DOE/SC-0149

Fusion Energy Sciences Advisory Committee

Report on

Opportunities for Fusion Materials Science and Technology Research Now and During the ITER Era

February 2012

U.S. Department of Energy
Office of Science

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Office of Fusion Energy Sciences
Germantown, Maryland 20874-1290

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March 6, 2012

Dr. William F. Brinkman
Director - Office of Science, SC-1
U.S. Department of Energy
1000 Independence Avenue, SW
Washington, DC 20585

Dear Dr. Brinkman,

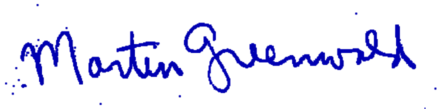
With this letter, the Fusion Energy Sciences Advisory Committee transmits two reports addressing your charge of July 22, 2011. The first, from the Panel on International Collaboration in Fusion Energy Sciences Research, responds to charge points 1 and 2. The second report, from the panel on Materials Science and Technology Research Opportunities responds to point 3. We want to thank Dr. Meade, Dr. Zinkle and members of both panels for their efforts.

As requested, the first report summarizes research opportunities on international facilities and reviews mechanisms and research modes best suited for fusion science. The panel has put the collaboration opportunities into a strategic context, defining a set of criteria for assessing scientific opportunities and recommending research areas with the greatest impact and net benefit to the U.S. To address the charge's second point, the panel carefully reviewed approaches taken in different fields of science and incorporated experiences already accumulated in the fusion program. Taken together, their recommendations point toward more consistent, systematic and effective processes for developing and executing impactful collaborative research.

Turning to the second report, the materials panel has conducted a thorough survey of the technical issues, identified a set of scientific grand challenges and proposed guiding principles and detailed research campaigns that would address the relevant science and technology. The report noted a critical role for computation, but did not detail the advances in applied math or computer science that would be required. A follow-on study on this subject could be timely and productive. The panel has produced a well-organized, comprehensive set of findings and recommendations, backed up by extensive detail and cogent argument. Aside from its principal aim, to inform DOE planning, we expect this report to be a valuable reference work for years to come.

Both studies generated wide-ranging and lively discussion at the recent FESAC meeting after which the committee voted unanimously to endorse each report.

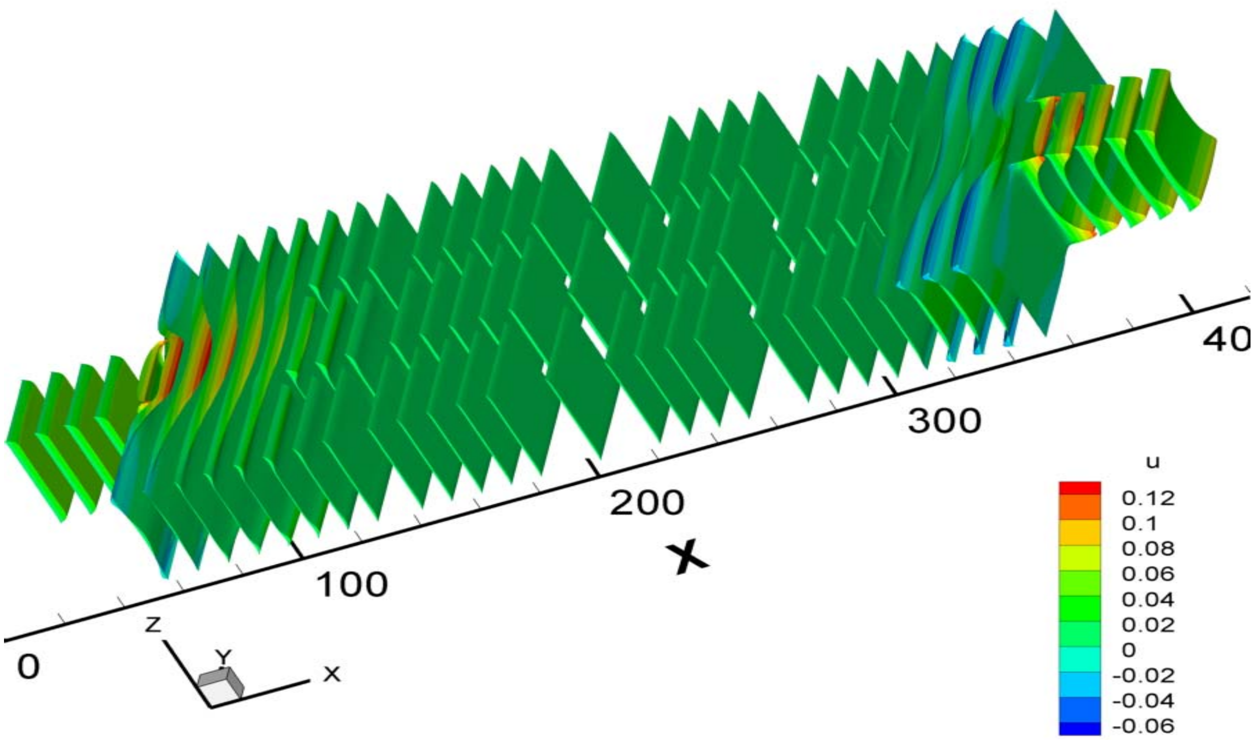
Sincerely,



Martin Greenwald
Chair, Fusion Energy Sciences Advisory Committee

Cc: Patricia Dehmer
Edmund Synakowski
Al Opdenaker

Opportunities for Fusion Materials Science and Technology Research Now and During the ITER Era



A Report to the Fusion Energy Sciences Advisory Committee
February 2012

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On the cover: 3D MHD simulation of flow distribution to 3 blanket channels from a common manifold. In the absence of a magnetic field (not shown), the flow is restricted to a central channel.

Acronyms

appm	atomic parts per million (fractional chemical concentration)
ASME	American Society of Mechanical Engineers
BCC	body centered cubic
BES	U.S. DOE Office of Basic Energy Sciences
BNL	Brookhaven National Laboratory
BT3F	Blanket thermomechanics thermofluid test facility
CERN	European Organization for Nuclear Research
DCLL	Dual coolant lead lithium
DEMO	demonstration fusion power plant
DOE	U.S. Department of Energy
dpa	displacements per atom
EM	electromagnetic
FCDF	Fuel Cycle Development Facility
FCI	Flow channel insert
FESAC	Fusion Energy Sciences Advisory Committee
FNSF	fusion nuclear science facility
FW	First wall
HEP	high energy physics
HFIR	High Flux Isotope Reactor
HPC	high performance computing
HTS	high temperature superconductor
IAEA	International atomic energy agency
IEA	International energy agency
IFMIF	international fusion materials irradiation facility
ISDC	ITER structural design criteria
ITER	originally acronym for International Thermonuclear Experimental Reactor
LBNL	Lawrence Berkeley National Laboratory
LHC	Large hadron collider
LTS	low temperature superconductor
MFE	magnetic fusion energy
MHD	magnetohydrodynamic
MIT	Massachusetts Institute of Technology
MTOR	MagnetoThermofluid Omnibus Research lab
MTS	materials test station (proposed neutron facility at Los Alamos)
NASA	National Aeronautics and Space Administration
NE	U.S. DOE Office of Nuclear Energy
NFA	nano-structured ferritic alloys
NGNP	Next Generation Nuclear Plant
NMR	nuclear magnetic resonance
NNSA	U.S. National Nuclear Security Administration
OFES	U.S. DOE Office of Fusion Energy Sciences
PFC	plasma facing component

PKA	primary knock-on atom
PMI	plasma-material interaction
R&D	research and development
RAFM	Reduced activation ferritic-martensitic (type of steel)
ReNeW	Research Needs Workshop (for MFE fusion energy sciences)
SINQ	Swiss Spallation Neutron Source
SNS	Spallation Neutron Source
SOL	Scrape off layer
TBM	test blanket module
TBR	Tritium breeding ratio
TBEF	Tritium Breeding and Extraction Facility
TRL	Technology readiness level
UTS	ultimate tensile strength
YBCO	yttrium-barium-copper-oxide

Executive Summary

The foundational goal of the fusion energy sciences program is to provide the science and engineering basis to recreate and control the power of the sun on earth. Achievement of this goal, which is ranked among the top few Grand Challenges for Engineering in the 21st Century in a listing organized by the US National Academy of Engineering, will require resolution of numerous materials science and engineering issues largely stemming from the extreme operating environment associated with a fusion reactor. This report summarizes the results from a six month study of fusion materials science and technology research opportunities. The evaluation was performed in response to a charge to FESAC in July, 2011 by the DOE Director of the Office of Science:

"What areas of research in materials science and technology provide compelling opportunities for U.S. researchers in the near term and in the ITER era? Please focus on research needed to fill gaps in order to create the basis for a DEMO and specify technical requirements in greater detail than provided in the MFE ReNeW (Research Needs Workshop) report. Also, your assessment of the risks associated with research paths with different degrees of experimental study vs. computation as a proxy to experiment will be of value."

The foundation for the evaluation was provided by considering recent evaluations on research opportunities [1, 2, 3] and by broadly soliciting research community input in order to develop a set of scientific grand challenges in the three topical themes that comprise fusion nuclear science: Taming the plasma-materials interface, Conquering nuclear degradation of materials and structures, and Harnessing fusion power (tritium science, chamber technology and power extraction). These scientific grand challenges constitute the key hurdles that must be resolved in order to establish the scientific proof of principle for fusion energy from a materials science and engineering perspective.

The scientific challenges associated with these three fusion nuclear science themes are extraordinary. For Plasma-materials interactions, the material surfaces directly in contact with the fusion plasma suffer extreme perturbations due to the continual energetic bombardment of plasma particles that both exhaust heat and "recycle" the hydrogen fuel. The surfaces are rapidly reconstituted and altered by this Plasma-Material Interaction; for example a surface atom may be removed and redeposited over a billion times in a single year. Simultaneously the surfaces impose strict boundary conditions for the fusion plasma, making for a highly non-linear, evolving coupled physical system. The neutron radiation damage to the materials and structures involve atomic- and meso-scale physical processes that span more than 20 orders of magnitude in time scale and over 8 orders of magnitude in length scale. The magnetohydrodynamic interactions of flowing liquid metal coolants in the strong and spatio-temporally complex magnetic field leads to highly non-linear 3D fluid physics. Similar to the plasma itself, these MHD effects in liquid metal coolants can exceed viscous and inertial forces by five or more orders of magnitude – dominating the flow behavior and heat transfer and thereby controlling the ultimate operating temperature, pressure, stress fields, and transport properties of the in-vessel systems. Finally, tritium must be handled at an unprecedented scale in fusion. Flow rates of many kilograms per day must be effectively processed over an incredible range of temperatures, pressures and material conditions (where vastly different chemical science mechanisms are operative), while observing stringent accountancy and environmental release constraints. A total of twelve scientific grand challenges were identified by the panel as a result of this evaluation. These grand challenges are summarized in the following chapter, and more complete descriptions are provided in Chapter 2.

Using the grand challenges as a scientific platform, most of the panel's work focused on potential research options to address these grand challenges. Quantification of the knowledge gaps and potential roles for different facilities was performed using Technology Readiness Levels (TRLs). Eighteen specific findings (summarized in the following chapter, and described in further detail in chapter 3) were identified by the panel. The three overarching findings are:

- There are inherent inefficiencies and costs associated with exploring multiple materials or concept options once the technological maturity has grown beyond the concept exploration stage (TRL 1-3). Thus research to explore the scientific proof of principle for fusion energy is most expediently accomplished by focusing research activities on the most technologically advanced option.
- Most existing US fusion technology test stands are no longer unique or world-leading. However, numerous compelling opportunities for high-impact fusion research may be achievable by making modifications to existing facilities and/or moderate investment in new medium-scale facilities.
- Computational modeling for fusion nuclear sciences is not yet sufficiently robust to enable truly predictive results to be obtained, but considerable reductions in risk, cost and schedule can be achieved by careful integration of experiment and modeling.

Building upon these findings and assessment of potential research options, a subset of particularly compelling US research opportunities were identified and seventeen recommendations were drawn (summarized in the following chapter and in more detail in chapter 4). The three overarching recommendations from this evaluation are:

- As fusion nuclear science matures from concept exploration studies (TRL 1-3) to more complex proof of principle studies (TRL 4-6), it is appropriate to focus R&D on front-runner concepts.
- Numerous fusion nuclear science feasibility issues can be effectively investigated during the next 5 to 10 years by efficient use of medium-scale facilities.
- The key mission of the next step US device should be to explore the integrated response of tritium fuel, materials and components in the extreme fusion environment in order to provide

the knowledge bases to contain, conquer, harness and sustain a thermonuclear burning DT plasma at high temperatures.

References

[1] M. Greenwald et al., "Priorities, Gaps and Opportunities: Towards A Long-Range Strategic Plan For Magnetic Fusion Energy," http://science.energy.gov/~/media/fes/fesac/pdf/2007/Fesac_planning_report.pdf (2007).

[2] R. Hazeltine (Chair), "Research needs for magnetic fusion energy sciences, Report of the Research needs Workshop (ReNeW)," June 8–12, 2009, DOE/OFES Report (2009). http://burningplasma.org/web/ReNeW/ReNeW.report.web2.pdf

[3] C. E. Kessel, M. S. Tillack, V. S. Chan, M. A. Abdou, L. R. Baylor, L. Bromberg, R. Kurtz, S. Milora, W. R. Meier, J. V. Minervini, N. B. Morley, F. Najmabadi, G. H. Neilson, R. E. Nygren, Y-K. M. Peng, D. Rej, R. D. Stambaugh, G. R. Tynan, D. G. Whyte, R. S. Willms, J. R. Wilson, B. Wirth, K. M. Young, Princeton Plasma Physics Report PPPL-4736, February, 2012

Summary of Grand Challenges, Findings and Recommendations

To address the charge, the panel followed three sequential steps (with some subsequent iteration): Firstly, in order to provide the appropriate scientific anchors for the analysis, twelve scientific grand challenges were identified that must be overcome to provide the scientific foundation for the development of practical fusion energy. Secondly, from this list of grand challenges, R&D options for resolving these issues were subsequently evaluated. As part of this evaluation step, a series of findings were formulated to summarize the current state of knowledge. As an important tool for evaluating and developing these R&D options, a series of charts were created that analyzed the potential roles of key facilities in advancing the scientific and technological knowledge that would be needed to successfully resolve the grand challenges. Finally, the information compiled in the R&D options was further analyzed to produce a limited number of compelling research opportunities for US researchers, along with a list of key recommendations.

The subcommittee organized its activities around three overarching fusion materials science and technology themes: Taming the plasma-materials interface, Conquering nuclear degradation of materials and structures, and Harnessing fusion power (tritium science and chamber technology). Although these three science themes are largely distinct, there are some overlapping R&D challenges. Within this report, overlapping challenges from the 3 main areas were not fully consolidated; repetition of certain R&D issues may be taken as an indication of the importance of the scientific issue at stake. The following provides a summary of the Grand Challenges, Findings and Recommendations that are described in more detail in Chapters 2, 3, and 4, respectively in the report.

Scientific Grand Challenges

In general terms, the overarching fusion nuclear science challenge may be stated as:

Overcome the multifold materials science and technology obstacles to establish the scientific foundation for the realization of practical fusion energy. There are five key corollary scientific challenges from a materials and engineering science research perspective:
A. Creation and control of hot dense burning plasmas; the key materials technology systems include heating/current drive systems, plasma diagnostics, and magnets.
B. Contain the plasma within material boundaries (and prevent undue contamination of the plasma) by understanding plasma-materials interactions (plasma and material operational limits) for both steady-state and disruption conditions.
C. Conquer fusion neutron-induced degradation of materials and structures by understanding how to design high-performance, self-healing material architectures.

D. Harness fusion power by extracting the energy released from the fusion reactions and converting it to practical electricity.
E. Sustain fusion power: produce, recover and recycle the tritium fuel needed to perpetuate the fusion reaction over a wide range of temperatures and multiple orders of magnitude in concentration.

From these five scientific topics, the panel focused the evaluation on three major themes that encompass the key fusion-relevant materials science and technology (fusion nuclear science) areas: Taming the Plasma-Materials Interface (topics A, B), Conquering Nuclear Degradation of Materials and Structures (Topics A, C), and Harnessing Fusion Power (Topics D, E). The three key scientific grand challenges compiled for the topical theme of Taming the Plasma-Materials Interface are:

CP1. Understand and mitigate the deleterious effects in plasma facing materials from both intense fusion neutron and plasma exposure that continuously damages the materials surrounding the plasma.

Relevant questions arising from these considerations include:

- Can plasma facing materials, for divertors, first walls, and RF launchers and mirrors be developed that have reasonable lifetimes when subjected to neutron produced volumetric defects and transmutation, and plasma induced surface and near-surface modifications?
- How does the coupling of intense heat flux, high temperature, and associated thermal gradients provide failure modes for plasma facing components?
- Can the basis for atomistic and meso-scale modeling be developed to support prediction of the long-term evolution of thermo-mechanical material properties of plasma facing materials, allowing mitigation of failures, and provide the capability to predict the lifetimes of plasma facing components?
- Can appropriate robust materials be developed that can be applied as armor or coatings in components that face the plasma, which include the first wall or structures such as RF launchers and mirrors that will require active cooling and have additional specialized functionality?

CP2. Understand, predict and manage the material erosion and migration that will occur in the month-to-year-long plasma durations required in FNSF/DEMO devices due to plasma-material interactions and scrape-off layer plasma processes.

Key questions here include:

- Can the boundary plasma and plasma-material interface be sufficiently manipulated to ensure that year-long erosion does not exceed the material thickness ~5-10 mm anywhere in the device?
- What combination of wall thermal conditions, plasma boundary and material choices can be used to control and/or removal redeposited material layers such that they do not perturb or disrupt the plasma through their failure and removal?

- What combination of wall thermal conditions, plasma boundary and material choices can be used to assure that the devices do not exceed operational safety limits for in-situ tritium retention and mobile dust?
- Can atomistic and continuum models be developed to predict the evolution of quasi-equilibrium microstructures and characteristics of plasma-affected surfaces?

CP3. Understand the coupled evolution of the plasma and PFCs under prototypical thermal, physical and chemical conditions expected in an FNSF/DEMO.

Questions arising from these issues include:

- What combination of experiments and modeling can be used to assess the highly temperature sensitive plasma-material physics phenomena that drive the coupling between the wall and plasma such as self-fueling of hydrogenic species from the wall through thermal releases, volatile impurity release and surface micro-structure evolution?
- How will the ambient and operating temperatures of a FNSF/DEMO affect the structure and fuel/tritium content of migrated and re-deposited material layers and particulates/dust?
- How will elevated temperature simultaneously affect the bulk PFC properties under both plasma and neutron bombardment? Will self-annealing effects be important for critical macroscopic quantities such as thermal conductivity and hydrogen trapping/diffusion?
- What tools to perform measurements in the plasma edge and in plasma facing components, *in-situ* and in *real-time* in a fusion environment, need to be developed to achieve significant understanding of the PFC issues?

The five scientific grand challenges compiled for the topical theme of Conquering Nuclear Degradation of Materials and Structures are:

CD1. *Develop a rigorous scientific understanding and devise mitigation strategies for deleterious microstructural evolution and property changes that occur in materials exposed to high neutron fluences and high concentrations of transmutation-produced gases from a 14 MeV peaked neutron source*.

Pertinent scientific questions to be explored include:

- Is there a practical limit to the maximum amount of accumulated transmutation-produced gases that can be tolerated, considering effects on deformation and fracture behavior, irradiation-induced swelling and creep, and high-temperature creep rupture lifetime?
- How do we extrapolate single-effect 14-MeV neutron degradation phenomena to the synergistic fusion nuclear degradation environment?

CD2. *Develop science-based design criteria that account for degradation of materials subjected to severe time-dependent, thermo-mechanical, high-temperature loadings, including the effects of 14-MeV neutron irradiation*.

Specific near-term questions that should be addressed include:

- Can physical models of high-temperature deformation be developed that account for thermo-mechanical property variations associated with processing conditions, and the impact of synergistic degradation modes (e.g., creep-fatigue)?
- How is high-temperature creep and creep-fatigue performance degraded by high-concentrations of helium?
- Can materials be created that simultaneously possess both high strength and high ductility or fracture resistance?
- What is the appropriate balance and integration of experimentation versus theory and simulation to qualify materials for service in the fusion environment?

CD3. *Comprehend and control the processes that drive tritium permeation, trapping, and retention in neutron radiation-damaged materials with microstructures designed to store large amounts of helium in numerous, nanometer-scale bubbles*.

Basic science questions include:

- How do radiation damage and helium gas generation impact tritium storage, retention and permeation in materials?
- Do materials development strategies that seek to manage radiation damage and helium production through embedded nanometer-scale precipitates lead to unsafe tritium retention?
- Will tritium retention levels saturate with continued radiation damage and transmutation?
- What is the effect of materials exposure history on tritium retention and permeation?
- Does the high partial pressure of tritium in lead-lithium breeding coolants require permeation barriers, and are practical barriers even possible in the fusion nuclear environment?
- What is the path to developing truly predictive system-level models of tritium inventory, including all pertinent permeation and retention mechanisms?

CD4. *Understand the fundamental mechanisms controlling chemical compatibility of materials exposed to coolants and/or breeders in strong temperature and electro-magnetic fields*.

Specific science challenges include:

- How do magneto-hydrodynamic (MHD) effects impact coolant behavior and corrosion?
- How do (electrically insulating) flow channel inserts respond to MHD instabilities?
- What are the effects of ionizing irradiation (radiolysis) on compatibility/ corrosion?

CD5. *Explore the potential opportunity for transformational advances in fabrication and joining technologies to provide high-performance materials with properties that enable construction of fusion power systems that fulfill safety, economic and environmental attractiveness goals*.

