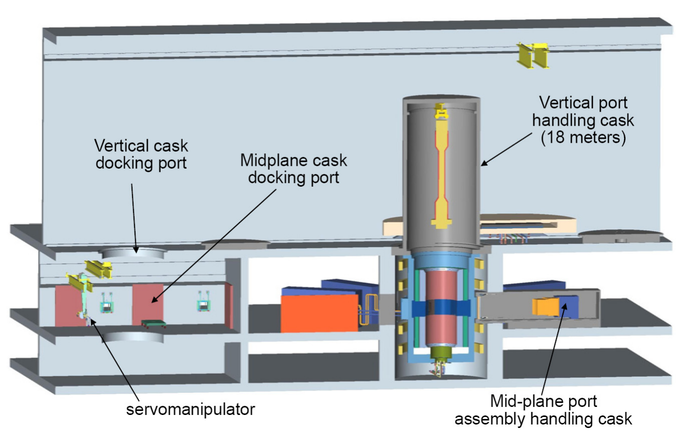


FIG. 6. Remote handling concept.

The replacement times, not including facility shutdown and startup, are summarized in Table 1. The facility shutdown and startup is estimated to require 4 weeks, which encourages simultaneous multiple replacements using parallel operation that require parallel RH capabilities. To replace both divertors, six mid-plane port assemblies (half in number from the total), and the NBI ion sources, applying multiple casks in parallel operation, will require an estimated time of 9 weeks, assuming that the spares are available, leading to a total down time of 13 weeks (25% of a year). A single unplanned shut-down and replacement of a mid-plane port assembly, however, would require a total of 7 weeks. Such maintenance capabilities represent an order of magnitude improvement beyond the present designs of major toroidal facilities including ITER.

TABLE 1. ESTIMATES OF CUMULATIVE RH REPLACEMENT TIMES

Component	Time (wks)
Neutral beam ion source	1
Mid-plane port assemblies	3
Neutral beam internal components	3
Upper divertor module	4
Lower divertor module	6
TFC center stack	6
Upper breeding blanket	6
Lower breeding blanket (mid-plane modules retracted)	9
Scheduled replacement of 2 divertors, 6 mid-plane modules, and NBI sources	9

Assuming an equal amount of time for unscheduled maintenance, about 50% time of a year would be available for fusion nuclear science R&D. A duty factor of ~10% annually would be within reach if a duty factor of 20% can be achieved during this R&D time period.

4. Disruption avoidance, mitigation and reduced divertor heat fluxes

To achieve a duty factor of 10% will further require a sound strategy to avoid disruptions, mitigate their impact, and limit the divertor heat fluxes to the ITER design levels. This would help ensure reliable plasma operation over increasing plasma durations beyond 10^3 s.

Plasma operation with large margins to known stability limits: Major disruptions can be triggered by large scale MHD instabilities [14], which in turn could be initiated by internal reconnection, ballooning, or neoclassical tearing modes, or by external resistive wall or locked modes. The latter are caused by proximity to the stability beta and safety factor q limits and enhanced by the presence of significant non-axisymmetric error fields. There are further disruptions near the density limit [14], which is lowered by the presence of significant impurity content.

The parameters in FIG. 2 for the deuterium and separately for the deuterium-tritium operation producing 1 MW/m^2 have included relatively large margins to these instability limits. These include a factor of 1.5 in β_N [8] and q_{cyl} [15], a factor of 3 in density, and minimization of error fields by designing the conductors near the plasma with a high degree of periodic symmetry (see FIGs. 2 and 5). The relatively stable regime will provide increased flexibility for the test facility to control and maintain stable plasma profiles as the plasma duration is increased beyond 10^3 s.

Disruption mitigation and reduced divertor heat fluxes: Using the plasma parameters for 1 MW/m^2, the disruption and disruption consequences [14] for the test facility can be estimated in contrast with JET and ITER. The results are provided in Table 2.

It is seen that the test facility has poloidal and thermal stored energies, heating power, the relative force due to induced eddy current (enhanced by the TPF), and eddy current wall heating factor no more than twice the JET values. For thermal quench, the effective divertor area can be extended by a factor of 2 using the Super-X Divertor (SXD) [10] (see FIG. 7). As a result, the heat pulse on the divertor is estimated to be about 2.2 times and 1/6 of the