Specific scientific questions to be explored in the evaluation of advanced fabrication techniques include:

- How should complex, thermo-mechanically loaded fusion nuclear components be fabricated or joined to enable maximum performance?
- Is it possible to nondestructively evaluate the integrity and margin-to-failure of materials, components and structures in complex fusion blanket components?
- How can these components be repaired or replaced in a nuclear environment with minimal occupational exposure and waste generation?

The three scientific grand challenges compiled for the topical theme of Harnessing Fusion Power are:

CH1. Develop and validate a predictive capability for the highly non-linear thermo-fluid physics and the transport of tritium and corrosion products in tritium breeding and power extraction systems.

Key scientific questions to be explored include:

- Can we develop the needed material for these systems with a reasonable temperature window, lifetime, and thermo-physical properties?
- Can we simulate the 3-D MHD effects in flowing liquid breeders to the degree necessary to fully predict the temperature, temperature gradients and stress states of blanket components and materials?
- Can tritium be extracted from PbLi with the required high efficiency to limit tritium permeation below an acceptable level?
- Can the coolant chemistry be controlled to the level required to reduce deposition of activated corrosion products deposit on ex-vessel piping to an acceptable level?
- To what degree do impurities stemming from corrosion and transmutation impact high tritium extraction efficiency?
- Can we develop the needed practical material systems that meet reasonable temperature window, lifetime, and thermo-physical property requirements?

CH2. Understand physical phenomena that limit the life of the first wall and supporting structures in a fusion device.

Important questions to be answered include:

- Can we achieve the required heat transfer rate and flow stability in complex coolant channel geometries?
- What are the life-limiting phenomena in a gas-cooled first wall and can we engineer material properties and designs in order to extend the life of the first wall and structure?

CH3. Identify the rate-controlling chemical processes and devise practical solutions for handling unprecedented (kgs/day) levels of tritium over a vast range of temperatures, materials, ionization fields, and tritium concentrations.

Key questions to be answered include:

- Can fuel cycle processing time be shortened through better understanding of tritium chemistry, partial isotope separation, or processing modifications to reduce tritium inventory circulating through the fuel cycle system?
- What are the major processes by which tritium might be at risk; what tritium accountancy levels can be achieved?
- What techniques can be used that ensure steady state fusion burn operation with high tritium system availability?

Findings

Overarching Finding: There are inherent inefficiencies and costs associated with exploring multiple materials or concept options once the technological maturity has grown beyond the concept exploration stage (TRL 1-3). Thus research to explore the scientific proof of principle for fusion energy is most expediently accomplished by focusing research activities on the most technologically advanced option.

Overarching Finding: Most existing US fusion technology test stands are no longer unique or world-leading. However, numerous compelling opportunities for high-impact fusion research may be achievable by making modifications to existing facilities and/or moderate investment in new medium-scale facilities.

Overarching Finding: Computational modeling for fusion nuclear sciences is not yet sufficiently robust to enable truly predictive results to be obtained, but considerable reductions in risk, cost and schedule can be achieved by careful integration of experiment and modeling.

The four key findings reached for the topical theme of Taming the Plasma-Materials Interface are:

FP1. Power handling on the first wall, divertor, and special plasma facing components is challenging in steady state, and is severely aggravated by non-steady loading. Power handling at the plasma material interface must endure steady state, transient, and large off-normal heat fluxes, but at present, these fluxes cannot be predicted with sufficient accuracy. Efforts to eliminate or mitigate transient and off-normal loads are critical, with the ultimate solution being a compromise between loading conditions, plasma operating modes, material properties optimization, design solutions, and component lifetimes.

FP2. Materials suitable for plasma facing components (PFCs) are limited and their performance in the fusion environment is highly uncertain. The PFCs of a fusion device will experience the combined effects of plasma and nuclear loading and the material choices are severely limited. Important considerations are the impact on the core plasma via impurities, their response to plasma particle bombardment, their nuclear damage response, their thermal performance under high heat flux and operating temperatures above 500^{o}C, and their implications for safety and nuclear waste. The uncertainty in establishing PFC solutions is high, as the environment is severe and the requirements for long lifetime are challenging. Establishing material and design candidates will require significant efforts in experimentation and multi-scale simulation, and the coupling of plasma science, materials science, and advanced engineering and manufacturing technology.

FP3. Observing behavior at the plasma material interface during integrated month-long plasma operation requires capabilities beyond present day and planned facilities. Considerable knowledge will be established in present and planned tokamaks in terms of PFC/PMI related plasma science and engineering. However, the exposure to plasma durations ranging between several weeks to several months per year requires very high duty-cycle operations in a nuclear environment, e.g., in FNSF. During operations, the integrated plasma and material response will continuously evolve operations because of the continuing plasma-surface interactions. Predicting the long-term system behavior in light of this response requires some combination of non-nuclear month-long plasma exposures in PFC/PMI linear and confinement facilities and an extensive non-nuclear (or DD) phase of FNSF in order to alleviate risk to the nuclear (DT) phase of the FNSF. Comprehensive data on the PMI/PFC response are needed and require development and deployment of new diagnostics.

FP4. Developing measurement systems and the launching structures for plasma heating that can survive the fusion environment is a significant challenge. The required measurements for operation, control and safety, and those that can technically be applied in the anticipated FNSF environment are poorly understood. In addition, the heating and current drive systems, which face the plasma, are subject to the same severe exposure conditions as the first wall. Moreover, present experience shows these systems are generally sensitive to the scrape-off layer plasma, plasma material interactions, and have low reliability. A significant effort is required to establish viable materials, configurations, operating modes, and overall feasibility in the combined plasma and nuclear loading conditions expected in a FNSF.

The six findings for the topical theme of Conquering Nuclear Degradation of Materials and Structures are:

FD1. The lack of a fusion relevant neutron source for conducting accelerated single-variable experiments is the largest obstacle to achieving a rigorous scientific understanding and developing effective strategies for mitigating neutron-induced material degradation.

FD2. Identification of a prime candidate first-wall/blanket structural material is hindered by lack of an integrated engineering design and testing approach for materials development.

FD3. Knowledge of the processes controlling tritium permeation and trapping in advanced nanostructured alloys designed to manage high levels of helium is inadequate to ensure safe operation of next-step plasma devices.

FD4. Current understanding of the thermo-mechanical behavior and chemical compatibility of structural materials in the fusion environment is insufficient to enable successful design and construction of blankets for next-step plasma devices.

FD5. Transformational advances in fabrication and joining technologies may have the potential to provide high-performance materials with properties that enable construction of fusion power systems that fulfill safety, economic and environmental attractiveness goals.

FD6. The performance and economics of Magnetic Fusion Energy is significantly influenced by magnet technology. This provides a compelling motivation to continuously explore improvements in superconducting magnet capability (e.g. higher fields, higher tolerance to neutrons, lower manufacturing costs, etc.) as well as adapting the latest improvements in strand technology and new high temperature superconducting materials.

The four findings in the topical area of Harnessing Fusion Power are:

FH1. The ultimate attractiveness of a fusion system depends on the performance of power extraction and tritium breeding systems that surround the plasma. But, at present these systems are at a low technical readiness level with high uncertainty as to the performance of envisioned solutions and material systems.

FH2. The significant challenges in the ultimate development of in-vessel components that extract power and breed tritium for fusion energy are numerous and range from lack of fundamental data (single-effect experiments) to the need to develop validated, integrated simulation capabilities adequate for the design and analysis of nuclear systems. The efforts to meet these challenges are hampered by a lack of resources and test facilities.

FH3. The US has developed a potentially attractive family of blanket concepts, in which a lead-lithium eutectic alloy serves as both breeder and coolant. The reduced-activation ferritic steel integrated first wall and structure have separate gas cooling, and thermal- and electrical-insulating inserts based on silicon carbide control the structural material temperatures at critical interfaces.

FH4. Public acceptance and safety of fusion energy is strongly dependent upon the ability to reliably control the chemistry and permeation of tritium. Compared to fission reactors, fusion requires five orders of magnitude better control of tritium losses per unit of production. ITER represents a large step forward in the handling of DEMO-scale tritium flow rates, but ITER will not address removal and processing of tritium from candidate breeder blanket systems not will its tritium systems be available as test facilities to develop improvements still needed in processing time and system availability.

FH5. A fully integrated and coherent strategy to develop and utilize non-nuclear test facilities, fission reactors, fast neutron sources, and fusion devices to explore, understand and demonstrate the engineering feasibility of in-vessel materials and components does not currently exist in the US fusion program.

Recommendations

Overarching Recommendation: As fusion nuclear science matures from concept exploration studies (TRL 1-3) to more complex proof of principle studies (TRL 4-6), it is appropriate to focus R&D on front-runner concepts.

Overarching Recommendation: Numerous feasibility issues in fusion nuclear science can be effectively investigated during the next 5 to 10 years by efficient use of medium-scale facilities.

Overarching Recommendation: The key mission of the next step device beyond ITER should be to explore the integrated response of tritium fuel, materials and components in the extreme fusion environment in order to provide the knowledge bases to contain, conquer, harness and sustain a thermonuclear burning DT plasma at high temperatures.

The four recommendations for the topical theme of Taming the plasma-materials interface are:

RP1. Significant confinement plasma science initiatives are required to provide any confidence in the extrapolated steady and transient power loadings of material surfaces for a FNSF/DEMO. The mechanisms governing the steady-state perpendicular power width on open-field lines must be determined. Integrated plasma scenarios, and operating techniques must be developed that eliminate or mitigate transient heat loading from tokamak disruptions and intermittent edge instabilities such that surface damage to solid PFCs does not compromise their viability.

RP2. The leading FNSF/DEMO candidate solid material to meet the variety of PFC material requirements is tungsten due to its projected erosion resistance, high melting temperature and high thermal conductivity. Initiatives with the following objectives are required: 1) Identify and characterize suitable tungsten-based materials in appropriate plasma, thermal and radiation damage environments; and 2) Develop engineering solutions for tungsten PFCs with high-pressure helium gas coolant. The majority of PFC material research should be oriented towards tungsten, however due to open questions on tungsten melting and microstructural evolution, a parallel effort should be maintained in carbon-based solid materials with similar objectives.

RP3. Opportunities to access plasma pulse lengths in relevant exposure environments must be pursued in order to bridge the extremely large gap in pulse lengths between present experiments and FNSF/DEMO. Linear plasma devices, with appropriate upgrades from existing capabilities, can provide nearly unlimited plasma duration and should be advanced, in conjunction with modeling, to provide the first look at long time-scales at "as close as possible" exposure parameters. In confinement devices, where PMI and plasma effects such as material migration are integrated, a coherent strategy using the range of present short-pulse US tokamaks to the ~1000 s pulses available in international tokamaks is called for. It will also be necessary to assess and pursue a dedicated non-nuclear PMI/PFC facility, and/or utilize an early phase of FNSF operation with hydrogen or deuterium operation, to access the required days to weeks plasma duration in a relevant exposure environment. This will be required to provide the technical basis for, and reduce the risk of, a DT FNSF mission. Comprehensive data on the PMI/PFC response through diagnostic development and deployment will be simultaneously required.

RP4. A programmatic thrust is required to maintain substantial efforts in the areas of measurements (and their diagnostics) and heating/current drive systems that can survive the harsh fusion environment anticipated in a FNSF. The safe and effective operation/control of such a fusion device will rely on accurate measurements spanning both plasma and engineering systems. The heating and current drive systems, which face the plasma, are critical to sustaining the burning plasma for neutron generation. This should include materials selection to withstand high heat and particle flux, nuclear degradation, and erosion, while maintaining the required functionality of these complex components.

The four key Recommendations in the topical area of Conquering nuclear degradation of materials and structures are:

RD1. Re-engagement in the IFMIF Broader Approach Engineering Validation and Engineering Design Activity (EVEDA) should be initiated, in parallel with limited-scope neutron irradiation studies in upgraded existing spallation sources such as SINQ or SNS.

RD2. A detailed engineering design activity should be initiated that is closely integrated with materials research activities and is supplemented with limited neutron irradiation data (~10 to 20 dpa) from spallation neutron sources to permit selection of a prime candidate reduced activation steel for FNSF.

RD3. A robust experimental and theoretical effort should be initiated to resolve scientific questions associated with the permeation and trapping of hydrogen isotopes in neutron-irradiated materials with microstructures designed to mitigate transmutation produced helium.

RD4. Experimental and theoretical investigations to develop science-based high-temperature design criteria, and understanding of the fundamental mechanisms controlling chemical compatibility in the fusion environment should be significantly enhanced.

The four Recommendations in the topical area of Harnessing Fusion Power are:

RH1. A fully integrated strategy to advance the scientific and engineering basis for power extraction and tritium breeding systems should be established. Such a strategy should address this challenge through a mix of single- and multi-effect experiments in non-nuclear environments, fast neutron sources and fusion devices together with a comprehensive simulation capability.

RH2. Several key feasibility issues for the lead-lithium based blanket concepts should be examined as soon as possible in order to provide confidence in successful development of these concepts. These feasibility issues include tritium extraction from hot PbLi and He; liquid metal MHD effects on flow control and heat, tritium and mass transfer; chemistry control and compatibility of PbLi with, and thermomechanical loading of, ferritic steel structures and ceramic flow channel inserts.

RH3. The development of coupled models and predictive capabilities that can simulate time-varying temperature, mass transport, and mechanical response of blanket components and systems should be emphasized. These predictive capabilities should be validated against the experimental database; used to explore the coupling between disparate phenomena and loading conditions; and used to extrapolate beyond testing conditions to help guide and interpret further experimentation.

RH4. The performance and reliability of blanket and tritium extraction systems must be understood, demonstrated and made predictable with prototypic geometry, and in multi-material unit cells and mockups under combined loads where phenomena studied in separate effects tests can produce interactions that may lead to unanticipated synergistic effects. Planning for multiple-effect test facilities combining simulated thermal, mechanical, chemical, and electromagnetic conditions should begin in earnest as multi-effect experiments are essential prerequisites to any integrated testing program in fusion devices.

Chapter 1: Introduction and Background

1.1. Background and Discussion of Charge

Pursuit of viable long-term clean energy options continues to be a topic of considerable worldwide interest. In the area of fusion energy, construction activities are proceeding rapidly for the ITER plasma physics machine in southern France (in which the US is an active international partner). Whereas ITER is intended to address the most pressing remaining plasma physics issues for fusion (i.e., confinement and stability issues in the burning plasma regime), realization of the vision of practical, economical and environmentally sustainable fusion energy will be largely dependent on successfully resolving a series of daunting fusion nuclear science challenges that are primarily associated with materials science issues. It is within this context that FESAC has been charged to examine materials science and technology issues associated with establishing the scientific basis for fusion energy. Specifically, in a charge letter from the Director of the Office of Science dated July 22, 2011 (Appendix A), FESAC was requested to "elucidate the research needed to fill the gaps in materials science and technology required to sustain fusion plasma operations and to harness fusion power." The accompanying relevant portion of the charge letter is listed below:

"What areas of research in materials science and technology provide compelling opportunities for U.S. researchers in the near term and in the ITER era? Please focus on research needed to fill gaps in order to create the basis for a DEMO and specify technical requirements in greater detail than provided in the MFE ReNeW (Research Needs Workshop) report. Also, your assessment of the risks associated with research paths with different degrees of experimental study vs. computation as a proxy to experiment will be of value."

During follow-on discussions when the charge was presented to FESAC, it was noted that specific recommendations should indicate whether the research could be performed using existing facilities or if new facilities were required (domestic or international), and whether the compelling research opportunities were associated with near-, medium- or long-term (defined as 0-5 years, 5-15 years, and 15+ years from now) research activities.

1.2. Context of activity related to other recent planning exercises

The Office of Fusion Energy Sciences has been proactive in engaging the national fusion research community in a series of broad planning exercises over the past decade that have been useful in identifying a finite set of major obstacles to the successful development of fusion energy [1.1-1.8]. In addition, the science and engineering direction of the US fusion energy sciences program has been beneficially informed by numerous other scientific community evaluations [1.9-1.12]. The unique aspect of the current activity is the focus on potential compelling science issues associated with fusion materials science and technology (also termed fusion nuclear science) rather than plasma science issues that have been a major concentration in prior evaluations.

For the purposes of this evaluation activity, a fusion demonstration reactor (DEMO) as listed in the Charge was assumed to be a generic toroidal magnetic-confinement machine that would be designed using predominantly federal funding with the mission to demonstrate the potential for fusion energy to deliver large quantities of safe, reliable, cost-effective, and environmentally sustainable power. It was assumed that the U.S. fusion research program (working with international partners as appropriate) would identify the most attractive plasma confinement configuration for the DEMO during the coming decade.

1.3. Approach used to address the Charge

A subcommittee of 14 scientists from academia, national laboratories and industrial research organizations (Appendix B) with broad expertise in the topics of materials science and fusion nuclear science was assembled to evaluate research topics that might be of high significance for the successful development of nuclear power. Most of the subcommittee members were active participants in the ReNeW [1.8] workshop. This provided valuable background knowledge of the ReNeW topics and provided a useful launching point to explore compelling research opportunities for fusion nuclear science in greater detail than what was documented in the ReNeW report, as requested in the Charge statement. Prior fusion nuclear science evaluations, including the recently completed community-driven fusion nuclear science pathways assessment [1.12], were also valuable resources that were utilized as part of the assessment.

For the purposes of the subcommittee evaluations, the most relevant portions of the ReNeW workshop report were deemed to include the following:

Thrust 14: materials research
Thrust 7: high temperature superconductors and magnet innovations
Thrust 11: power handling (PFCs)
Thrust 12: plasma-material interactions
Thrust 13: fusion power blankets (power extraction, tritium)
Thrust 15: integrated designs for attractive fusion power

The meeting schedule and community interactions of the subcommittee for this evaluation is documented in Appendix C. Due to the relatively short (6 month) time period from the date the charge was delivered to FESAC and the requested delivery date of January 31, 2012, most of the panel business was conducted via teleconferences and email exchanges. A total of 15 conference calls and two face-to face meetings involving the full subcommittee were held, with the face-to face meetings occurring on Nov. 7-8, 2011 in Gaithersburg, MD and on Jan. 5-6, 2012 at the University of California-San Diego (UCSD).

In response to a request for community input, 21 contributed white papers and 5 email comments were received from community researchers during the late Fall of 2011. The community input was discussed at the UCSD face-to-face meeting in early January. A listing of the community input is provided in Appendix D. In addition to the contributed input, the subcommittee solicited three invited teleconference presentations to the full panel on the topics of fusion safety, technology readiness levels and their application to fusion energy sciences research, and Lessons learned from the NGNP fission reactor project as applicable for fusion energy research. All of the contributed and invited input was posted on the subcommittee website, https://aries.ucsd.edu/FESAC_MAT/.

1.4. Overview of the organization of the assessment

Overall, the field of fusion nuclear sciences is in an early stage of development. As such, while several concepts for fusion nuclear components have been identified, these concepts have different feasibility and/or performance issues as well as material requirements. For example, the tritium extraction method from the breeder depends on the type of tritium breeder used. Therefore, an evaluation of fusion research opportunities cannot be performed in isolation on individual issues. Instead, the evaluation should address the needs of specific concepts, exploring in a comprehensive manner the key feasibility issues of a concept in an integrated fashion to identify the research elements that are essential for a successful next step device. The need to utilize systems-level guidance as part of the R&D prioritization process is considered to be a natural and necessary step in order for fusion nuclear technology to progress from its current status of single-effects concept exploration research toward multiple-effect synergistic phenomena research needed to establish the proof of principle for fusion plasma-facing, blanket, and energy- and fuel-conversion concepts.

As part of the response to the charge (what is needed to fill the gap and the risks), the panel considered potential frameworks to

1) identify an integrated research program for each concept,
2) ensure that key knowledge gaps are satisfactorily addressed,
3) define decision points in narrowing down the options for each concept to make progress.

Such a framework has value not only in helping assess the risk associated with different research paths, but also to define formulations for risk/benefit analysis. In particular, the panel considered Technology Readiness Levels to be a useful framework (in conjunction with other prioritization metrics such as those outlined in the Greenwald FESAC report [1.7]). The Greenwald panel prioritization used three major criteria: 1) *Importance* for fusion energy and degree of extrapolation needed from current state of knowledge, *Urgency* (level of activity required now and in the near future; is the activity on the critical path for key fusion energy decision points?), and 3) *Generality*; applicability of the solution to multiple designs or DEMO concepts.

The subcommittee organized its activities around three overarching fusion materials science and technology themes, which encompass the topics associated with thrust areas 7, 11-15 in the ReNeW [1.8] workshop:

- Taming the plasma-materials interface
- Conquering nuclear degradation of materials and structures
- Harnessing fusion power (tritium science and chamber technology)

As described in further detail in Chapter 2, the key aspects of the scientific challenges are considerably different for these three themes. For plasma-facing materials, the challenge is associated with accommodation of extreme heat and particle fluxes along with intense fusion neutron damage. For material structures, the challenge is to maintain mechanical and structural integrity by designing highly efficient radiation self-healing nanostructures that are resistant to unprecedented levels of displacement damage and nuclear transmutation products. In order to harness fusion energy, it is mandatory to control the behavior of tritium in multiple materials over about ten orders of magnitude in concentration and several orders of magnitude in temperature, and identify practical approaches to simultaneously convert the sensible heat from the fusion reaction to useful power.

The subcommittee initially identified a limited number of high-level scientific challenges for each of the topical theme areas that must be successfully resolved in order to develop the scientific basis for fusion energy. These scientific grand challenges are discussed in Chapter 2. Following the identification of the grand challenges, the panel conducted numerous discussions via email and teleconferences on potential methods to identify the most expedient approach(es) to resolve these scientific challenges. A series of subcommittee findings that arose from these discussions, along with description of prospective research options and the proposed methodology used to evaluate their prioritization are outlined in Chapter 3. As part of this evaluation, a series of charts were constructed to identify potential roles of proposed major facilities. Finally, Chapter 4 summarizes a limited set of research opportunities that are considered to be particularly compelling, along with a series of recommendations by the subcommittee. Factors considered for determining compelling research opportunities included urgency and importance for fusion and the potential for US leadership in resolving the issue. The research opportunities included near-term (0-5 years), medium-term (5-15 years) and long-term (>15 years) activities.

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Chapter 2: Materials Science & Technology Grand Challenges on the Pathway to DEMO

2.1. Introduction

The foundational goal of the fusion energy sciences program is to provide the science and engineering basis to recreate and control the power of the sun on earth. With the dramatic improvements in scientific understanding of heating and confining dense plasmas over the past several decades (to be culminated in studies to be performed in ITER and complementary large scale plasma machines in the US and other countries), increasing attention is being placed on resolution of a series of high-level materials and fusion nuclear science feasibility issues that stand in the way of development of practical fusion energy. Building upon prior analyses by the fusion research community as documented in resources such as the Greenwald panel FESAC report and the ReNeW workshop report [2.1, 2.2], these fusion nuclear science feasibility issues require successful resolution of a variety of challenging scientific issues. In general terms, the overarching fusion nuclear science challenge may be stated as: Overcome the multifold materials science and technology obstacles to establish the scientific foundation for the realization of practical fusion energy.

There are five key corollary scientific challenges from a materials research perspective:

A. *Creation and control* of hot dense burning plasmas; the key materials technology systems include heating/current drive systems, plasma diagnostics, and magnets.

B. *Contain* the plasma within material boundaries (and prevent undue contamination of the plasma) by understanding plasma-materials interactions (plasma and material operational limits) for both steady-state and disruption conditions.

C. *Conquer* fusion neutron-induced degradation of materials and structures by understanding how to design high-performance, self-healing material architectures.

D. *Harness* fusion power by extracting the energy released from the fusion reactions and converting it to practical electricity.

E. *Sustain* fusion power: produce, recover and recycle the tritium fuel needed to perpetuate the fusion reaction over a wide range of temperatures and multiple orders of magnitude in concentration.

For the purposes of the subcommittee evaluation, these five Grand Challenges were grouped into three generalized materials science and technology themes: Taming the plasma-material interface (topics A and B above), Conquering nuclear degradation of materials and structures (topics A and C above), and Harnessing fusion power (Topics D and E above). These three materials and technology themes have emerged as recurring elements in several recent fusion community R&D assessment activities [2.1, 2.2] as well as in high-level summaries of the fusion energy sciences research challenges by OFES senior management. The specific scientific grand challenges identified by the panel for the three themes of plasma-materials interactions, nuclear degradation of materials, and harnessing fusion power are described in sections 2.2-2.4.

2.2. Grand Challenges associated with Taming the Plasma-Materials Interface

The pronounced science challenges associated with plasma-materials interactions include physical phenomena that occur over a vast range of length and time scales, as depicted in Fig. 2.2.1. The physical phenomena include processes that predominantly involve plasma edge physics, plasma facing materials, or interactive, coupled phenomena that encompass interactions involving both the plasma and the material.

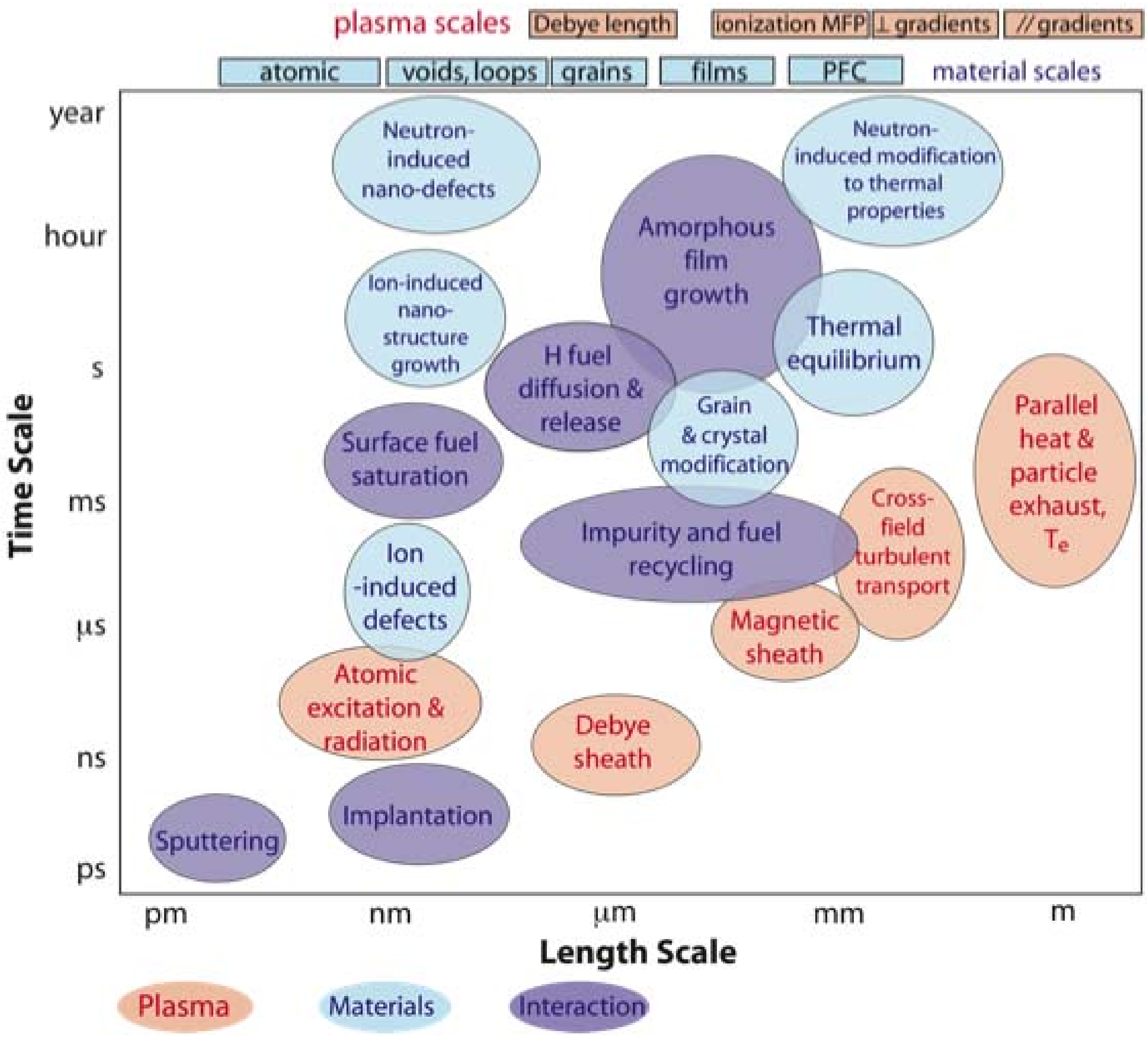

Figure 2.2.1. Overview of the coupled multi-scale phenomenon associated with plasma-materials interactions.

CP1 Understand and mitigate the deleterious effects in plasma facing materials from both intense fusion neutron and plasma exposure that continuously damages the materials surrounding the plasma.

Fusion neutrons produce volumetric defects and transmutation products, particularly helium and hydrogen. The plasma presents strongly perturbing physical processes at material surfaces, through erosion and re-deposition, and hydrogen and helium implantation. While these effects are largely separable due to the different scales, the intense heat flux and high material operating temperatures, and associated thermal gradients couple these multi-scale effects. Thermal loading can have steady, transient, and off-normal features that aggravate degradation mechanisms and can lead to failure. It is necessary to understand and predict failure modes of surfaces, bulk material, and material interfaces in this environment due to micro-cracking, thermal and mechanical fatigue and transient E-M loading, combined with radiation damage effects. Development of meso-scale and continuum modeling is required that explores failure modes and approaches to mitigate these failures, and ultimately the ability to predict the lifetime of plasma facing materials. The applications are armor, coatings and bulk materials used in components that face the plasma, and also include structures such as RF launchers and mirrors that will require active cooling and have specialized functionality. Relevant questions arising from these considerations include:

- Can plasma facing materials, for divertors, first walls, and RF launchers and mirrors be developed that have reasonable lifetimes when subjected to neutron produced volumetric defects and transmutation, and plasma induced surface and near-surface modifications?

- How does the coupling of intense heat flux, high temperature, and associated thermal gradients provide failure modes for plasma facing components?

- Can the basis for atomistic and meso-scale modeling be developed to support prediction of the long-term evolution of thermo-mechanical material properties of plasma facing materials, allowing mitigation of failures, and provide the capability to predict the lifetimes of plasma facing components?

- Can appropriate robust materials be developed that can be applied as armor or coatings in components that face the plasma, which include the first wall or structures such as RF launchers and mirrors that will require active cooling and have additional specialized functionality?

CP2 Understand, predict and manage the material erosion and migration that will occur in the month-to-year-long plasma durations required in FNSF/DEMO devices, due to plasma-material interactions and scrape-off layer plasma processes.

The boundary plasma continually rearranges plasma-facing materials through sputtering, plasma transport and redeposition. The modifications to materials, such as the erosion depth, become macroscopic for pulses of months-to-year duration. It will be necessary to reduce the impact of this material erosion and migration to an acceptable level for plasma performance and continuation. Particular issues of concern are the complete removal of PFC materials through net erosion, the reduced thermo-mechanical response of thick plasma deposits which may lead to perturbing macroscopic removal of materials that perturbs the plasma, the microstructural evolution of surfaces, tritium fuel retention and the production of dust. Achieving this understanding requires that we develop approaches to perform measurements in the plasma edge and in plasma facing components *in-situ* and in *real-time* in a relevant plasma environment. Questions here include:

- Can the boundary plasma and plasma-material interface be sufficiently manipulated to ensure that year-long erosion does not exceed the material thickness ~5-10 mm anywhere in the device?
- What combination of wall thermal conditions, plasma boundary and material choices can be used to control and/or removal redeposited material layers such that they do not perturb or disrupt the plasma through their failure and removal?
- What combination of wall thermal conditions, plasma boundary and material choices can be used to assure that the devices do not exceed operational safety limits for in-situ tritium retention and mobile dust?
- Can atomistic and continuum models be developed to predict the evolution of quasi-equilibrium microstructures and characteristics of plasma-affected surfaces?

CP3 Understand the coupled evolution of the plasma and PFCs under prototypical thermal, physical and chemical conditions expected in an FNSF/DEMO.

The unique environment presented by a fusion plasma vacuum in the form of particle and radiation fluxes, in combination with the high temperatures of plasma facing components anticipated in a FNSF and DEMO, presents an unconventional interface. The physical processes, like hydrogen chemistry, at the plasma material interface are strongly modified by these boundary conditions, as well as by the modification of the material surface itself by plasma exposure. The core plasma is affected by this interface via the SOL plasma, creating a coupled evolution that must be understood for the control of the burning plasma and the wide range of particles and debris (dust) in the vacuum. Self-consistent solutions are sought that provide high performance plasma operation, long PFC lifetime, and safe operation. Questions arising from these issues include:

- What combination of experiments and modeling can be used to assess the highly temperatures sensitive plasma-material physics issues that drive the coupling between the wall and plasma such as self-fueling of hydrogenic species from the wall through thermal releases, volatile impurity release and surface micro-structure evolution?
- How will the ambient and operating temperatures of a FNSF/DEMO affect the structure and fuel/tritium content of migrated and re-deposited material layers and particulates/dust?

- How will elevated temperature simultaneously affect the bulk PFC properties under both plasma and neutron bombardment? Will self-annealing effects be important for critical macroscopic quantities such as thermal conductivity and hydrogen trapping/diffusion?

- What tools to perform measurements in the plasma edge and in plasma facing components, *in-situ* and in *real-time* in a fusion environment, need to be developed to achieve significant understanding of the PFC issues?

2.3. Conquering Nuclear Degradation of Materials and Structures

Five key scientific challenges were identified for the topic of degradation of materials and structures in a fusion-relevant environment.

CD1 Develop a rigorous scientific understanding and devise mitigation strategies for deleterious microstructural evolution and property changes that occurs to materials exposed to high-neutron fluence, and high concentrations of transmutation produced gases from a 14 MeV peaked neutron source.

The fusion nuclear environment is extremely hostile for materials because they are continuously exposed to unprecedented heat fluxes, corrosive chemicals, large time-varying thermal and mechanical stresses and intense damaging neutron irradiation. Atomic displacement damage in a DEMO reactor corresponds to ejecting every atom from its lattice site more than 150 times. This enormous damage strongly interacts with reactive and insoluble gases produced by high-energy nuclear reactions to cause significant microstructural evolution. Consequently material properties gradually degrade with time through mechanisms such as low-temperature hardening and embrittlement, phase instabilities, solute segregation, precipitation, irradiation creep, volumetric swelling, and high-temperature helium embrittlement. These damage mechanisms are poorly understood because radiation damage is inherently a hierarchical, multi-scale phenomenon spanning many orders of magnitude in space and time as illustrated in Figure 2.3.1. The bubble diagram schematically shows how neutron damage creates atomic defects and transmutants at the shortest time and length scales, which evolve over long time through a multitude of transport and reaction sequences to produce significant changes in microstructure and macroscopic properties.

Recent workshops and planning exercises designed to identify knowledge gaps and research needs for development of a practical fusion energy systems have stressed the enormous materials challenges that must be overcome [2.1-2.5]. For example, the Greenwald report concluded that both materials and plasma facing components research are Tier 1 priority [2.1]. As defined by the Greenwald panel a Tier 1 priority indicates a situation in which “solution not in hand, major extrapolation from current state of knowledge, need for qualitative improvements and substantial development for both short and long term.” [2.1].

Fortunately, the extreme fusion nuclear environment, in which materials have to perform, share many common features with advanced fission designs, in that both advanced fission and fusion reactors seek to operate at higher temperatures and much higher radiation doses than existing fission reactors [2.4]. However, it is important to note that the fusion nuclear environment is even more extreme than for fission, as a result of severe thermo-mechanical loads, and the much harder neutron energy spectrum, which is peaked at ~14 MeV [2.4]. This high-energy neutron spectrum produces substantially larger nuclear transmutation reaction rates within first-wall/blanket materials. In particular, a significant goal of the worldwide fusion materials research is to quantify material degradation due to coupled displacement damage production and helium generation. In addition to copious production of gaseous hydrogen and helium, solid transmutation reactions may also significantly alter material composition resulting in considerable property degradation [2.4].

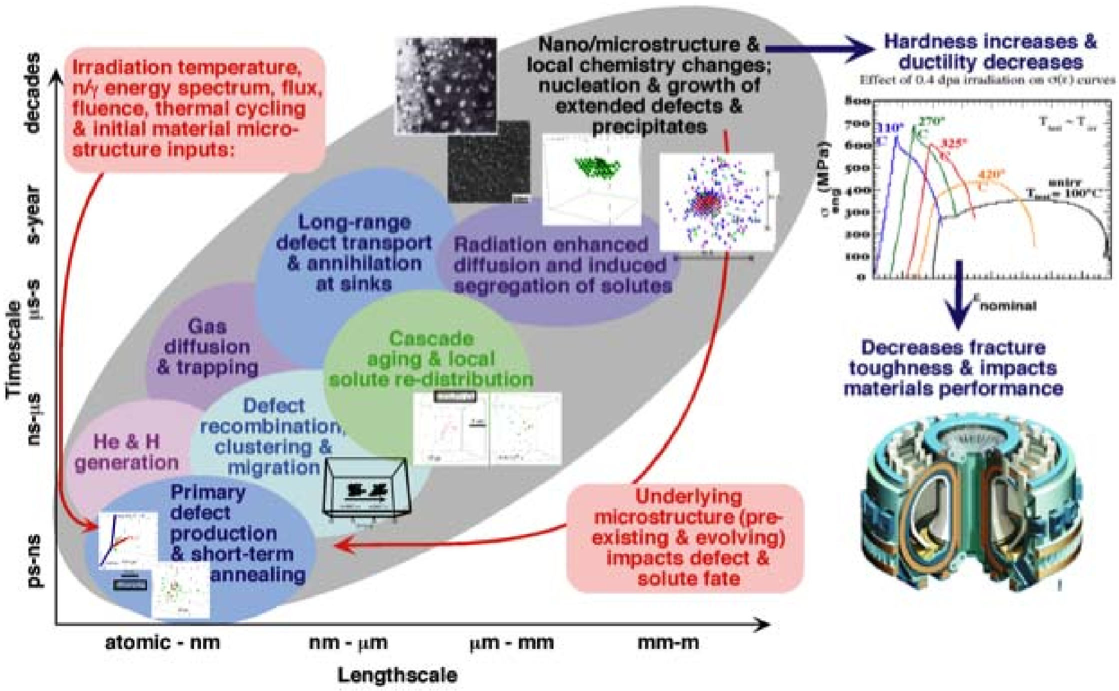

Figure 2.3.1. Radiation-induced microstructural evolution is a multi-scale phenomenon spanning enormous spatial and temporal regimes.

Managing the fusion neutron-induced property degradation is one of the most significant scientific grand challenges facing fusion power system development. As stated in the ReNeW report "Although considerable progress has been made exploring the resistance of structural materials to neutron irradiation in fission reactors (to damage levels on the order of ~30 dpa), the current knowledge base for reduced-activation structural materials exposed to fusion-relevant irradiation conditions is almost nonexistent" [2.2]. The ReNeW study also noted "The effects of helium on microstructural evolution have been investigated to near fusion-relevant levels in a limited set of materials systems in time-accelerated ion irradiation experiments, but these simulation experiments were not able to provide the bulk mechanical and physical property information needed by fusion power system designers" [2.2]. The potential impact of helium-induced degradation of reduced activation ferritic/martensitic steels over the entire postulated operating temperature range effectively illustrates the magnitude of the scientific challenge. Due to its low solubility, helium precipitates into gas bubbles, which can only grow by adding helium atoms [2.6]. At low irradiation temperatures, there is growing evidence that helium synergistically interacts with damage induced hardening resulting in severe degradation of fracture toughness and potentially causing intergranular fracture when the grain boundary fracture stress drops below the cleavage fracture stress [2.6, 2.7]. At intermediate irradiation temperatures helium bubbles may cause unstable void growth, possibly leading to unacceptable volumetric swelling and enhanced creep [2.6, 2.7]. At high irradiation temperatures, helium bubbles at grain boundaries can grow and coalesce under stress, resulting in severe degradation of creep and fatigue properties [2.6, 2.7]. This 14 MeV neutron-induced material degradation underscores the critical need for a fusion relevant neutron source enabling investigation of the effects of irradiation on bulk material mechanical and physical properties. Pertinent scientific questions the panel identified that could be explored in such a facility include:

- Is there a practical limit to the maximum amount of accumulated transmutation-produced gases that can be tolerated, considering effects on deformation and fracture behavior, irradiation-induced swelling and creep, and high-temperature creep rupture lifetime?

- How do we extrapolate single-effect 14-MeV neutron degradation phenomena to the synergistic fusion nuclear degradation environment?

CD2 Develop science-based design criteria that account for degradation of materials subjected to severe time-dependent, thermo-mechanical, high-temperature loadings, including the effects of 14-MeV neutron irradiation.

The ASME Boiler and Pressure Vessel Code [2.8] has been used for decades in the fission industry to design reliable, safe, economical reactors. However, it has limitations that will hinder the design and fabrication of high performance fusion components unless significant advances are made. Efforts to modify the existing codes to be suitable for fusion power plants are under way, but the emphasis has been on the addition of design criteria that address fusion-specific loads and failure modes, rather than taking a fundamentally different approach. This is exemplified by the ITER Structural Design Criteria (ISDC) [2.11, 2.12], which follows the ASME Code, but adds rules for phenomena such as flow localization and creep ratcheting [2.12, 2.13]. Despite these additions, there is still substantial room for improvement in the design approaches suggested by these Codes.

The existing design rules are limited by their overall philosophy. In some cases, such as ratcheting, the current design rules are simplified to permit conservative design without sophisticated analysis. In other cases, such as creep rupture, empirical rules are employed because there is no adequate scientific basis for anything more comprehensive. In this latter case, there is no guarantee that these rules are conservative, especially if the operating conditions are far removed from the conditions under which the rules were developed. Hence, a move towards science-based design rules can have a major impact on the field.

One example where substantial progress can be made is creep-fatigue interaction, which is material damage resulting from a combination of cyclic loading and creep deformation. The design codes address this in an empirical fashion, assuming a correlation for the component lifetime and then relying on substantial data to determine the limits [2.13]. Since there is only a minor mechanistic basis to the assumed correlation, we cannot extrapolate the data to other temperatures, materials, or loading conditions. A scientific, multi-scale approach to this problem, with substantial contributions from both the materials and design communities, will permit the design of higher performance components with reduced testing and higher confidence. This would be accomplished through the consideration of the interaction between fatigue crack propagation and grain boundary cavitation caused by creep, thus addressing the actual failure mechanism from first principles, rather than just providing a correlation backed by a large experimental database. By building this level of mechanistic understanding into the design process, our ability to address unexpected events and recognize unexpected deformation processes would be greatly enhanced.

Other weaknesses inherent in the existing design rules include:

- Limited understanding of the role of large quantities of helium generation on failure modes expected in these devices;

- Incomplete treatment of the higher temperatures and large temperature gradients (especially during transient events) expected in fusion devices;

- Limited understanding of appropriate design rules for materials of limited ductility (tungsten, steels irradiated at low temperatures, etc.);

- Limited understanding of creep rupture in a high temperature fusion environment;

- Minimal understanding of the interaction of creep and fatigue in a fusion environment (from the perspectives of material data availability, failure criteria, and design rules);

- Limited data for materials of interest, especially in a 14 MeV neutron environment (including fatigue crack growth rates, creep rupture, fracture toughness of as-manufactured and repaired materials, etc.).