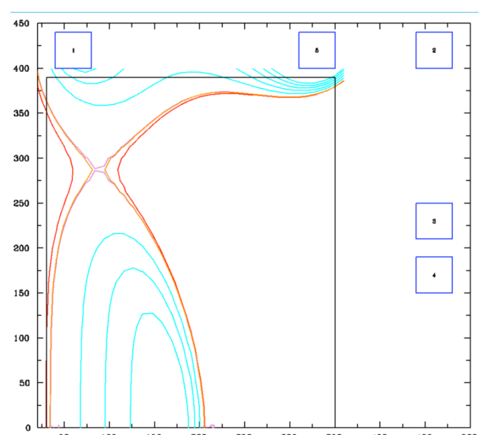


FIG. 7. An example of SXD applied to the test facility

TABLE 2. DISRUPTION AND DISRUPTION CONSEQUENCES

Parameter	JET	test facility	ITER
R (m)	2.9	1.2	6.2
a (m)	0.95	0.8	2.0
κ_{95}	1.6	2.8	1.7
V (m^3)	86	47	831
S (m^2)	145	95	683
B_T (T)	3.45	2.2	5.35
I_p (MA)	4.0	8.2	15
q_{95}	3.0	7	3.0
$P_{heating}$ (MW)	30	46	150
Poloidal field W_{mag} (MJ)	~11	~10	395
Plasma thermal W_{th} (MJ)	~12	~23	353
Current quench t_{CQ} (ms)	9.4	6	35.6
Relative eddy current force $B_T * dB_p/dt * TPF$ (T^2/s)	374	356	322
Melt layer eddy current heating factor $W_{mag}/(A_{FW} * t_{CQ}^{0.5})$ ($MJ/m^2/s^{0.5}$)	0.78	1.35	3.1
I_{halo}/I_p	0.45	0.4	0.4
Toroidal peaking factor (TPF)	1.7	1.2	2
Divertor radius R_{div} (m)	2.9	2.5	6
Effective H-mode divertor area A_{div} (m^2)	~1.6	~1.4	~3.5
Thermal quench deposition $U_{TQ} = W_{th}/7A_{div}$ (MJ/m^2)	1.07	2.4	14.1
Thermal quench t_{TQ} (ms)	0.32	0.2	0.70
Melt-layer energy deposition $U_{TQ}/t_{TQ}^{0.5}$ ($MJ/m^2/s^{0.5}$)	60	170	530

JET and ITER values, respectively. The thermal quench melt layer heating of a W divertor in the test facility is about 1/3 the ITER value, which is still 3-4 times the W melt onset value.

Using the SXD on the test facility limits the value for $P_{heating}/A_{div}$ to ~0.8 times the ITER value. Given a similar divertor design and operating scenario to ITER, the test facility divertor steady state and transient heat fluxes are expected be less than those of the ITER design [16]. The major divertor R&D needed by Demo therefore stems from the extension of the plasma duration beyond the ITER level (~10^3 s) in the presence of increasing neutron fluences while requiring very stringent tritium accountability [1].

Divertors with different or higher heat fluxes designs can nevertheless be accommodated for testing in this facility, including during Phase II, if such designs are required by Demo.

5. Summary and discussion

In this paper we clarified how a mission to carry out fusion nuclear science R&D in the broad areas of plasma material interface and fusion power production will require plasma durations increasing from the ITER level (~10^3 s) progressively to 10^6 s. This can be obtained by substantially enhancing the availability and the plasma reliability of an integrated component testing facility. These in turn will require: 1) extensive modularized components and remote handling to improve the time-efficiency of the cycle of testing, discovery, understanding, improvements, and efficient replacement, and 2) sound strategies for disruption avoidance and mitigation through the use of conservative plasma conditions.

A set of design parameters for a spherical torus (spherical tokamak) device is refined from a previous assessment [2] to have a minimized R_0 (~1.2 m) while satisfying a set of conservative plasma and engineering conditions. This led to high q_{cyl} (≥3.7), moderate β_N (≤3.8), and modest density relative to the Greenwald density limit. By extending the divertor channel to $R_{div} \sim 2R_0$ via the SXD [10], the continuous and transient divertor heat fluxes could be reduced to below the ITER level.

Results from this work encourage the study of the following questions related to disruption avoidance: 1) What techniques are available for controlling the plasma conditions and profiles over very long duration beyond ~10^3 s in a fusion nuclear environment? 2) In what way does the probability of disruption depend on β_N, q_{cyl}, resonant error fields, and normalized density as the plasma parameters recedes from the stability limits? 3) What instabilities still remain, and what control techniques remain necessary?

In the area of remote handling, a number of R&D needs can be identified. The dexterous manipulation and precise positioning of heavy highly activated in-vessel components, both vertically and horizontally, is well beyond the present state-of-art, including the dose capabilities. Precise remote metrology system (laser ranging and mapping) needs to be developed to measure component and first wall alignment and erosion in the extreme fusion environment of radiation, baking temperatures, and high vacuum. Remote handling systems for the in-vessel components and to support the hot cell facility will also need to be developed.

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