These deficiencies must be addressed as design rules are formulated for fusion components. This effort will require sophisticated modeling, both atomistic and continuum, to identify key failure mechanisms and develop design rules that will ensure the reliable operation of these devices. The effort will also require substantial experimentation to validate the new design rules. Some of these experiments can be carried out at a coupon level, but others will require larger tests to study synergistic effects. Still others will require a 14 MeV neutron source to study irradiation effects, preferably in a component-level experiment.

The most prominent of the efforts to update traditional design rules to account for fusion-specific phenomena is the ITER Structural Design Criteria (ISDC) [2.11]. These criteria are similar to the ASME design rules, addressing failure criteria such as plastic instability. To address fusion specific issues, these criteria also address flow localization and exhaustion of ductility [2.12], which are characteristic of materials with limited ductility. The ISDC also updates the high temperature design rules, considering phenomena such as creep ratcheting. These criteria represent a significant advance in design rules for fusion reactors, but substantial work must be carried out to validate and improve these efforts.

In addition to validating the proposed criteria in the ISDC, there are efforts to establish the level of conservatism inherent in these rules (despite recent updates) by comparing the performance of reactor components designed by the existing rules to those designed purely by analysis. A design by analysis approach requires a more sophisticated analysis, such as a thorough, 3-dimensional, transient model, but does not require the conservative approximations of the simplified design rules and can lead to reduced design margins. For example, a recent paper addressing ratcheting, a failure mode addressed in both the ASME Code and the ISDC, predicted a 22% increase in design margin when using design by analysis, as opposed to the typical code-based design rules [2.13].

In order to begin to advance the state of the art in fusion component design, the panel identified the following list of questions that should be addressed in the next five years:

- Can physical models of high-temperature deformation be developed that account for thermo-mechanical property variations associated with processing conditions, and the impact of synergistic degradation modes (e.g., creep-fatigue)?

- How is high-temperature creep and creep-fatigue performance degraded by high-concentrations of helium?

- Can materials be created that simultaneously possess both high strength and high ductility or fracture resistance?

- What is the appropriate balance and integration of experimentation versus theory and simulation to qualify materials for service in the fusion environment?

CD3 Comprehend and control the processes that drive tritium permeation, trapping, and retention in neutron radiation-damaged materials with microstructures designed to store large amounts of helium in numerous, nanometer-scale bubbles.

Controlling the flow of tritium will be an enormous technical challenge in fusion energy systems because of the very large inventory of tritium involved, and the potential for significant retention in irradiated materials. The basic mechanisms of tritium adsorption and absorption at surfaces, diffusion kinetics in irradiated metals and ceramics, and the interaction with microstructural features such voids, helium bubbles, and defect clusters are poorly understood. To illustrate the impact of neutron irradiation, a dose of as little as 20 dpa in steels produces a trap site density that could quadruple retained tritium in the blanket. All previous planning studies emphasized the need for increased research to better understand tritium transport and retention mechanisms [2.1- 2.5].

Controlling tritium permeation is essential to minimize accumulation of tritium in certain areas of a fusion power system. The latest planning study [2.5] highlighted the need for tritium permeation barrier development. Permeation barriers will be required to contain tritium in a number of high temperature and/or high tritium concentration regions of the plant [2.14]. Such barriers may also mitigate corrosion or provide electrical insulation. It is important for these barriers to demonstrate long lifetime in their particular environments, which can include irradiation, high temperatures, thermal cycling, and corrosion. The panel also recognized the potential for enhanced trapping and retention of diffusing tritium due to defect production associated with neutron irradiation. This issue has received little attention and could have significant safety implications.

The panel observed that recent research to develop microstructures resistant to displacement damage and transmutation produced helium may significantly enhance tritium retention. Of particular concern are advanced materials known as nano-structured ferritic alloys (NFAs). The microstructure of NFAs is comprised of an ultra high density of nanometer-scale oxide particles, which impart excellent high-temperature creep strength, radiation damage resistance, and provide numerous sites to benignly trap helium and protect grain boundaries. When NFAs are exposed to 14 MeV neutron irradiation a large number of helium bubbles will be produced, but because the bubble density is high, their sizes remain below a critical diameter, preventing significant dimensional change of the NFA. Thus, the NFA microstructure offers considerable benefits for managing helium and mitigating displacement damage, but tritium retention may be increased.

There is growing scientific evidence that retention and accumulation of tritium in irradiated materials is controlled by the concentration of microstructural trapping sites. The most effective tritium trapping sites appear to be helium bubbles, voids, and the displacement fields of defect clusters [2.15]. Triple ion beam irradiation studies on iron-chromium model alloys [2.16] and vanadium alloys [2.17] indicate that the largest dimensional changes are observed when helium and hydrogen are co-implanted with displacement damage. A recent study [2.18] involving an austenitic stainless steel correlated deuterium retention with the level of implanted helium, but the correlation was weak at high-temperatures, suggesting there are several fundamental issues associated with tritium permeation and retention that require basic science investigations to resolve. The panel noted several basic science questions, delineated below, that should be addressed in the next five years, including:

- How do radiation damage and helium gas generation impact tritium storage, retention and permeation in materials?

- Do materials development strategies that seek to manage radiation damage and helium production through embedded nanometer-scale precipitates lead to unsafe tritium retention?

- Will tritium retention levels saturate with continued radiation damage and transmutation?

- What is the effect of materials exposure history on tritium retention and permeation?

- Does the high partial pressure of tritium in lead-lithium breeding coolants require permeation barriers, and are practical barriers even possible in the fusion nuclear environment?

- What is the path to developing truly predictive system-level models of tritium inventory, including all pertinent permeation and retention mechanisms?

CD4 Understand the fundamental mechanisms controlling chemical compatibility of materials exposed to coolants and/or breeders in strong temperature and electro-magnetic fields.

The current scientific understanding of corrosion mechanisms at high temperatures relevant for fusion energy applications is established only for relatively simple systems such as pure elements or relatively simple alloys in air or steam. The materials to be used for construction of fusion energy systems will be composed of complex multi-component phases and the operating environment will involve high levels of ionizing radiation that may exert unknown influences on the nucleation and growth of semiconducting or insulating corrosion product surface films. In addition, the effect of high magnetic fields on corrosion mechanisms (particularly for liquid metal coolants) is not understood. Furthermore, the root mechanisms responsible for stress corrosion cracking are not yet well understood even in less demanding operational environments such as chemical processing or fossil energy systems. Irradiation-assisted stress corrosion cracking has emerged as a vexing and costly issue in commercial fission power plants; an improved fundamental understanding of the molecular-level processes that control the kinetics of corrosion would be important not only for fusion energy systems but also for broader applications including fossil and nuclear energy systems.

In general, corrosion is controlled by either thermodynamic driving forces (free energy changes) or kinetic processes (e.g., diffusion-limited migration of an atomic species). Most of the relevant thermodynamic data for potential reaction products are known, although it is possible local variations in the atomic concentrations of species at the corrosion interface may occur due to unique conditions associated with the fusion energy environment (EM fields, etc.). The larger area of uncertainty for fusion applications is associated with poor current understanding of kinetic processes associated with chemical compatibility. Corrosion reactions may be controlled by diffusion of simple or complex species to the reaction interface from either a solid or a nonsolid (liquid or gaseous) material such as a coolant. The atomic diffusion mechanisms in the coolant depend on the dimensions of boundary layers adjacent to the solid surface; coolant velocity and turbulence can introduce marked effects. A particular near-term scientific challenge is to improve our understanding of fundamental corrosion mechanisms for ferritic steels in contact with flowing Pb-Li, including the roles of temperature, temperature gradient, coolant velocity and EM fields on the corrosion kinetics. Specific science challenges include:

- How do magneto-hydrodynamic (MHD) effects impact coolant behavior and corrosion?
- How do electrically insulating flow channel inserts respond to MHD instabilities?
- What are the effects of ionizing irradiation (radiolysis) on compatibility/corrosion?

CD5 Evaluate the potential for transformational advances in fabrication and joining technologies to provide high-performance materials with properties that enable construction of fusion power systems that fulfill safety, economic and environmental attractiveness goals.

The thermomechanical and radiation environment for structural materials in the first wall and blanket regions of DT fusion energy systems represent an extraordinary challenging environment. Whereas structural materials in other advanced technologies have been required to reliably perform under a variety of extreme conditions, the combination of high operating temperatures, cyclic thermomechanical loads, intense heat fluxes approaching ~10% of the value at the surface of the sun, and unprecedented energetic neutron irradiation exposures will require the development of new innovative materials and manufacturing methods. The two most challenging issues for structural materials in fusion energy systems involve accommodation of intense heat fluxes and maintaining mechanical and structural integrity during exposure to high doses of energetic neutrons. In some cases, further advances in high heat flux capability are limited by the inability to precisely fabricate intricate cooling channel geometries into components using conventional machining techniques.

There has been a recent explosion of research on advanced manufacturing techniques to enable near net shape fabrication and reconstruction of intricate geometries, including ultrasonic additive manufacturing, electron beam assisted deposition, laser assisted deposition, and fused deposition modeling. These innovations are enabling the possibility to fabricate complex geometries that could not be produced by conventional manufacturing methods, at potentially significantly lower cost. However, several of the methods utilize melting of the constituent material in the deposition process, which may destroy some of the nanoscale features currently being explored as the mainstream approach to designing radiation resistance. Hence, there may be a potential systems-level optimization tradeoff between designing high radiation resistance in the base material (but difficult to fabricate into intricate shapes needed for the high heat flux challenge) and designing for high heat transfer capability by using emerging advanced manufacturing methods (but potentially impaired radiation resistance). In specific locations of a component where high heat flux tolerance is mandatory and significant degradation of mechanical properties or dimensional stability can be tolerated, additive manufacturing may be appropriate. In other locations on the same component where stability to neutron irradiation is most important and a reduced heat flux capability is acceptable, conventional fabrication of radiation–resistant bulk materials may be preferred. Specific scientific questions to be explored in the evaluation of advanced fabrication techniques include:

- How should complex, thermo-mechanically loaded fusion nuclear components be fabricated or joined to enable maximum performance?

- Is it possible to nondestructively evaluate the integrity and margin-to-failure of materials, components and structures in complex fusion blanket components?

- How can these components be repaired or replaced in a nuclear environment with minimal occupational exposure and waste generation?

2.4 Harnessing Fusion Power

Fusion power is captured in an integrated first-wall/blanket system that surrounds the plasma. This system has to operate at a high temperature to efficiently convert fusion power into electricity or other possible end uses. Furthermore, tritium fuel has to be bred by capturing fusion neutrons in lithium. In addition to the integrated first wall/blanket, systems associated with harnessing fusion power include shielding of various components (e.g., superconducting magnets), heat transport loop and heat exchangers, recovery of bred tritium from the blanket, etc.

Safety and environmental features of a fusion power plant are mainly determined by materials and design of systems associated with harnessing fusion power. "Power extraction and tritium sustainability" was identified as a key area in both the Greenwald Report [2.1] and the subsequent ReNeW [2.2] community workshop.

The integrated first wall/blanket is subjected to the extreme conditions including the intense flux of energetic fusion neutrons that changes material properties through production of tritium, helium and other gases, atomic displacements, and many transmutation products. In addition, the first wall is subjected to intense fluxes of particles and x-rays. Because of its unique operating environment, harnessing fusion power systems represent a complex area involving subtle combinations of many scientific disciplines such as materials, thermo-mechanics, thermo-fluids, magnetohydrodynamics, chemistry, etc. Furthermore, fusion systems will operate in conditions that have not been encountered before. As such, not only does the scientific basis of many disciplines need to be extended to fusion conditions, it is expected that many synergetic effects would be discovered.

Harnessing fusion power is a complex topic involving interactions among many combinations of scientific disciplines. A multitude of fusion systems will be required to perform at unprecedented conditions. Figure 2.4.1 is a depiction of the range of conditions over which Harnessing Fusion Power systems will operate, quantifying the fusion plasma and the chamber technology environment in terms of tritium, neutrons, temperature and magnetic field. Within a fusion energy system these parameters will vary by 16 (for concentration), 17, 5 and 5 orders of magnitude, respectively, and span different regimes of physics and chemistry predominance. A broad and fundamental science-based approach will clearly be necessary to meet this challenge. Figure 2.4.1 shows some of the diverse science disciplines associated with the various systems needed to harness fusion power. Not only will these individual science topics need to be understood, but it is also likely that unanticipated synergistic effects will be discovered and, as necessary, controlled. A staged set of experiments performed in concert with computational modeling will be essential for success.

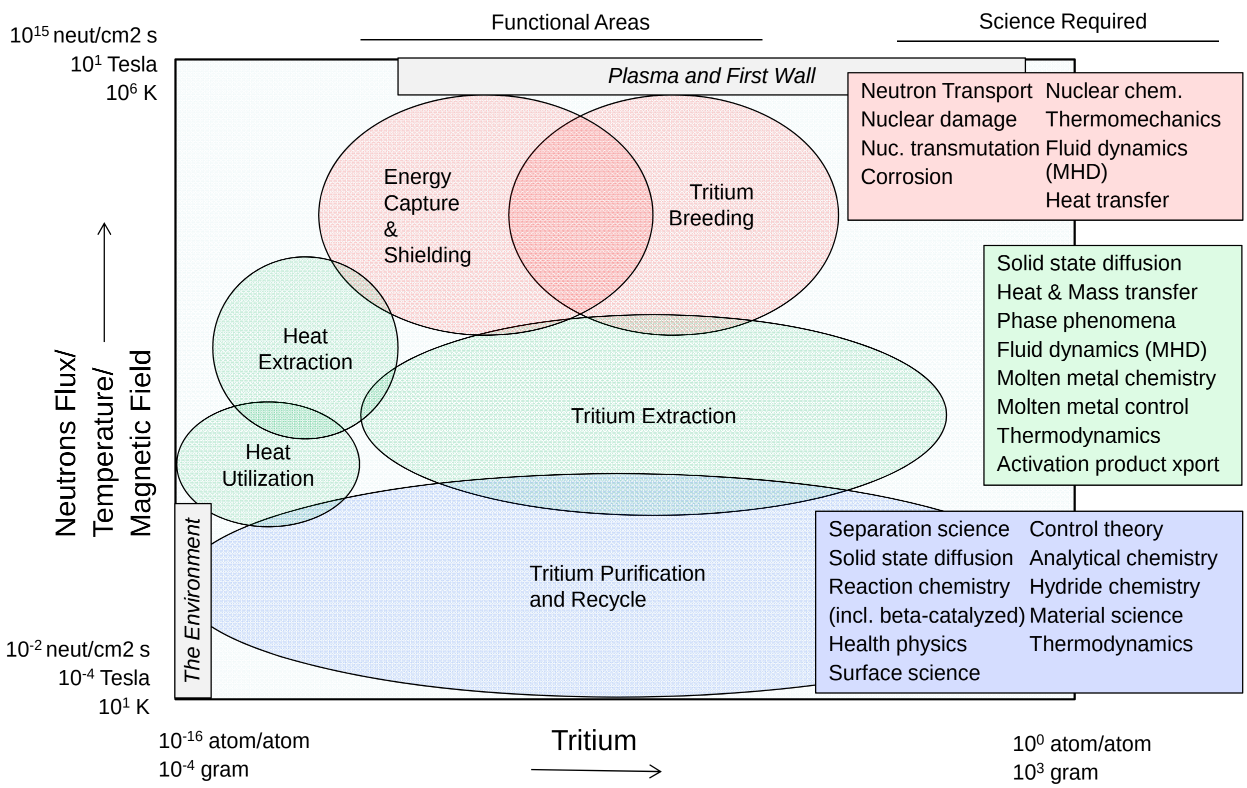

Figure 2.4.1: Science associated with the Harnessing Fusion Power functions plotted against the vast range of tritium, neutron, temperature and magnetic field parameter space extending between a fusion plasma and the environment

CH1 Develop and validate a predictive capability for the highly non-linear thermo-fluid physics and the transport of tritium and corrosion products in tritium breeding and power extraction systems.

A potentially attractive family of power extraction and tritium breeding blanket systems results from the use lithium-bearing, liquid metal alloys such as lead-lithium eutectic alloy as both a fluid for extracting fusion heat and for producing and transporting tritium. Over the past decade, the US program has focused its attention on a blanket concept called the Dual Coolant Lead Lithium system. The DCLL has been developed by power plant studies program and studied as a candidate for by US ITER-TBM [2.19-2.21]. The DCLL uses flowing PbLi as both breeder and coolant for the breeding zones, while utilizing high pressure helium to cool all structures including those surrounding the breeding zone. Flow channel inserts made of a SiC-composite placed in all liquid metal ducts serve as electrical and thermal insulator, enabling a liquid metal exit temperature about 200K higher than the maximum temperature of the steel structure. By this method the thermal efficiency in the power conversion system can approach 45%, compared to values of ~40% for entirely He-cooled blankets.

Flowing PbLi will experience highly non-linear, 3-D magnetohydrodynamic (MHD) forces and effects from electric currents induced by the motion of the coolant in a magnetic field. These MHD forces can exceed typical viscous and inertial forces by five or more orders of magnitude – dominating the liquid metal flow behavior, pressure, stability and heat transfer everywhere inside the toroidal field coils. These processes then ultimately control blanket operating temperature, stress fields, and transport properties in the blanket systems. To realize the benefits of utilizing liquid metals as breeders and coolants, an in depth understanding of and predictive capability for liquid metal MHD thermo-fluid flows are essential. There is a particularly interesting coupling of MHD effects in the PbLi with the performance of the SiC flow channel insert intended to electrically decouple the flowing PbLi from the conducting structural walls. Requirements on the FCI thermophysical properties and crack tolerances will be largely determined by the MHD requirements and effects [2.22].

The use of helium gas cooling for the DCLL first wall and structure has the distinct advantage that the PbLi need only serve as a coolant for volumetric heat deposited in the PbLi flow itself. The PbLi does not need to flow fast in order to cool the structures, especially the first wall, and the MHD impact is somewhat reduced. But the use of helium then requires development of flow distribution and control in very complex and interconnected cooling channels embedded in all blanket structures. In addition, the permeation of tritium from the breeder into the coolant must also be kept to minimum levels. Both of these considerations require new understanding of flow and heat transfer properties in helium and tritium transport properties in PbLi, SiC and ferritic steel structures.

Both the heat captured and the tritium produced in the blanket must be continuously extracted by ex-vessel systems and transferred to the power cycle and tritium fuel cycle systems. In addition to the high operating temperature of the coolant, issues associated with tritium permeation, corrosion and transport of activated material, coolant chemistry control, etc. necessities new coolant processing and heat exchanger technologies and materials. Developing the scientific basis of heat, mass and tritium transport physics, high temperature material compatibility and radioactive coolant chemistry in a fusion environment is essential. [2.23].

Many questions require single- and/or multiple-effect controlled experiments as well as validated modeling capabilities to effectively answer:

- Can we develop the needed material for this system with a reasonable temperature window, life time, and thermo-physical properties?

- Can we simulate the 3-D MHD effects in flowing liquid breeders to the degree necessary to fully predict the temperature, temperature gradients and stress states of blanket components and materials?
- Can tritium be extracted from PbLi with the required high efficiency to limit tritium permeation below an acceptable level?
- Can the coolant chemistry be controlled to the level required to reduce deposition of activated corrosion products deposit on ex-vessel piping to an acceptable level?
- To what degree impurities stemming from corrosion and transmutation impact high tritium extraction efficiency?

CH2 Understand physical phenomena that limit the life of the first wall and supporting structures in a fusion device.

The first material wall facing the plasma is a critical and integral part of blanket components that surround the plasma to harness the fusion power. The first wall has unique loading conditions and damage processes associated with its operating environment consisting of intense fluxes of high-energy neutrons, plasma particles and heat, and strong electromechanical and thermomechanical forces. Especially important is the coupling of the first wall and blanket structure to off-normal plasma events such as current disruptions. Such events may be unavoidable in a practical fusion energy systems.

The first wall also has many restrictive requirements including: (a) integration into the blanket, requiring low structural content and selected materials that do not overly inhibit tritium breeding potential; (b) high operating temperature so the recovered energy can be efficiently converted to electricity; and (c) extremely low failure tolerance with difficult access and replacement opportunities with no possibility of redundancy. In particular the first two requirements further distinguish the first wall from the divertor, where more flexibility in the structure, geometry, and choice of materials may exist.

The complex heat-transfer problem in a gas-cooled first wall is exacerbated by coolant channel complexity and the need for a relatively thin first wall that can withstand transient events. Many physical phenomena can compete to limit the life of the first wall and it is certain that fundamental material science understanding of material degradation described in Section 2.3 is an essential prerequisite. But component and system level understanding is also required. Large temperature gradients and associated stress and strain, thermal and irradiation creep, fatigue and system wide mass transport, and many other phenomena can all build towards non-linear synergistic effects. These issues for such complex material systems operating at a high temperature are poorly understood in the fusion environment.

Important questions that must be answered include:

- Can we achieve the estimated heat transfer rate and flow stability in complex coolant channel geometries?

- What are the life-limiting phenomena in a gas-cooled first wall, and can we engineer material properties and designs in order to extend the life of the first wall?

CH3 Identify the rate-controlling chemical processes and devise practical solutions for handling unprecedented (kgs/day) levels of tritium over a vast range of temperatures, materials, ionization fields, and tritium concentrations:

Plasma must be continuously fueled with tritium to sustain fusion reactions, but only a fraction of this tritium burns while the rest remains is exhausted from the plasma chamber. This exhaust must be processed to recover the tritium from the helium ash and other impurities so that the tritium can be re-injected into the plasma. The flow rate of tritium in this stream can be as much as 5-10 kg/day. For comparison, this is many times what a current fission reactors can produce in a year. In addition, as previously described, to replace the burned fraction of tritium fuel, tritium should be produced in and extracted from a breeding medium in the blanket that surrounds the plasma. This tritium must be continuously and efficiently extracted. Batch processing as used for tritium production in fission reactors is simply not workable given the need to breed and extract up to 0.5 kg/day from the blanket system.

For safety reasons, it is essential that tritium be effectively contained to protect workers, the public and the environment. Tritium is among the most mobile of elements and can readily permeate through metallic structures, especially those at elevated temperatures with large surface areas, such as those in the heat transport system. Expected tritium release limits for fusion are extremely low, in comparison. Add to this the fact that tritium is also a sensitive material in regards to proliferation. Accountancy of tritium for both safety and proliferation, given that kilogram quantities of tritium are circulating daily through the system, is another aspect of this challenge.

This grand challenge requires scientific understanding of many interconnected phenomena and material systems such as permeation, radiolytic chemistry, chemical kinetics, surface science, liquid metal magnetohydrodynamics, vapor-liquid phase behavior, and mass transfer. Systems and processes must be developed that can continuously process tritium flow rates and quantities far beyond current practice. And, to operate safely, highly effective tritium containment meeting ever stricter regulatory requirements must be maintained while contending with the harsh fusion environment.

Key questions to be answered include:

- Can fuel cycle processing time be shortened in DEMO by eliminating full isotope separation or modifying other steps to reduce tritium inventory circulating through the fuel cycle system?
- What are the major processes by which tritium might be at risk; what tritium accountancy levels can be achieved?
- What techniques can be used that ensure steady state fusion burn operation with high tritium system availability?

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Chapter 3: Assessment of Options to Address Grand Challenge and Ancillary Issues

3.1. Overview of Research Approach and Prioritization

Overall, the field of fusion nuclear sciences is approaching a critical juncture. A variety of innovative and high performance fusion nuclear science concepts have been proposed over the past two decades, and scoping experimental and computational research (single-effects feasibility studies) have resulted in a subset of these concepts emerging as potentially viable and attractive options for further development. Considering the current limited state of knowledge in fusion nuclear sciences, it is considered prudent for near-term research to focus on single-effects and multiple-effects phenomena, with preferential emphasis on the material, technology or concept that is considered to be the most technologically advanced option ("front-runner"). This focusing of research on the front-runner options is considered to be a natural evolution in the development of advanced technologies, based on experiences from NASA programs, aerospace programs, and advanced fission reactor projects such as the Next Generation Nuclear Plant. The basis for this focusing on front-runner concepts as projects move from concept exploration to the proof of principle stage is largely driven by budgetary considerations associated with maintaining parallel research tracks on multiple concepts. This focus should not be taken to mean the complete exclusion of alternate pathways, but instead is meant to indicate a majority portion of the research activities will be devoted to the front-runner, depending on the importance of and confidence in the system considered.

Finding: There are inherent inefficiencies and costs associated with exploring multiple materials or concept options once the technological maturity has grown beyond the concept exploration stage (TRL 1-3). Thus research to explore the scientific proof of principle for fusion energy is most expediently accomplished by focusing research activities on the most technologically advanced option.

The panel considered Technology Readiness Levels (TRLs) to be a useful framework for quantifying the technological maturity of fusion materials, components, and systems. "Technology readiness" is a concept increasingly employed in both government and industry in the US that uses quantitative metrics (TRL1-9) as one element to consider in program planning, progress evaluation, and R&D gap and risk assessment (TRLs are not a stand-alone program planning tool; expert judgment is required for the analysis and the TRL quantification depends on the specific application or goal that is desired to be achieved, i.e., the definition of TRL 9 for a given program strongly impacts the assignment of intermediate TRLs). The General Accounting Office issued a report in March, 2007 that recommended DOE and other federal agencies should use TRLs for major construction projects in order to achieve a consistent basis for evaluating readiness to proceed to construction (report GAO-07-336). Current fusion TRL estimates were weighed against other metrics such as potential attractiveness, importance and urgency for fusion development to determine front-runner concepts. Given likely restrictions on fusion development resources, a strategy is recommended where front-runner concepts receive the majority of available resources to move them beyond TRL3 (proof of concept) to TRL4-6 (relevant multi-effect to partially integrated environment). A small fraction of resources should be used to continue the development of back up options having markedly distinct feasibility issues or high pay off performance potential. A detailed resource allocation between front-runner and back up should be evaluated on a case-by-case basis.

The panel felt it should not be necessary to perform a new comprehensive analysis of the full range of potential options for fusion materials, blankets and structures in order to determine the front-runner and backup technology options.

In addition to consideration of TRLs, several other prioritization metrics such as those outlined in the Greenwald FESAC report [3.1] are important factors to consider when establishing programmatic R&D priorities. The Greenwald panel prioritization used three major criteria: 1) *Importance* for fusion energy and degree of extrapolation needed from current state of knowledge, *Urgency* (level of activity required now and in the near future; is this a scientific showstopper for fusion energy or on the critical path for fusion feasibility?), and 3) *Generality*; applicability of the solution to multiple designs or DEMO concepts. An additional significant factor in evaluating R&D priorities is whether the particular research activity provides an opportunity for US leadership, or whether the activity could be performed in partnership with other nations. For each of the three topical fusion nuclear science themes, a systems-level approach is recommended to guide and prioritize the long list of potential research activities.

Finding: Most existing US fusion technology test stands are no longer unique or world-leading. However, numerous compelling opportunities for high-impact fusion research may be achievable by making modifications to existing facilities and/or moderate investment in new medium-scale facilities.

Nearly all of the facilities currently being used for fusion technology studies are small- to medium-scale facilities that were constructed in the previous century. Although there are some notable exceptions in niche areas of fusion technology where US facilities are state of the art, in most cases the US fusion technology test stands are insufficient without upgrades to perform the more sophisticated exploratory research needed to establish scientific proof of principle for fusion energy. The experimental portions of the next stage of fusion nuclear science research are anticipated to be performed predominantly in dedicated medium scale fusion technology facilities (e.g., linear plasma device(s) might be profitably used to explore critical length scale and edge physics relationships that strongly affect redeposition of sputtered materials in plasma facing components; an intense irradiation facility might be used to explore new variations of computationally-designed ferritic steels that could provide sufficient resistance to degradation from fusion neutrons; one or more tritium science facilities could explore viable mechanisms to efficiently and reliably extract tritium from hot coolants and examine other chemical science issues; and a nonnuclear thermohydraulic/ magnetohydrodynamics facility might explore the complex fluid interactions and flow perturbations in fusion-relevant coolant channels and magnetic fields).

For near-term research activities, it is recommended to place the highest emphasis on scientific feasibility issues that will directly impact the path to be followed to a DEMO fusion device. It is recognized that numerous important engineering issues (such as safety criteria that regulatory and funding agencies will require as part of the evaluation and approval process for a next-step large scale fusion nuclear device)) must be addressed over the coming decade in order to proceed to a decision regarding an FNSF. Considering that at the present time it is uncertain what materials would ultimately be selected for an FNSF or DEMO (i.e., it is considered likely that several of the current leading candidates for particular components or concepts will require modifications in constituents, heat treatments or fabrication, to be determined by future research), it is premature to perform a materials qualification test program that would be needed for a traditional style of regulatory approval. However, some effort on establishing a framework for identifying the unique aspects of fusion energy systems, compared to fission reactors, and that builds upon experiences obtained in the regulatory approval of ITER, would be useful to initiate during the next 5 years.

Finding: Computational modeling for fusion nuclear sciences is not yet sufficiently robust to enable truly predictive results to be obtained, but considerable reductions in risk, cost and schedule can be achieved by careful integration of experiment and modeling.

The panel discussed the key roles of computational modeling to advance scientific understanding in a broad range of fusion nuclear science areas. The detailed research options discussed in sections 3.2-3.4 include assessment of the pros and cons of varying degrees of experimental versus modeling studies. In nearly all topical areas, computational modeling is not yet sufficiently advanced to enable stand-alone predictive results to be obtained in the absence of experimental data. For example, state-of-the-art first principle models are currently limited to simulation sizes of <1000 atoms due to many-body effects that must be included in the solution of the Schrödinger equation for atomic interactions (resulting in a scaling of order N^6 for truly first principles condensed matter physics models). Conversely, the number of atoms involved in a single isolated energetic displacement cascade event associated with neutron irradiation exceeds 1 million atoms, and much larger simulation sizes are needed to address microstructural features and stochastic events. Similarly, molecular dynamics simulations using fitted interatomic potentials (rather than true first principles methods) are limited to time scales of ~1-10 ns, whereas key diffusional interactions require time scales exceeding milliseconds. Consequently, although R&D scenarios with varying degrees of risk have been formulated for the different fusion nuclear science topical areas, the panel concludes that in general all scenarios should involve some aspect of experimental validation, i.e., computational modeling alone is not considered to be an appropriate proxy for experiment. As a corollary, experimental exploration without utilization of computational models to guide the test formulation and interpretation is also not recommended.

Looking further toward the future, exploration of complex, synergistic effects at the component level and under DEMO-relevant conditions would likely require an integrated fusion nuclear science facility (FNSF) in order to advance toward demonstration of the scientific and engineering feasibility of fusion energy. Considering the daunting complexity of a demonstration compared to the projected level of technological readiness that is anticipated to be achievable from ITER, other word class toroidal devices, and a suite of medium-scale fusion technology facilities, it would appear to be difficult for success to be simultaneously achieved in multiple technological areas that involve advancing the state of understanding from TRL~4 to 6 to a value of TRL~8 in a single step. Further details regarding the assessment of the role of various facilities in addressing key R&D topics and an estimation of the anticipated TRL levels that could be achieved in facilities prior to the construction of an FNSF or DEMO are given in sections 3.2-3.4. A brief listing of potential scientific objectives to be addressed by an FNSF is given in the following.

1. Identify potential operational windows (plasma and materials) and practical mechanisms for sustained operation of the vital interface between the 1st and 4th states of matter at DEMO-relevant conditions
   - explore scrape-off layer plasma physics and validate the predictions of computational models that enable adequate operating margin for PFCs (including eliminating or mitigating disruptions)
   - identify impact of hot walls on plasma materials interactions in an integrated fusion environment (heat flux, particle flux, neutron damage, thermomechanical stress)
   - explore the performance and reliability of front-runner divertor concepts that have emerged from research to be performed over the next decade (tungsten characterization, nanostructured tungsten, graphite, liquid metals, or SuperX, Snowflake, etc.)

2. Demonstrate the ability of DEMO-relevant components to withstand the fusion environment, including combined 14-MeV neutrons, surface heat and particle fluxes, and relevant operating pressures, temperatures, and magnetic fields.
   - build upon coupon-level results from a fusion-relevant neutron source (well-designed materials with well-characterized properties and coolant compatibilities) and non-nuclear component testing (validating ability to withstand heat and particle fluxes, as well as expected transients)
   - validate fusion-specific design rules accounting for all expected failure mechanisms

3. Explore practical conditions for tritium fuel cycle self-sufficiency and reliable extraction of fusion power at a DEMO-relevant scale
   - investigate reliable, large scale continuous tritium processing on DEMO coolants
   - achieve tritium burnup fractions >5% in the plasma and rapid exhaust to injection approaches in order to demonstrate the potential for reducing the magnitude of recirculating tritium.
   - investigate tritium containment and leakage mechanisms in DEMO-scale components to address chemical engineering science grand challenge of controlling tritium over a vast range of temperatures and concentrations
   - explore/demonstrate integrated heat extraction at efficient power conversion temperatures (identify potential synergistic failure mechanisms, etc.)

- quantify thermomechanical fatigue of components due to plasma modulation, coolant MHD instabilities, etc.

4. Evaluate viable solutions for plasma diagnostics, heating and current drive systems, and other technologies at DEMO-relevant conditions (high availability, radiation environment, high temperatures)

The plasma physics configuration envisioned for such a fusion nuclear science facility has not been precisely defined, however, various studies point to some generic characteristics. Such a facility would need to demonstrate parameters scalable to DEMO, but not necessarily accessing the same values as DEMO. The plasma would be expected to have fully non-inductive current, sufficient plasma beta for the desired neutron wall loading, tritium fueling/pumping cycle and particle control sufficient to sustain the burning plasma, ~ weeks to a month plasma durations (compared to the current values of a few seconds), viable plasma facing components for handling power with sufficiently long lifetimes to replacement, and proven plasma control systems in conjunction with diagnostics and actuators for the long operation scenarios. In general, the fusion plasma gain does not need to be at the same level as required in DEMO, to meet the requirements of a FNSF mission, and the need for an electricity production element is similarly not required. These later aspects, and a number of others, will need further study to clearly identify how technical gaps to DEMO are properly bridged. The particular mission scope taken on by a FNSF will determine the plasma requirements, and gaps remaining in plasma characteristics will need to be investigated using other facilities.

3.2 Taming the Plasma Material Interface (PMI/PFC)

A Fusion Nuclear Science Facility (FNSF) will be the first fusion device in which 1) the plasma pulse will extend to days-weeks-months, 2) both plasma loading and nuclear loading are integrated together, 3) the very long time-scale issues of PFC lifetime and plasma duration will be seen (including dust, debris, material migration), 4) continuous plasma exhaust and rapid turn-around fueling will be needed, and 5) long timescale tritium behavior in the plasma chamber – PFC - blanket environment will be observed. Providing these features in present and currently planned experimental facilities is not possible and therefore acquiring the knowledge needed to confidently design, build and operate an FNSF requires a multi-pronged scientific research program involving linear plasma devices, toroidal confinement devices, and a series of offline non-nuclear and nuclear testing facilities. Linear plasma devices can provide long uninterrupted exposures of materials to a range of plasmas, with varying degrees of integrated effects, although missing the toroidal geometry and a number of self-consistent features. The toroidal confinement devices are required to simulate the actual magnetic geometry and associated plasma flows and core-SOL-material wall couplings with limited durations, while not at the full plasma parameters expected in a FNSF. Offline facilities (i.e. high heat flux) can provide the engineering tests necessary for basic materials and integrated material and coolant solutions, as well as fission reactors and a fusion relevant neutron source for neutron irradiation effects. In combination with ITER and theory/simulation, these facilities provide the experimental database in which to project to the FNSF regime, as well as progressively more aggressive operation within the FNSF program.

3.2.1 Overview of Research Options to Address PMI/PFC Grand Challenges

We have summarized the work required to meet the Grand Challenges outlined earlier in this report requires work on a significant number of facilities, near-term (<5 years), medium-term (5-15 years) and longer-term (>15 years) research thrusts. In the sections below we discuss these program elements in detail.

3.2.1.1 Facilities for PMI/PFC related Research

Dedicated facilities, whether existing, upgradable, or new, are critical to performing the experimentation to provide the technical basis for the design, construction and operation of a FNSF, and ultimately a DEMO. For the PFC/PMI area the major facilities envisioned are listed below, along with their primary experimental demonstration.

Plasma and Other Testing Facilities

Linear plasma devices (PISCES, PISCES/TPE, etc., see Table 3.2.1 for estimated range of operational parameters)

- Upgrades required to include steady and transient high heat flux, temperature control of samples, mixed materials, tritium and irradiated samples (TPE), ion energy distribution. Aim is to target as many features as can be accommodated to simulate FNSF/DEMO characteristics

Magnetized Plasma RF test stand (MPRF)

Antenna conditioning testing with plasma and magnetic field

- Validate advanced RF antenna designs,
- RF sheath physics modeling
- Plasma-wall (antenna element) interaction
- Qualify shield and coating materials

MicroWave Test Stand (MWTS)

Test gyrotrons, transmission lines, and launchers for ECRH

- Validate advanced EC launcher designs
- Plasma-wall (launcher element) interaction
- Qualify mirror coating materials

High Heat Flux facilities (HHF)

- Examine simple block/tile materials coupon response to high heat flux, and scale up to actively cooled integrated scaled components. It would be useful to be capable of handling irradiated samples to determine radiation-damage effects on HHF response. These studies should include:
  - Collaboration with confinement devices for environmental testing of engineering concepts

- Integration with blanket testing facilities for FW PFCs
- Steady and transient heat loads
- Water and helium coolants
- High temperature samples 500-700^{o}C

Liquid Metal PFCs testing (LM)

- Establish feasibility of liquid metal surfaces for steady and transient heat and particle loading, simple samples to more complex components
- Based on feasibility assessment pursue application on confinement device(s)
- Parameters include flow rate, magnetic field, access to high heat flux

Additional facilities

- Tungsten material initiative and associated facilities
- Fission irradiation facilities
- Fusion relevant neutron source (IFMIF, SNS, MTS/LANSCE)
- Facilities for examination of dust production, hydrogenated dust explosion testing
- Remote handling mockup facility

MAJOR US & INTERNATIONAL LINEAR DIVERTOR PLASMA SIMULATOR PROGRAMS									REACTOR	
Institution		UCSD US	NAG U JP	UCSD / INL US	FZJ GER	DIFFER NL	DIFFER NL	MIT US	ITER ORG FR	
		PISCES-B	NAGDIS-II	PISCES-A / TPE*	PSI-II	PILOT PSI	MAGNUM PSI	DIONISOS	ITER	DEMO
Discharge type		Refl. Arc	Penning	Refl. Arc	Penning	Arc Casc.	Arc Casc.	Rf Helicon	Divertor	
Power	(kW)	5–15	10.5	5–10	6.5	45	270	< 5		
p_{target}	(Pa)	0.01–1	0.1–4	0.01–1	0.01–0.1	1–10	< 10	0.01–1		
T_i	(eV)	2–5	2–5	2–5	< 12	0.1–5	0.1–10	2–5		
E_i (bias target)	(eV)	10–300	50	10–200	-	0.1–5	0.1–10	20–350	10–300	< 100
T_e	(eV)	3–50	10	3–20	1-20	0.1–5	0.1–10	5–10	< 100	< 100
n_e	(m^{-3})	10^{17}–$2 \cdot 10^{19}$	$6 \cdot 10^{19}$	10^{17}–10^{19}	10^{19}	$4 \cdot 10^{21}$	10^{20}	10^{17}–10^{19}	$> 10^{19}$	$\sim 10^{20}$
Ion flux	($m^{-2}s^{-1}$)	10^{21}–$2 \cdot 10^{22}$	10^{22}	10^{21}–$3 \cdot 10^{22}$	10^{22}	$5 \cdot 10^{24}$	10^{24}	10^{20}–10^{22}	$> 10^{23}$	10^{23}–10^{24}
Energy flux	($MW \cdot m^{-2}$)	1–10	0.01	2	0.1	30	10	0.6	5–10	10
B	(T)	0.04	0.25	0.1	0.1	1.6	3	< 0.1	5	10
Beam dia.	(cm)	5	2	5	6–15	1.5	10	5		
Length	(m)	1.5	2.8	1	2.5	0.5–1	5	1		
Ion Heat (bias target)	(kW)	< 3 dc	~ 50 rf	< 1 dc	5	10 dc	50 dc + rf	< 1 dc	Thermal	Thermal
		PMI Research Capabilities								
Gas species		D, He	D, He	D, He, T*	D, He	D, He	D, He	D, He	D, T, He	D, T, He
Targets / PFC material		Be, C, W	W, C	Be*, W, C	W, C	W, C	W, C	W, C	Be, C, W	W
Pulse length	(s)	Steady	Steady	Steady	Steady	10	Steady	Steady	300–500	Steady
Impurity PMI		Be, C							Be, C, W	W
Transients		YAG	YAG						ELM	ELM
Damage								MeV Ions	Neutron	Neutron
Ion fluence (/ 8h day)	(m^{-2})	$\sim 10^{27}$	$\sim 10^{26}$	$\sim 10^{26}$	$\sim 10^{26}$	$\sim 10^{26}$	$\sim 10^{28}$	$\sim 10^{26}$	10^{26}–10^{27}	$> 10^{28}$

IN SERVICE	NOT YET OPERATIONAL	Near ITER condition	Simulates ITER condition	Beyond ITER condition

Table 3.2.1. Operating parameter capabilities of existing Linear Plasma Devices.

Toroidal Confinement Devices

US short pulse tokamak devices (C-Mod, DIII-D, NSTX, see Table 3.2.2 for parameters)

Major SOL and material response measurement initiative, including: characterization of SOL plasma behavior, including power width, parallel and transverse flows, near and far SOL conditions, divertor detachment, and impurity, fuel, He particle transport. In addition, work is needed in:

- ELM characterization/mitigation
- Major disruption elimination/mitigation initiative
- PFC temperature impacts
- Ex-situ, in-situ and real-time measurement of material surface properties

Asian long pulse tokamaks (EAST, KSTAR, JT-60SA)

Encourage aggressive PFC/PMI programs:

- Longer plasma durations, 400-1000 s, combined with high performance core plasma, integrated with FNSF-like first wall and divertor to the extent possible.

ITER (2020 H/He, 2028 DT operation)

- Self-consistent burning plasma, high heat flux, fueling/pumping, detached divertor operation, ELM and disruption transient integration on PFC/PMI
- Tungsten divertor with Be first wall
- Dust production and material migration
- Multiple test blanket module (TBM) evaluation

Non-nuclear PFC/PMI very long pulse confinement device (also referred to as PFC Test Device)

- Examine PMI and PFC physics and engineering in tokamak configuration, with hydrogen operation and limited DD, utilizing high injected power density, high density operation, and 100% non-inductive operation to access ~days to weeks plasma pulse durations, to simulate as closely as possible the FNSF/DEMO environments

FNSF (can include a DD early phase of operation for PMI/PFC research)

- Integrated demonstration/testing of engineered FW and divertor solution (armor, structure, coolant) under combined fusion nuclear and plasma heat and particle loading
- Reliable lifetime to plasma surface and nuclear bulk degradation, at progressively higher plasma duration/performance and accumulated neutron dose (dpa)
- Integrated demonstration of burning core plasma, SOL plasma, and PMI on very long time scales of ~ days to weeks, with self-consistent fueling, pumping, particle control, erosion and migration, tritium retention and dust production.

Description	Parameter	Alcator C-Mod	NSTX-U	DIII-D	FDF	ARIES-RS
Global power density	P/S (MW/m^2)	1	0.48	0.35	0.87	~0.95
SOL collisionality	ν^*_{SOL}	5.5	2.7	2.1	4.2	~4
Peak heat flux at divertor	MW/m^{2} [g]	~20	~15	~9	< 10*	< 10*
Material		Mo + W	Li+ Graphite	Graphite	C/W	W
Cooling		Inertial	Inertial	Inertial	Gas	Gas
Ambient divertor material T	$T_{ambient}$ (K)	~300 – 900+	~300	~ 300	> 800	>1000
Ionization MFP for divertor material atoms	λ_{MFP} (mm) [k]	0.02	0.3	0.3	0.02-0.06	0.02
Divertor material ion gyroradius	ρ_I (mm) [e]	0.3	0.5	0.3	0.1-0.4	0.2
Pulse length	T_{pulse} (s)	≤ 3	~2	≤ 10	~2x10^6	~ 3x10^7

+ with hot divertor upgrade
* set by actively cooled limit

Table 3.2.2. Major PMI/PFC related parameters for US tokamak devices, a proposed FNSF device (FDF) and DEMO power plant. The US devices on a short timescale have similar characteristics in global power density, SOL collisionality, peak divertor heat flux and divertor recycling.

3.2.1.2 Research Activities to Address Plasma Material Interactions and Plasma Facing Component Grand Challenges:

These facilities described above then need to be incorporated into a coherent scientific research program. Research activities are presented for three time frames, near-term (<5 years), medium-term (5-15 years) and longer term (>15 years) foci for such an effort. We summarize the components of these activities below.

Near term thrust (1-5 years)

Extensive research can be achieved with existing US tokamaks, linear plasma simulators and offline testing facilities, which provide a significantly aggressive environment to begin the exploration of high heat loads, SOL plasma behavior, new diagnostic concepts, enhanced RF launcher designs, and material responses. All grand challenges associated with the PMI/PFC area require vastly better descriptions of the plasma behavior in the scrape-off layer region and the response of materials to the plasma exposure. Research to address this is provided by the near term (NT) activities listed below, Activities for non-DT short pulse confinement devices, with low and high temperature walls, and inductive and non-inductive plasma operation

(NT-1) Extend significantly the measurement coverage outside the plasma last closed flux surface in confinement devices, and include real time in-situ materials surface measurements, and sufficient experimental operation time to understand the plasma processes in the entire SOL. This includes determination of the radial power scrape-off width, operating conditions and stability for divertor detachment, flows of both main plasma ions and eroded materials throughout the SOL to other regions and into the plasma, behavior of fusion fuel and helium ash, and integration of solutions for the plasma facing components that are consistent with the high performance core plasma. Pursuing high temperature plasma facing components (typical of FNSF/DEMO) should be examined to the extent possible.

(NT-2) Aggressively pursue experimental and theoretical studies to eliminate or ameliorate the impact of disruptions to the extent possible in tokamaks, and transport regimes that have small energy content ELMs or no ELMs. The severity of particle and heat loading to plasma facing components is strongly influenced by the presence of transients, since relatively high steady state heat loads can be tolerated on solid material surfaces (~10-15 MW/m^2) in combination with gas cooling. The primary transients are from edge localized modes (ELMs) which have energy and particles released periodically with very short timescale, and disruptions which release the plasma's stored energy over short timescales and lead to wall-plasma contact and strong electromagnetic loads. Both the first wall and divertor receive high loading in these transients.

The possibility of divertor magnetic geometry optimizations (super-X and snowflake) should be explored with heat flux handling in conjunction with particle pumping and plasma control, in order to assess their feasibilities.

(NT-3) Develop the theory and computational models for the SOL plasma physics, plasma material surface interactions, neutral transport, and atomic and molecular processes which are critical to establishing a predictive capability for this area. In addition, modeling of PFC material evolution by neutron irradiation and hydrogen transport through and trapping within these materials are needed. The experimental validation of these simulations is of high priority.

(NT-4) Studies should be pursued to examine the need, benefit, and feasibility of a very long pulse non-nuclear PMI/PFC confinement facility to provide access the long plasma time scales anticipated in a FNSF.

Linear plasma devices and non-nuclear test stands (high heat flux, RF and microwave test stands, and extensions to confinement devices)

(NT-5) Tungsten is considered the front-runner solid plasma facing material based on high melting temperature, low sputtering yield, and high thermal conductivity;

however, it does not have an extensive material database characterizing its properties for a possible FNSF application. A significant effort is necessary to establish a feasibility assessment for basic non-nuclear material properties, nuclear exposure properties (fission and fusion to the extent possible), plasma particle and high heat flux exposure properties, and manufacturing limitations to achievable properties.

Simultaneously, basic feasibility assessments are required for liquid surface materials under particle and heat loading. This activity could include a dedicated toroidal facility or other facility types.

(NT-6) Linear plasma devices require a series of upgrades from their present capability to begin to simulate the anticipated fusion environment, including 1) heating of ion species to provide spectrum of incident angles, 2) steady state and transient heat flux simultaneous with prototypical particle flux, 3) in-situ PMI measurements to better track material evolution during loading, 4) temperature control of samples, and 5) capability to examine radiation damaged materials. The complex behavior of mixed materials, typical of an actual fusion device, should be a strong focus in these experiments.

(NT-7) Testing of small mockups for tungsten and ferritic steel based PFCs in high heat flux facilities and in linear plasma devices is required, and should be closely coordinated subsystems design studies to identify potential geometry and coolant concepts, representative of the divertor, first wall, and special in-vessel components for use in these facilities. These PFCs have significantly different loading and operating limits which must be accounted for in their development. This includes a coordinated modeling activity to both design and interpret experiments, and integrate multi-physics phenomena to provide failure mode predictions.

Extend the mockups testing above by beginning the use of fusion relevant material in-vessel components on existing confinement devices (i.e. tungsten and ferritic steel). In addition, an incremental approach to the deployment of helium cooled refractory components in confinement devices should begin as precursors to future integrated testing.

(NT-8) The development of reactor relevant diagnostic systems, including the development of reliable and verified models for improved predictive capability to reduce the measurement requirements, testing of coating techniques and advanced refractory alloys and metal doping, validating techniques for predicting particle, heat, and neutron fluxes on passive components (e.g. mirrors, sensors, tec.) in realistic geometry in both normal and off normal operating conditions using test stands, and developing designs that incorporate active cooling necessary to address the thermal management needs for long pulse.

(NT-9) To meet the demands of burning plasma devices, RF launcher (ICRF & LH) technology has the challenge to improve RF Sheath & near field modeling, including long pulse and high power effects, increase antenna power density, understand the effects of heat load, neutrons & T retention on antenna materials. Magnetized plasma long pulse based RF test stands can be used to validate antenna concepts and performance.

(NT-10) To advance the technology of microwave launchers the effects of high heat, particle, erosion, neutrons & T retention on launcher materials has to be understood, this requires that reliable and verified techniques for computing self-consistent heat and particle fluxes to high-power, energized components be developed. This may require the development of new materials, and the development/extrapolation/innovation of new concepts, which qualify structural, shield and coating material with which to construct microwave launchers, including joining/bonding technologies. This effort can be supported by the use of microwave test stands to validate innovative launcher concepts and performance, validation of techniques for computing self-consistent heat and particle fluxes to high-power, energized microwave launchers, and for in-situ tests of material surface dynamic response

Medium term thrust (5-15 years)

Based upon the successful development of SOL plasma and material surface diagnostics, and theoretical/computational developments on US facilities (or foreign), and engineering material and design evolution, pursue experiments in the longer plasma pulse length Asian tokamaks (EAST, KSTAR, JT-60SA) with associated upgrades in heating systems and in-vessel structures. In addition, results from the ITER non-nuclear phase would become available in this timeframe. Engineering advances on materials and design concepts can be pursued in both shorter pulse devices and these longer pulse devices. If the need for a very long pulse non-nuclear PMI/PFC confinement device is established then activities to support the design, construction and operation of this device will be necessary.

<u>Non-DT short pulse and long pulse confinement devices with low and high temperature walls, and inductive and non-inductive plasma operation</u>

(MT-1) Demonstration of high performance core plasma configurations integrated with DEMO-like SOL plasma regimes and plasma material interface solutions.

(MT-2) Disruption elimination/avoidance results explored on short pulse devices (NT-2) are extended and confirmed in the longer pulses of the Asian tokamaks aimed at development of the ability to eliminate/avoid disruptions, mitigate any disruptions that do occur, and have PFCs that survive mitigated disruptions in long pulse tokamaks. Approaches and techniques should be consistent with, and tested in, ITER operations, and be integrated into predictive models for FNSF.

(MT-3) Explore ELM control and avoidance (continuation of NT-2) while seeking to provide capability for no ELMs, small ELMs, and/or highly mitigated ELM operating regimes. A validated predictive understanding of the accessibility and relevance of regimes, tolerable operating space in terms of ELM energy, frequency, character, requirements for mitigation (coil location, field spectrum, pellet speed and repetition) is required.

(MT-4) Demonstrate integrated computational ability to predict controlled ELM evolution, core plasma response, divertor plasma response, and material surface response; eroded material migration from one region to another; fueling, pumping, recycling, neutral penetration of hydrogen and He over range of plasma screening regimes and divertor operating regime (attached, partially detached, detached).

(MT-5) Extensive application of PFC solutions for fusion relevant environments on existing US confinement devices, longer pulse Asian tokamaks, and/or days-weeks non-nuclear PMI/PFC confinement device, including materials and high temperature operation.

<u>Linear plasma simulators and non-nuclear test stands (high heat flux, RF and microwave test stands, and extensions to confinement devices)</u>

(MT-6) Detailed PFC design solutions for materials, geometries and coolant with prototype testing in a high heat flux facility. Expand the science-based program of modeling and testing to develop predictive capability in PFC performance and failure modes.

(MT-7) Establishment of maximum permissible steady heat loads and tolerable levels of transient loads for actively cooled solid material PFCs (tungsten) imposed by thermo-mechanical material properties under prototypical plasma conditions.

(MT-8) Liquid surface testing in an integrated environment to determine feasibility of liquid PFCs in high heat flux regions.

(MT-9) Development of continuous pumping, cryodiffusion, and no-processing fuel pellet fabrication approaches on dedicated facility.

(MT-10) Continuing studies of irradiated samples in linear plasma devices for PMI and tritium permeation studies, complemented by ion-beam induced radiation damage studies combined with PMI experiments. Aim is to develop understanding of PMI, retention, and permeation in radiation damaged materials.

(MT-11) Develop and apply new concepts, which qualify structural, shield and coating material with which to construct internal diagnostic components (e.g. mirrors, sensors, tec.), including joining/bonding technologies, implement these diagnostics on low-neutron facilities, use long-pulse facilities to develop and test in-situ approaches and procedures for calibration, use these new concepts to upgrade the diagnostics for ITER, and test nuclear-capable diagnostics on ITER, test nuclear-capable diagnostics on non-nuclear facilities

(MT-12) Perform integrated nuclear-capable antenna (RF & microwave) testing on non-nuclear confinement devices, this effort will develop antenna/launcher/mirror concepts using nuclear grade materials in a high heat environment.

Long term (15+ years) FNSF and ITER era

Assuming here the successful pre-requisite research and development, the FNSF construction and operation would begin, roughly simultaneous with DT operation in ITER for pulses of 500-3000 s. The program on the FNSF would need to address very long plasma operation of days to weeks or longer, in combination with high plasma on-time per year (ITER is ~ 5%). If sufficient plasma duration in conjunction with PFCs and their operation has not been obtained, this would be a required component of the FNSF program, presumably in a non-DT phase, with DD to allow for sufficiently high performance plasma operation. In addition, the fusion neutron environment, at a series of progressively higher fluence accumulation, at levels well beyond those of ITER over its lifetime, will be accessed. The materials and coolants used in the FNSF are directly power plant relevant, while those of ITER are generally not. A constant level of qualification of PFCs will be required during the FNSF program as the plasma duration and duty cycle, neutron exposure, and loading conditions evolve. This would require the continued operation of offline facilities discussed above that can take advantage of more capable computational tools in combination with experimental data.

(LT-1) The demonstration of PFC operating lifetimes over time scales of days, weeks, and ultimately durations that provide confident projections to DEMO for ~ 1 year of operation (between maintenance periods) or 3-5 years for replacement. This demonstration needs to include the generation of dust and debris and methods for reducing and removing it, as well as bulk material degradation, and surface erosion and redeposition of PFCs. This also includes the establishment of acceptable tritium buildup and control strategies in the plasma vacuum chamber.

(LT-2) The demonstration of first wall PFC compatibility with the blanket functions of heat removal, tritium breeding, and resistance to plasma loading in steady, transient, and off-normal regimes as necessary. This would include the assessment of the tritium permeation characteristics over extended exposure.

(LT-3) Demonstration of a high performance plasma operating scenario with very long duration self-consistent with the PFCs and the resulting plasma material interactions. This operating mode would be the product of high performance non-burning plasma demonstrations in the long pulse Asian tokamaks, and possibly a non-DT very long pulse facility, and burning plasma demonstrations in ITER, but must be 100% non-inductive for the durations anticipated.

(LT-4) The fueling, pumping and overall particle control must be demonstrated in the burning plasma regime, consistent with the injected (fuel and impurities for radiation), helium ash, and eroded species, which includes the processes of fuel burnup, recycling, neutrals penetration into the core plasma, detached divertor operation, and material migration within the plasma chamber.

(LT-5) Demonstrate long duration operation of specialty PFCs (heating and current drive and diagnostics) with sufficiently long lifetimes to project viable replacement times in a DEMO.

(LT-6) Theory, modeling and simulation development will enable the accurate prediction of PFC response to plasma surface loading conditions, implantation particle flux, fusion nuclear degradation, and thermo-mechanical evolution from the full life cycle. The SOL plasma behavior will be accurately predicted allowing detailed design and optimization under the resulting plasma processes.

3.2.1.3 The Critical Need for Theory and Simulation Development in PFC and PMI science

Simulation of PMI is typically separated into three areas: the scrape-off layer (SOL) plasma and neutrals, the electrostatic sheath region just above the material surface, and the material itself, including the surface layers and some distance into the bulk (from a few microns to mm). The main workhorse codes for the SOL use 2D multispecies fluid models for the plasma, though Monte Carlo impurity models are sometimes used and kinetic main species plasma models are beginning to appear. Neutrals are treated via Monte Carlo or fluid models. Sheath region models are inherently kinetic to include ion orbit effects, using either Monte Carlo techniques with many species and prescribed EM fields, or self-consistent particle-in-cell techniques for the main plasma. Material codes are comprised of a range of models, from Monte Carlo with binary collisions, continuum diffusion models, to molecular dynamics models. The pedestal region inside the last closed flux surface, can be considered another relevant region, and can be integrated with these edge simulation tools since it is the link between the SOL plasma and the core burning plasma. For example, edge localized modes originate in the pedestal and send their particle and energy burst into the SOL, and ultimately to the divertor or first wall.

Individually, these simulations have given confidence that we understand some of the very basic PMI issues to roughly a factor of ~2, though sometimes better. For example, this level of understanding applies to (1) agreement with D-alpha measurements confirming that strong particle recycling occurs at surfaces; (2) core impurity content can be reasonably well reproduced with physical and chemical sputtering models, though radial plasma transport coefficients must be assumed and RF effects in the SOL are poorly understood; (3) material erosion and local re-deposition on small samples are close to measurements; and (4) hydrogen concentration profiles within (undamaged) materials have the expected scale lengths. Many of these comparisons have been done in carbon-walled machines, though some include Mo- or W-coated devices.

The peak heat flux to divertor and wall surface is a critical parameter impacting the feasibility and design of the divertor of the FNSF, and yet there is no clear understanding of the physics that determines this quantity, which as a result, is usually projected using empirical studies. For the design of a FNSF, it will be essential to understand the physics governing the divertor and first-wall heat and particle loads; such understanding will require a careful program of confinement device experiment, theory and modeling.

A second key question requiring new modeling work involves the long-time equilibration of PMI in which the concentration of plasma-deposited and implanted species within the material, and subsequent evolution of the material and sheath come to a new state. This problem requires the development of well-coupled SOL-sheath-material models and then validation of these models against off-line and confinement device experiment. It will also be important to improve the transport models for the plasma and the materials where damage from high-energy particles, especially neutrons, must be included. The coupled model should be extensively validated with data from US devices and by building collaborations with operating long pulse superconducting confinement devices.

A third requirement for an FNSF device is likely the need for a kinetic model of the pedestal and SOL plasma owing to the higher pedestal temperature of the plasma in a FNSF and thus a reduced collisionality pedestal/SOL, which then invalidates the full fluid models on which many existing SOL models are based. While such models have begun to be developed, they have focused mostly on the pedestal region, which is weakly collisional; the SOL is characterized by a rapid transition from long to short mean-free paths, which is computationally more demanding.

It is worth noting at this point that the above modeling issues present examples where the essential elements of the problem to be modeled are multi-scaled, in that spatio-temporal scales cutting across many orders of magnitude are linked. This is not unlike many core plasma physics issues; thus techniques and approaches that are already being developed for the core region may be applicable to the PMI and PFC region as well.

It is distinctly possible that the heat flux presented by conventional divertor approaches will simply be impossible to handle, and thus research into alternate heat-flux handling configuration should be supported. Modeling can play a vital role here and could guide the evaluation and possible choice for testing in existing confinement devices. Possible new concepts include configuration changes such as the super-X and snowflake divertor designs, extended radiative divertor designs, liquid walls, and other possible concepts. Research into liquid divertors/walls will bring in the need to model the flow of the liquids in a strong magnetic field, which is especially challenging for liquid metals.

3.2.1.4 Design Studies and Modeling of PFC Performance

In parallel with the need to better develop theory and simulation of the plasma edge is the need to expand the coordinated science-based program of modeling and testing of PFCs. The objective is development of the capability to predict the performance of PFC subsystems by incorporating increased computational capability and in-depth science-based engineering evaluations of subsystems for FNSF and DEMO. This includes increasing our knowledge base on understanding of failure modes of PFCs and how to improve their performance.

The coordinated program initially provides information about the limits in the thermal performance of PFCs for specific heat loads. As noted above, we cannot at this time accurately predict the heat loads for an FNSF or DEMO. However we can investigate the performance limits of what we consider to be DEMO relevant heat sinks, e.g., tungsten-based systems with helium cooling. Understanding the underlying science in the fluid flow and heat transfer, demonstrating enhanced heat transfer for these systems, and understanding what limits the thermal performance for a range of heat loads guides us toward what is possible in engineered systems for FNSF and DEMO. We anticipate that high heat flux and other testing in the future will be coordinated with modeling to provide benchmark data and that the testing program will evolve into an activity where the test conditions are developed to confirm predictive models. At this point we do not have a program that investigates and tries to understand and mitigate the failure modes of PFCs, but this will clearly be needed as the program proceeds.

Another role for design development and high heat flux testing is for the development of actively-cooled PFCs to be deployed for example in upgraded US confinement devices or in Asian long pulse tokamaks that will be needed to increase power handling and access high wall temperatures. Incorporating new components into devices where the primary mission is R&D on plasma physics is not simple, and confirmation of performance as a "proof test" for acceptance will be a requirement. The path forward in this area is not clear and the possibility of a non-nuclear PMI/PFC confinement device at TRL Level 6 is included before FNSF.

For the FW of the breeding blanket, although ferritic alloys are the front runner, the solution is less clear because the more recent concerns noted above about power convected to the FW have not been yet incorporated into our design studies. The thermal conductivity of ferritics limits the maximum heat flux in the range of 1-2 MW/m^2. One design alternatives is to increase the space between the plasma and the wall. However, this may bring challenges with heating and the requirement for space creates a larger fusion core and larger magnets. Another alternative is to protect the first wall with shaped panels, as in the first wall design for ITER, or with separate limiters that intercept most of the convected power with the larger area of the first wall recessed to receive primarily radiated power and lower heat loads. The first requires that first wall be made of a refractory materials with higher conductivity (e.g., tungsten-based) that are integrated with the ferritic structure. The second requires separate structures and space that complicate requirements for cooling and breeding. A combination of design studies with detailed subsystems and some targeted testing of subcomponents will be needed to carry forward possible solutions.

3.2.1.5 High-level Metrics for Issue Resolution via this Program

Shown in Tables 3.2.3-3.2.6 are charts demonstrating the relationship between the facilities required to perform the needed research and the advancement of the high level issues toward resolution. These cover 1) plasma material interactions, 2) plasma facing component development, and 3) measurement/diagnostic and heating/current drive special plasma facing components. This advancement is characterized with Technical Readiness Levels, a common method in a number of other technical areas, such as aerospace and defense development. The TRLs are described in detail in Section 1. In addition, the red, yellow and green color shading signifies whether the development process is in its early stages (TRL 1-3, red), intermediate (TRL 4-6, yellow), or advanced (TL 7-9, green), ultimately showing retirement of the issue or completion of pre-requisite development for FNSF/DEMO.

The scientific area covered by the PMI/PFC topic is very broad with a number of strongly interacting physical processes. These processes span several orders of magnitude in space and time, as noted in the Theme description. There are several critical high level issues (metrics) that determine progress toward relevant goals in the PMI/PFC area, and these are pursued as foci for research in the tables. These tables are described below in terms of these metrics and the facility evolution to address them.

First Wall and Divertor PFCs

Both the FNSF and DEMO devices will operate with very high wall temperatures in an extreme plasma and nuclear irradiation environment. To date there is no experience with confinement devices operating under such conditions. Because the core plasma performance is strongly linked to the boundary conditions imposed by the walls, and the wall conditions in turn are driven by the conditions imposed by the core plasma, the plasma-wall system forms a tightly coupled interacting system. It is therefore likely that core plasma performance will thus be significantly influenced by these new operating conditions.

A number of critical capabilities must be met by the PFC system, including adequate steady-state and transient heat and particle exhaust, erosion control to provide adequate PFC lifetime and acceptable material migration rates; management of redeposition to minimize dust production; management and control of tritium retention in PFCs; adequate fueling and ash removal; all of these must be under conditions in which a high performance core plasma is integrated with a reactor-relevant wall system, and most of them will be faced in the pulse lengths anticipated in the new superconducting confinement experiments beginning operation in Asia and in Europe. Understanding these issues is a prerequisite for FNSF mission success, and requires assessing the highly temperature-sensitive plasma-material physics issues that drive these issues on plasma discharge timescales that are comparable to current experience.

As plasma operations begin to extend towards the long-pulse conditions anticipated for FNSF/DEMO like devices, significant PFC erosion can occur in regions exposed to high particle and thermal loads. These eroded atoms and molecules then enter the SOL plasma and are entrained within the flows occurring in that plasma region which then carry them elsewhere in the device where they are deposited in more remotely located surfaces. As a result these regions accumulate the material that was eroded from the PFCs. Hydrogenic and helium species can also be trapped in these redeposited surfaces, leading to the formation of co-deposited layers. These effects begin to emerge on timescales of 100s to many 1000s of seconds of plasma operation, and thus require suitable facilities for detailed study. Indeed, experiments have shown that these processes seem to occur throughout the duration of the plasma discharge, and thus lead to cumulative impacts on component lifetime, T inventory management and control, fuel self-sufficiency, and plasma fueling. These re-deposited materials have thermo-mechanical properties that are considerably different that the original un-eroded materials. As a result, the response of the newly formed surfaces to imposed plasma loads will deviate significantly from that of the origin material. This can lead to formation of particulates and dust within the vessel that contain tritium that could potential form safety hazards.

Our current understanding is insufficient to reliably predict the magnitude, rate, and impact of these processes and, if needed, to develop mitigation approaches. Thus research efforts are required to provide for credible predictions of both the material migration problem as well as its impact on FNSF performance, safe operation, and mission success. A multi-pronged research effort is needed to advance from the current status of these issues to the level of readiness required for a successful FNSF.

Table 3.2.3 below provides a look at science and technology issues of PMI, and Table 3.2.4 below provides a look at the facility requirements needed to develop PFCs. Proceeding along columns in Tables 3.2.3 and 3.2.4 below, such a program would need to advance sequentially in time from the left to the right of the figure exploring a wide range of topics, including divertor and wall PMI; edge and SOL plasma conditions and thermal loading physics and DEMO relevant disruption detection, avoidance and mitigation. The majority of the work should be focused on solid-wall W-based PFCs since these are the current leading candidate approach, but should likely maintain the option of exploring either carbon-based solid materials due its lack of melt damage or liquid metals to remove high divertor heat loads should the W-based approach be found to be unworkable, as well as optimized divertor magnetic configurations. Furthermore, a reliable and qualified source of PFC materials will ultimately be needed. The issues arising from each of these topics would then need to be resolved using plasma test stands, non-DT short pulse confinement experiments, and non-DT long pulse confinement experiments. Moving horizontally across Table 3.2.4, many of the divertor and PMI issues could be studied by working first at low wall temperatures. Such work would likely advance most of these issues to the TRL 4-5 stage. Extending these studies to high wall temperature in non-DT long pulse devices, combined with ITER experience, could advance our understanding for most issues into the

TRL 5-6 range. Ultimately though, experiments in a DT-based long pulse FNSF are then needed to advance to the TRL 7-8 level necessary to confidently move to a DEMO scale device.

Similarly, the device requirements, solid PFC configuration work, liquid divertor and PFC materials qualification topics could advance through the TRL3-4 stage with test stands, design studies, work in short pulse confinement devices and in upgraded test stand facilities. Advancement through TRL~6 then requires additional work in non-nuclear steady-state confinement devices and in ITER, combined with some new test stand work. Again, these efforts would then need to culminate in final development and demonstration efforts in an FNSF, which would take the readiness to TRL~7.

Major Science & technology issues: Plasma-Material Interactions

Facility	Plasma test stands	Non-DT confinement: inductive, low T	Non-DT confinement: non-inductive, low T	Non-DT confinement: non-inductive, high T	ITER: DT, inductive, low T	FNSF: DT, non-inductive, high T	DEMO
Divertor + Wall PMI							
Quiescent plasma heat/energy exhaust	1.-3. Sheath heat transmission, basica parallel plasma physics	3.-5. Non-stationary T, but possible high parallel power loading at small size	3.-5. Varying P/S ~ 0.2 - 1 MW/m2, activley cooled water	3.-6. Varying P/S ~ 0.5 - 1 MW/m2, activley cooled with gas, constant T	4.-5. Power density P/S~0.2 MW/m2,at reactor size, water cooled	7 - 8. Power density P/S~1 MW/m2, peak <10 MW/m2 one year /w neutron damage	
Transient plasma heat exhaust	1.-3. Surface response > 0.1 MJ/m2	4.-5. Disruption/ELM dynamics, too low W/S<0.02 MJ/m2			5.-7. Enery density W/S ~0.5 MJ/m2 in ~ms, pulsed	6.-7. Energy density W/S~0.5 MJ/m2 for one year	7.-8. Energy density W/S~1.5 MJ/m2 for one year
Erosion control	1.-3. Sputter yield + morphology evolution	4.-5. Cumulative erosion < 10 microns/year, local measurement rates + plasma Te reduction for control	4.-6. Cumulative erosion per shot > micron --> cumulative yearly erosion ~ mm		4.-5. Erosion at reactor size, W/C divertor, pulsed	7.-8. Peak divertor erosion < 5-10 mm/year, main-wall erosion < mm /year	
Dust and redeposit control	1.-3. Response of redeposits to plasma load, dust transport	3.-4. Basics of dust production and transport, redeposit properties	4.-5. Basics of dust production and transport, redeposits at cumulative depths > 0.1-1 mm	4.-6. Basics of dust production... > 0.1-1 mm with T > 500 C	4.-5. Deposits at reactor size, T<200 C	7.-8. <10-100 kg mobile dust, no disrupting UFOs from deposits after one year (~1e4 kg eroded)	
Tritium fuel retention	1.-4. Implantation & permeation from RT to > 500 C	3. High recycling but low and varying T	3.-4. High recycling with constant low T	4.-6. High recycling with constant high T	4.-5. Beryllium or carbon at low T, reactor-level inventory	7.-8. < 1 kg retained tritium per year, T>500 C	
Fueling, burn fraction & ash control		3.-4. Helium confinement, transport, de-enrichment		4.-5. Helium recycling control with hot W + surface morphology (fuzz)	4.-6. Fueling at reactor size, divertor He ash exhaust required	7.-8. <10% density variation, burn fraction > 1%, core He < 10% for one year	
Integrated viability of PMI with core plasma		3.-5. Core contamination, Zeff	3.-6. Erosion and power control at non-inductive densities towards P/S ~ 1 MW/m2		4.-5. Inductive scenario with low T walls	6.-7. Robust non-inductive low-Q scenario near density limit & heat removal limit	7.-8. Robust non-inductive high-Q scenario near density limit & heat removal limit
Integrated viability of PMI + nuclear damage effects	4. Irradiated sample testing					6.-7. <10 dpa radiation damage, ~30 TJ/m2 convected energy	7.-8. >10 dpa radiation damage, > 30 TJ/m2 convected energy
Device Requirements for Materials Development							

Table 3.2.3. Major PMI Science & Technology Issues

Facility	test stands and design studies	Short pulse toroidal devices; upgraded test stands	Non-nuclear SS Toroidal Devices	ITER and ITER TBM	PFC test device and/or Blanket Test Stand	FNSF	DEMO
Device Requirements for PFC Development							
Understand power flow, predict heat loads	2 edge modeling, development of new diagnostics	3 edge modeling, deployment of new diagnostics	5 better models, new diagnostics, more power	4 improved edge modeling, ITER H and D plasmas	*materials for diagnostics in IFMIF*	7 predictive models, right plasma edge (need rad-hard diag.)	8 confirm peformance for DEMO size & power
Relevant divertor size (area ratio to FW)	3 design studies	?4 MAST Super-X divertor experiment	5 EAST or other	3 ITER plasmas, wrong divertor	?6 PFC Test Device	7 right configuration, based on modeling	8 confirm peformance for DEMO size & power
DEMO relevant disruption mitigation	3 modeling, design studies	4 improved models, DIII-D experiments	5 improved techniques, experiments	6 ITER plasmas, system information, wrong edge plasma	?6 PFC Test Device	7 solution with right plasma	8 confirm peformance for DEMO size & power
Representative edge (density; parallel power flux)	3 modeling, design studies	4 improved models, experiments	5 better models, higher power	4 ITER plasmas, system information, wrong edge plasma	?6 PFC Test Device	7 solution with right plasma	8 confirm peformance for DEMO size & power
Solid PFC Configuration			develop poloidal limiters				
Relevant high temp operation	3 design studies, HHF Tests 400-600C, small mockups	4 design studies; HHF Tests 600-800C; hot wall tiles	5 400-600C PFCs	*some experience with recessed FW*	*NA*	8 relevant operation, confirm performance	8 preferred, optimized mat. & tempertures
DEMO relevant launchers, mirrors, etc.	2 modeling, design studies	3 better models & designs, HHF experiments	5 mature designs, deployed units, confirm performance	4 ITER plasmas, system information, wrong edge plasma	?6 PFC Test Device	7 relevant operation, confirm performance	8 preferred, optimized mat. & tempertures
W-based divertor	2-3 design studies, HHF experiments	4 HHF mockups; W tiles, hot wall (C-MOD); W div EAST	5 W div East-U (higher power) & Satellites	4 W div in ITER H & D plasmas	*divertor materials in IFMIF*	7 relevant operation, confirm performance	8 preferred, optimized mat. & tempertures
W-based limiters, recessed or highly-shaped high temp FW	2-3 design studies, HHF experiments	4 improved models, HHF experiments	?5 deployed W lim & recessed FW	*data from port plugs, FW modules*	*FW materialsin IFMIF*	7 relevant operation, confirm performance	8 preferred, optimized mat. &
Understand PFC failure modes, predict lifetime	2-3 design studies, HHF experiments	4 improved models, HHF experiments	4 models + HHF tests, anecdotal data	*anectdotal data, new failure modes*	?6 PFC Test Device	6-7 real data, predictive models, benchmarks	8 mature models, optimized mat.
Demonstrate acceptable div. life	2 design studies, HHF tests 400-600C	3 improved models; HHF tests, He-600C; He-cooled div., EAST	4 hot He-cooled W div East-U (more power) & Satellites	*anectdotal data*	*test samples with n-damage*	7 real data, confirm performance, models, data 20 dpa; He-600C	8 opt. mat., confirm performance, data 50-100 dpa; He-600C
Demonstrate acceptable life, integrated FW	2 design studies, HHF tests 400-600C	3 improved models; HHF tests, He-600C and blanket coolant	*NA*	*anectdotal data, shaped FW*	4-5 Blanket Test Stand	7 real data, confirm performance, models, data 20 dpa; He-600C	8 opt. mat., confirm performance, data 50-100 dpa; He-600C
Liquid Divertor (with solid FW)							
Demonstrated heat removal	2 modeling, design studies	3 improved models, HHF tests small area	4-5 HHF mockups; deployed liq. divertor	*NA*	?5 liquid divetor test in blanket test stand	6-7design based on models, confirm performance	8 opt. mat., confirm performance
Integrated op., acceptable life	2 modeling, design studies	3 improved models, improved designs and materials data	4 improved design models, mat. irradiations and PIE	*NA*	?5 data on failures from liq. div.tests in blanket test stand	5-6 design based on models, progressive phases to confirm performance, 20 dpa	7-8 opt. mat., matue life model, 50-100 dpa
Qualified PFC Supply							
Develop/qualify divertor fabrication process	2-3 design studies;	4-6 HHF and mat tests, irrad. data, improved mat., ITER experience & processes, off-line work with vendors				7 opt. process, large material lot., good QA	8 near commercial material and QA
Develop/qualify limiter fab process	2-3 design studies;	4-6 HHF and mat tests, irrad. data, improved mat., ITER experience & processes, off-line work with vendors				7 opt. process, large material lot., good QA	8 near commercial material and QA

Table 3.2.4. Device Requirements for Plasma Facing Component Development

Measurements/Diagnostics

The safety and operation of burning plasmas in FNSF and DEMO will require precise control of the plasma as well as monitoring of the engineering systems. These will require complex plasma diagnostics and instrumentation embedded in the high radiation environment. The severe environment of nuclear radiation and high temperatures require a strong integration into the design of an FNSF or DEMO. There are distinct PMI issues that arise when considering measurement and diagnostic requirements in an FNSF, as well as unique issues that arise when considering heating and current drive systems within an FNSF as well. Diagnostic components near the plasma pose the greatest difficulty in extrapolation to future devices. A chart of the high level issues for Diagnostic Research and Development along with the pathway to resolving these issues is shown in Table 3.2.5.

The limited access in an FNSF implies that plasma diagnostics will be less numerous and have less resolution than on current tokamaks. Furthermore they will have to be extremely reliable, maintain precise calibration and have redundant components available. Determining and providing the minimum necessary set of measurements for a successful FNSF is a key physics issue. As illustrated in Table 3.2.5, these issues can be advanced to TRL~ 3 statuses by off-line single effect studies combined with work in short pulse toroidal devices. Suitable work in the new non-nuclear steady-state toroidal devices and in ITER can then move this readiness to TRL~6. Ultimately, work in an FSNF is then needed to advance to TRL7-8 in preparation for fielding a DEMO device.

In addition, the performance of many other auxiliary systems (e.g. plasma fueling, breeding in the blankets, high heat flux PFC health) may need to be monitored by instrumentation close to the radiation source. This instrumentation, while not being as well defined as the plasma diagnostics, will have very similar needs for operation in the radiation environment. A similar development program for the instrumentation to support the control of all the engineering systems will thus likely be needed. Such a program would need to advance the measurement capabilities from the current low readiness to very high technological readiness as illustrated in Table 3.2.5.

One key aspect of such a program would be selection of the materials in diagnostic PFCs, taking into account irradiation induced changes such as volumetric swelling thermal conductivity degradation, radiation induced conductivity in insulating ceramics, radiation induced electrical degradation, radiation induced electro-motive force, color center formation and radio-luminescence and surface effects that reduce the optical quality of materials. Initially survivability assessments at lower irradiation dose should be the research focus. This would need to be followed by an intermediate to long-term research effort that would pursue the FNSF and DEMO levels of exposure. Work with simulations, modeling, non-nuclear test stands, ion beams and fission reactors and non-nuclear short-pulse confinement devices could move these issues up to TRL~3 development stage. Advancing to TRL~6 would then require additional work using non-nuclear long pulse toroidal devices in concert with ITER based work. Ultimately then advancing these issues to the highest readiness levels then requires an FNSF.

The development of real-time interpretation and analysis of diagnostic measurements for operational control and decision making is also a critical element of the Measurement and Diagnostic challenge. Reactor-relevant burning plasmas will operate for long times, and measurements will need to be available continuously. Control systems will require the availability of real-time measurements to control actuator systems, and provide machine operators to command and control changes in device operating state. Significant progress in this area has allowed for real-time measurement of plasma profiles and equilibria in present devices. These techniques will need to be significantly advanced, expanded, and tested for application to burning plasmas. Furthermore, the plasma in an FNSF or DEMO will be expected to recover from transient "off-normal" events, such as disruptions or "near-disruptions" (where some mitigation technique has prevented complete plasma quenching). Developing control tools for such events is an active part of the present tokamak program. However, since a full-power disruption is likely to be very damaging, studies of ways to avoid or mitigate its impact in existing experiments must have the highest priority. Careful evaluation of the plasma diagnostic needs to supply the necessary data for control through the event will have to be studied. Controlling, and possibly preventing, harmful ELMs requires a similar effort.

Research and development for in situ diagnostic and measurement maintenance and calibration techniques that can be performed during D-T plasma operations will also be required. Development of large scale, radiation-hard robotic devices that can provide dexterous manipulation and precise positioning of large, multi-ton, highly activated in-vessel components, preferably with simple linear and time efficient motions. Development of specialty remote tooling and end-effectors, including precision remote metrology systems to measure plasma facing component alignment, swelling, and erosion in the extreme fusion environment (dusty, high radiation, high temperature, and high vacuum) is required. Development of hot cell remote handling systems and tooling necessary to refurbish and/ or waste process the activated in-vessel components will also be needed.

As illustrated in Table 3.2.5, advancing these capabilities and issues from the current low TRL levels to those needed for DEMO requires sequential work moving from simulation and modeling, through non-nuclear test-stands and into confinement experiments together with ion beam and fission reactor based studies. These will provide TRL~3-4 levels of readiness; moving beyond this will then require work in the new long-pulse non-nuclear Asian devices combined with ITER. Ultimately TRL~7 and higher readiness will then require development and demonstration on the FNSF.

DEMO Relavent Diagnostics R&D

Key facilities/Hardware	Computer simulation / integrated modeling	Non-nuclear test stands (mechanical, fabrication) (HHF, PSI, LM PFC, etc)	Ion beams & fission reactors	Short pulse Toroidal devices (DIII-D, CMOD, NSTX, etc.)	Non-nuclear SS Toroidal Devices (EAST, KSTAR, JT60-SA)	ITER	FNSF	DEMO
Define Minimum Set of Measurement Requirements for DEMO Control								
DEMO relevant diagnostics		2. Can computer simulation be used to compensate for lack of spacial resolution?		3. Explore if "detuned" diagnostics can safely control the plasma				
DEMO relevant FW & divertor instrumentation		2. Prototype robust instrumentation "smart ties,"		3. Deploy "smart tiles," on short pulse toroidal devices.				
Sensor proximity *viability*				2 -3.Explore sensor proximity effectiveness				
Diagnostic robustness		2.Test prorotypes of new diagnostic sub-elements on single-effect test stands		3.Test prototypes of new diagnostics on existing devices	4-6.. Develop prototypes of new diagnostics using reactor relevant material and test on non-nuclear devices and/or ITER		7.Test new diagnostics using reactor relevant material on FNSF	8. Validate new diagnostics at full parameters
Material Selection of FW Components								
Material qualification: mirrors, front end effectors	2. Modeling , design studies	2.Better models & designs, HHF experiments	3. Rad. stability of joints & fabrication approaches, up to 70 dpaino He	3.Test mirrors fabricated from reactor relevant material and test on DIII-D	4.-5. Develop prototypes of mirrors fabricated from reactor relevant material and test on non-nuclear SS toroidal devices and/or ITER confirm performance		7. Relevant operation, confirm performance	8. Validatef mirror performance at full parameters
Material qualification: insulators		2. Identify candidate materials, prioritizes into primary and secondary candidates	3. Fabricate, and test samples for irradiation affects, up to 150 dpa/With He, stress.	3. Develop prototypes of ceramic insualtor material and test on existing devices	4. Develop protctypes of mirrors fabricated from reactor relevant material and test on non-nuclear SS toroidal devices and/or ITER for long pulse PMI effects			
Develop Real-Time Interpretation and Analysis of Diagnostic Measurements								
Explore real-time measurement and analysis capability	2. Evaluate enhanced modeling of diagnostic measurements			3. Demonstrate the effectiveness of modeling on DIII-D, CMOD diagnostic	4.Use detuned diag. & modeling to show effective plama control			
DEMO relevant disruption mitigation	3.Develop control tools for mitigation and recovery from "off-normal" events			3.-4 Improved models, DIII-D experiments	4.-5. Improved techniques, experiments	5.-6. ITER Plasmas, systm information, (wrong edge plasma)	7. Validate with right plasma edge	8.Confirm performance for DEMO size & Power
Calibration, Reliability and Robustness								
In-situ Clibration		2.-3. Explore real time in-situ calibration systems on single-effect test stands		4. Demonstrate techniques for in-situ diagnostic calibration	5.-6. Demonstrate in-situ diagnostic calibrate on SS toroidal devices			
Integrated remote maintenance strategy		2.-3. Develop llarge scale, radiation-hard robotic devices				6.-7. Demonstrate llarge scale, radiation-hard robotic devices.		8.Confirm performance for DEMO size

Table 3.2.5. Research and Development needs and progression for DEMO relevant diagnostic development.

Ion Cyclotron, Lower Hybrid, and Electron Cyclotron Antenna/Launcher Research and Development Plan

Reliable performance of radiofrequency antennas and microwave launchers is needed for the very long plasma duration envisioned for FNSF and DEMO. Many of the issues concerning the interaction of the near field of the antenna and launcher with the scrape-off layer plasma (SOL) are not well understood. There is a need to predict, measure, adjust for, and modify this edge environment since it is the region through which power is coupled. Radiofrequency breakdown and arcing in the antenna structure are some of the main power limiting issues with operating RF antennas in the plasma environment, and are poorly understood. In addition, the antenna structure and Faraday shield will likely be constructed from layered or coated materials that require good conductivity and high heat resistance. The behavior of these structures in a nuclear environment and their survivability in long-term operations are concerns. A chart of the high level issues for Plasma Heating Research and Development is shown in Table 3.2.6 and briefly described below.

Advancement of IC/LH/EC Antenna/Launcher technology requires a number of capabilities. Accurate modeling of the edge geometry, including the PFC-RF sheath interaction and edge gradients that are on the scale of the radiofrequency wavelength, is needed. These tools must succeed in predicting radiofrequency coupling and loading of an antenna to the edge plasma in the reactor environment, where antenna-plasma separation will be larger than in present-day experiments. Integrated modeling needs include development of RF-MHD and RF-turbulence interactions.

Diagnostics for launched wave and mode converted wave detection, as well as fast particle detection, should be deployed and compared with synthetic diagnostics implemented in radiofrequency codes. Radiofrequency probe arrays in the SOL would allow improved radio frequency sheath physics and wave coupling understanding. Measurements of radiofrequency sheaths and heat flux measurements on plasma facing components using infrared thermography and calorimetry with extensive 3-D coverage, and in-situ erosion and redeposition measurements, are needed for rigorous testing of thermal load predictions as well as prediction of impurity generation rate and distribution. Data from integrated high heat flux, steady-state confinement devices with relevant wall conditions, complemented by basic plasma physics devices and plasma-wall interaction test stands, will be required for validation.

A dedicated radiofrequency test facility can be used to develop and validate antenna performance issues and radiofrequency sheath formation. Arcing and breakdown issues can be addressed where multiple parameters can be controlled and tested for long-pulse operation. Other issues that can be addressed include simulation of the antenna interaction with the SOL , including radiofrequency sheath dynamics, antenna phasing effects, hot spot formation, localized erosion, transport along and across magnetic field lines, and wave-particle interactions.

In order to reduce the impact of these structures on tritium breeding, it is desirable to maximize the Antenna/Launcher power density. In particular, even though ITER will establish and demonstrate the technology needed for reactor level ICH systems, the results may indicate that the standard strap antennas do not have the coupling efficiency and performance needed for DEMO. The decreased ICRF coupling with large antenna-plasma gaps could be addressed by developing alternate antenna configurations, which have higher antenna-plasma gap tolerance. It is anticipated that the present and planned RF test stands will provide most of the validation needed for DEMO, other than the validation of the compatibility and robustness of the antenna system within the fusion nuclear environment. Presently there are no test stands devoted to the development and validation of ECH mirrors that will be needed for DEMO. The information for the design of such mirrors is expected to come from the testing and validation of DEMO qualified PFCs.

DEMO Relavent Ion Cyclotron, Lower Hybrid, and Electron Cyclotron Launcher R&D

Key Facilities/Hardware	Computer simulation / integrated modeling	Non-nuclear test stands (mechanical, fabrication) (HHF, MPRF, MWTS, etc)	Ion beams & fission reactors	Short pulse Toroidal devices (DIII-D, CMOD, NSTX, etc.)	Non-nuclear SS Toroidal Devices (EAST, KSTAR, JT60-SA)	ITER	FNSF	DEMO
Advancement of IC/LH/EC Antenna/Launcher Technology								
Improve RF Sheath & near field modeling	2.-3. Improve models for sheath physics							
Measure the SOL near and in front of the IC/ LH/EC antennas/launchers		3. Validate diagnosic performance on plasma-wall interaction test stands.		4. Perform diagnostic performance validation on DIII-D, CMOD & NSTX	5.-6. Perform diagnostic performance validation on SS toroidal devices			
Effects of heat load, neutrons & T retention on antenna/launcher materials	2. Model heat and particle fluxes to high-power, energized components.	2.-3. Qualify structural, shield and coating material for antennas/launchers	3. Test coating techniques and advanced refractory alloys and metal doping	3.-4. Develop and test designs that incorporate active cooling	5.-6. Validate antenna/ launcher performance on SS toroidal devices.	6.-7. Evaluate the anticipated contribution to dust formation from the materials selected for IC/LH/ EC antenna/launcher front ends		8. Validate IC/LH/ EC antennas/launchers at full parameters
Increase Antenna/Launcher Power								
Validate new IC/LH/EC antenna/launcher concepts	2. Develop improved models of the properties that cause arcing.	3. Validate antenna/launcher concepts on long pulse MPRF, MWTS test stands		3. Test advanced IC/LH/EC antennas/launchers on non-nuclear facilities	4.-5. Test IC/LH/EC antenna/ launcher concepts using nuclear grade materials	6. Test advanced IC/LH/EC Antennas / launchers in a nuclear environmen	7. Perform engineering validation (RAMI) on a FNSF device	8. Validate advanced IC/LH/EC antennas/ launchers at full parameters

Table 3.2.6. Research and Development needs and progression for DEMO relevant plasma facing heating and current drive development.

3.2.2 Linking Grand Challenges, Research Program Elements and Facilities

The three Grand Challenges associated with the PMI and PFCs of an FNSF/DEMO device are related to elements of the near-term, medium term, and long-term research program discussed above. In general, the various facilities needed to perform the experimental work contribute to all the grand challenges in varying degrees, and it is not possible to relate a single facility and a single issue in any case.

3.2.2.1 Understand the coupled evolution of the plasma and PFCs under prototypical thermal, physical and chemical conditions expected in an FNSF/DEMO*.*

Critical elements of this challenge include creating and sustaining a sufficiently high performance plasma, which is 100% non-inductively driven, for the FNSF mission, and ultimately a DEMO with a power production mission. This is integrated with plasma facing materials of long-term fusion relevance operated at relevant temperatures associated with high thermal conversion efficiency. Providing for the exposure of the plasma facing materials to a SOL plasma with fusion relevant parameters at sufficient plasma durations accesses the regime of self-consistent behavior of the fusion plasma, the SOL plasma, and the plasma facing materials. An evolution is required to provide the FNSF and DEMO environments. Examining the research program plan, we identify the near-term activities including NT-1 through 4, medium term activities MT-1, MT-4, MT-9, and long-term activities LT-3, LT-4 as providing important contributions to retiring this issue.

3.2.2.2 Understand, predict and manage the material erosion and migration that will occur in the month-to-year-long plasma durations required in FNSF/DEMO devices, due to plasma-material interactions and scrape-off layer plasma processes.

Critical elements of this challenge include providing long-duration confined plasmas to provide the correct SOL plasma physics processes for material removal and movement within the vacuum vessel and associated in-vessel components. ln-situ real-time measurement of the evolution both of the eroded materials and the re-deposited materials and the associated co-deposition of hydrogenic and helium species is required. An important element of this is the self-consistent fueling, pumping, and overall particle control to understand the particle inventories, particularly those associated with tritium retention through co-deposited layers and containment in dust and debris. Relevant research activities would include elements from NT-1, NT-2, NT-4, NT-5, NT-6, NT-8, NT-9, NT-10, MT-1, MT-5, MT-7, MT-9, MT-13, and LT-1 through LT-6

3.2.2.3 Understand and mitigate the deleterious effects in plasma facing materials from both intense fusion neutron and plasma exposure that continuously damages materials surrounding the plasma.

Critical elements of this challenge include the exposure of plasma facing components to the simultaneous loading of neutrons (largely affecting the bulk material), plasma particles, and plasma heating both conducted via plasma and radiated from the plasma. The plasma loading conditions must be consistent with the operating regime of interest, and therefore may contain transients and off-normal features. The development of plasma facing materials that are resilient to this environment, and have significant safety and nuclear waste advantages is pursued. The development of engineering solutions to integrated plasma facing component materials, geometries, and operating parameters that can accommodate the loading environment and provide sufficient lifetime to meet the needs of a FNSF or DEMO. Research activities include nearly all of the activities discussed above, including NT-1 through 10, MT-2 through 12, and all of the long-term research activities identified above.

3.2.2.4 Options to Address Grand Challenges and Ancillary Issues

The challenge of the plasma material interface, and the plasma facing materials that must endure it, is understanding the strongly coupled physical system including the fusion plasma, the scrape-off layer (SOL) plasma and the material walls that interface with this SOL plasma. The loading conditions seen by the plasma facing materials include particles with a wide range of energies, conducted and radiated heating, and neutrons. Simultaneously, the SOL plasma processes, which include strongly anisotropic transport, flows, and neutralization and ionization, determine the materials surface evolution. The plasma facing material will operate at high temperature and sustain large temperature gradients between its coolant and plasma facing surface. The presence of transients, which may be unavoidable, will aggravate several degradation mechanisms in the material surface and bulk. The FNSF, and DEMO, will present a unique and untested environment for plasma facing components, however, research and development in preparation for the FNSF can provide an experimental basis, with the aid of theory and simulation, that can be used to construct a confident projection for describing the environment and the resulting impact on materials.

The research required to establish a technical basis to proceed to the FNSF conditions is composed of facilities for doing the research, research activities and theory and simulation development, all described in earlier subsections. The primary issue with research in the PMI/PFC area is that of providing all the physical conditions simultaneously. In fact, in advance we already know we can not provide this situation experimentally until we reach FNSF, and so the question arises *what are the appropriate steps to take in providing partial environments to develop our understanding and predictive capability prior to FNSF?* This question is also relevant to the approval or licensing of a FNSF, in that a collection of physical data and simulations will be relied upon to project the PFCs behavior in the FNSF, in spite of the fact that the integrated environment in the FNSF has not been reproduced in any facility. This collection must be “convincing” and “defendable” to a technically knowledgeable review board. As it is outlined in Tables 3.2.3-3.2.6, and the discussion of research activities, there are a number of parameters we are trying to meet, albeit separately or in combinations on various facilities.

Heat flux

Particle flux

Steady and transient fluxes

Plasma exposure time (plasma duration)

Integration with a SOL plasma and core plasma (confinement devices only)

Integration with a SOL plasma and non-inductive/high performance core plasma (confinement devices only)

Integration with a SOL plasma and fusion burning core plasma (ITER only, prior to FNSF)

Plasma facing material

Plasma facing material temperature

Presence of tritium as hydrogenic species

Neutron exposure (damage) of plasma facing material, and the associated energy spectrum (fission, fusion, or other, and dpa and He)

Examining the Table on Plasma Material Interaction in Table 3.2.3, moving to the right is equivalent to increasing the integration of variables noted above, particularly performing experiments on confinement devices. The linear plasma devices could likely access (with the appropriate upgrades) very long plasma exposure times and a number of FNSF-similar parameters (weeks duration, heat flux, particle flux, and steady and transient fluxes, with appropriate materials and at the correct temperatures), however they do not provide the self-consistent nature of the plasma facing material and the core and SOL plasma. The information derived from linear plasma devices is quite valuable for seeing these parameter effects, but should be interpreted with connections to the confinement device results in regimes that can be accessed by both. This connection between the two PMI environments would benefit greatly by theory and simulation, which would accelerate the understanding of differences in experimental results and interpretations.

The nuclear impact on plasma facing components would largely be addressed separately, and not integrated with plasma exposure. The possible exception is the plasma exposure of an irradiated sample in a linear plasma device, for example at the STAR facility at INL, where such capability exists, or high heat flux facility with the required upgrades.

The Table 3.2.4 on PFC development also demonstrates the evolution to more relevant parameters, or more simultaneous parameters as one moves to the right. The primary engineering function of PFCs is power handling, and the high heat flux capability of candidate materials and component designs requires early assessment, which is best handled in dedicated facilities. The second engineering function is to provide power handling capability for a reasonable lifetime, and so the evolution of research moves to exposure of the materials and component designs in actual confinement devices, where the combination of high heat flux and the numerous other conditions can be sampled. The modeling activities associated with the engineering of plasma facing components is invaluable in both interpreting experimental results and deriving new concepts. The modeling capability will almost certainly have to expand beyond conventional thermo-mechanics to incorporate the extreme loading conditions, the bulk material property evolution, the material surface evolution, and the need for integration of several physics models to properly account for overall behavior and failure mechanisms.

In terms of confinement devices and the progressive integration of more parameters relevant to PMI and PFC development, Table 3.2.7 shows how these are advanced in different facilities. The plasma duration, the time over which the plasma is on in a given plasma pulse, is increasing from the short pulsed US tokamaks (~ 10 s) to the long pulse Asian (300-1000 s) and ITER (500-3000 s) tokamaks. However all parameters are not necessarily advancing with plasma duration. In fact, the short pulse tokamaks are likely to access higher heat fluxes than the long pulse Asian tokamaks. Although EAST has recently called for more PMI/PFC relevant research (metal walls, tungsten, high operating temperature), KSTAR has not indicated a programmatic element emphasizing this. Meanwhile ITER provides the unique feature of high heat loads and a burning plasma, which is critically dependent on the particle control inside the vacuum vessel (as will a FNSF and DEMO), and should see the effects of tritium retention, PFC material erosion and migration, but utilizes materials and operating temperatures that are not FNSF relevant. The importance of plasma duration can be seen in the table below where a series of collective processes and their time scales are listed. Since the erosion process never stops, the migration of this material continues to evolve, which is noted below as “continuous”.

Wall Thermal Equilibration	few seconds
Core Plasma Response to Boundary Changes	1-10 sec
Surface morphology evolution	100’s -10^3 sec
Near-surface (few microns) saturation with incident ions	100s – 1000s sec
D/T Permeation into bulk	10^3-10^4 sec
Macroscopic migration leading to significant redeposition (e.g. flaking, spalling, impacts on density control, dust formation)	> 10^3 - 10^4 sec, continuous
Radiation damage microstructural effects	>10^5-10^6 sec
PMI/PFC Component Lifetime (Minimum)	few 10^7 sec

Table 3.2.7. Summary of key time scales for several physical phenomena associated with plasma-material interactions.

The importance of exceeding the short pulse US tokamaks plasma durations is due to a series of collective effects of plasma material interactions and cumulative phenomena, such as re-deposition of eroded materials. Access to the longest plasma duration projected for EAST of 1000 s, and also any ITER experience allows the first observation of these processes. However, these experiments would provide more relevant information if they were conducted with FNSF materials and at the elevated temperatures expected for FNSF.

Moving to another extension of the plasma duration beyond 1000 s, would require a new confinement device, for example a non-nuclear PMI/PFC hydrogen facility accessing a plasma duration of weeks or an early phase of the FNSF itself, operated in DD since it is nuclear ready, and expected to operate for plasma durations of weeks.

An example of considering 3 options with associated facility assumptions are considered below, with perceived consequences in terms of reliance on modeling and technical risks. It is difficult to assess the consequences precisely from our present technical knowledge, for example, whether simulations will acquire the fidelity for projecting the integrated behavior of the SOL plasma and material interaction, or whether reaching 1000 s plasma pulses will provide the needed basis for projecting all material migration and particle balance issues to 2 week long plasma pulses (1.2 million s). Regardless, it is necessary to make some judgment about the options to prepare for the FNSF, and the resulting gaps left between the FNSF and DEMO.

<u>Option A</u>: (with modeling and analysis integrated into the development effort)
US linear test stands, and high heat flux facility(s)
US short pulse toroidal devices,
US fission reactors
ITER
US FNSF-DD
US FNSF-DT
US DEMO

US linear plasma sources assumed to have upgrades to provide examination of PMI at very long pulses, but lack the self-consistent aspects of a tokamak which leads to migration and plasma/SOL coupling. US high heat flux facilities assumed to have upgrades provide testing with heat loads and cooling that are relevant for FNS and DEMO, but will not have the capability to test mockups with irradiated materials unless specific expansions or a new facility are built for this purpose.

US short pulse tokamaks (with upgrades) will provide a self-consistent plasma/SOL with PMI, but lack long durations and do not approach FNSF plasma or other parameters, except heat flux. Higher operating temperature of PFCs can be explored, and FNSF relevant materials can be explored at some level.

Int'l ITER provides a self-consistent plasma/SOL and PMI with a burning plasma, and accesses plasma durations similar to Asian long pulse tokamaks, although plasma and other parameters are not FNSF-like, except heat fluxes. Provides initial evaluation of ELMs and disruption control scenarios and material damage. Materials and operating temperatures for PFCs are not FNSF relevant, except the tungsten in the divertor. Time frame for this data is uncertain, so it may not arrive in time to use for FNSF design.

US FNSF-DD operation phase can provide a PMI/PFC database for operation in DT by accessing the very long plasma durations and DD higher performance plasmas. Design of PFCs prior to this step must rely more on a limited database and computer modeling and simulations. Potentially this can significantly delay DT operations on FNSF. Operating for the anticipated plasma pulse lengths for DT operation in the device itself is considered an advantage.

US PFC materials nuclear evaluation has relied only on fission reactor exposure which does not include gas generation (mainly He) and associated degradation effects. In absence of fusion relevant neutron exposure, must rely on "expose and see" approach, say in steps of 5-10 dpa, (~10 dpa is considered low enough to guarantee survival for structural materials). Stepping method requires FNSF-DT program oriented to a large number of on and off campaigns with associated impacts on integrated testing and cost/scheduling of component replacement.

Modeling has been relied on to do the following:

> Project to FNSF core and SOL plasma environment from short pulse tokamaks and ITER, integrating ITER burning plasma experience (if available), and linear device at near-FNSF parameters for PMI.
>
> Project fusion neutron effects from fission only, and ion beam data, allowing design of FNSF.
>
> Evaluate PFC subsystems through detailed design studies that incorporate test results on thermal performance and failure modes and projected materials properties with radiation effects included.

Risks:

> Establishing long duration self-consistent plasma/SOL/PMI relies on ITER results, although burning plasma integration is a plus, the materials and operating regimes are not correct for a FNSF. The ITER DT phase may not be available, the H/He phases may be limited in scope and duration, and the plasma operating modes may not be long term relevant (inductive vs. non-inductive).
>
> Relying only on short pulse tokamaks, we can access neither those phenomena whose impacts become evident with long duration, e.g., continuing migration of eroded material, nor those that arise when there are large areas of PFC materials operated at high temperatures. These critical operating conditions relevant for DEMO and for an FNSF will not be available except insofar as some confinement device were directed to undertake such a mission with dedicated upgrade for this purpose.

Although FNSF-DD operation provides a viable way to obtain long pulses and self-consistent aspects of operation, we are faced with some issues. How do we design the PFCs for the DD operation given the limitations in supporting data noted above? The basis in modeling at that point would likely have quite limited validation for confident projections. Even with additional knowledge gained in DD operation, time is required for incorporating changes to design and building of new PFCs for the DT phase.

Major risk taken on in nuclear qualification due to limited or strongly distorted He/dpa effects. Could lead to severe extension of FNSF program to operate with expose and see approach in small dpa steps.

Option A+1:

Same as option A, but including the Asian tokamaks for long pulse confirmation

The addition of the long pulse Asian tokamaks allows non-DT plasma operation for up to 1000 s, which is expected to provide important information for cumulative PMI effects, which include critical safety issues such as tritium retention. The relevance of these activities could be significantly enhanced with appropriate materials and operating temperatures, and improved in-situ PMI measurements. These experimental demonstrations would provide significantly better validation for theory and simulations that would ultimately be used to project to FNSF. The programs on these devices are devoted to high performance and 100% non-inductive core plasma operation, which is the operational regime that FNSF and DEMO are targeting.

Option A+2:

Same as option A, but including IFMIF (or other fusion relevant neutron source)

The addition of a fusion relevant neutron source allows the projection to FNSF material damage levels in advance of operating on the FNSF to those exposure levels. Although this would likely be on single material or limited combinations, folding this data into a projection for the plasma exposure of PFCs is considered critical. The influence of the unique fusion neutron damage on processes such as tritium permeation through the first wall or divertor targets would be of high priority in safety assessments. Exposing these irradiated samples to plasma environments would further build confidence in the projecting to the simultaneous environment of the FNSF.

Option A+3:

Same as option A, but construct a dedicated PMI/PFC non nuclear confinement device

Here a non-nuclear PMI/PFC confinement facility would be constructed to access similar plasma durations as a FNSF (weeks), although operation would be with hydrogen and limited DD. FNSF relevant materials and operating temperatures would be used. Due to hydrogen operation and possibly likely small size, the plasma parameters in the core and SOL would not approach FNSF values, although 100% non-inductive plasmas would be required for these pulse lengths, with strong heating and current drive.

Options can also include the use of international facilities as opposed to pursuing US capabilities, and in fact the long pulse Asian and ITER tokamaks are examples of such collaborations. Linear plasma devices, as noted in Table 3.2.1, exist in various capacities around the world. High heat flux facilities also exist in several different modes (e-beams, plasma guns, etc.) internationally. The IFMIF is another example.

The ability of simulations to provide a sufficiently high fidelity physics description of both detailed *single mechanism* and *integrated phenomena* predictions, with the capability to project to an experimentally untested regime in order to remove the need to perform an experiment, is a highly desirable goal. In general, this level of computational simulation fidelity is not available in most areas of fusion science; plasma, materials, or engineering. This lack of predictive capability is aggravated by the extreme environmental conditions expected in a fusion device, which are difficult to reproduce in advance (of FNSF), and which tend to push computational requirements beyond commonly available software tools, either vendor supplied (i.e. ANSYS) or laboratory developed (i.e. UEDGE). There is a need to develop both single mechanism models, and the integrated simulations to predict relevant collective behavior. With time and space scales that span several orders of magnitude, as depicted in Fig. 2.2.1 for the PMI/PFC area, the challenges of modeling collective behavior can be appreciated.

The focus of simulation development must transition from exploration to that of experimentally validated prediction. The basic elements required are:

- single mechanism model development with progressively higher fidelity,
- development of efficient computational techniques that take maximal advantage of the computer hardware developments,
- validation using experimental data as specific to the mechanism as possible and over as wide a range of conditions as possible,
- model extension from time independent to time dependent including dynamics,
- proper physics integration with other single mechanism models into a framework for integrated simulations (i.e. equations describing geometry and underlying medium evolution, equations of state, etc.).

The focus on validated simulation capability forces the experimental (which requires adequate measurements) and computational development to occur simultaneously. It requires the full development of the capability up through time-dependent integrated simulation, and aggressively pursues comparison and correction throughout the simulation development. The combined evolution of experiment and simulation is expected to provide benefits in risk reduction, cost reduction and schedule control throughout the plasma science and fusion nuclear science programs. This type of simulation program does not exist in the fusion energy sciences. The idea that faster and more capable computers can solve such problems is incorrect in the absence of simulation development described above.

It is considered that the design, construction and operation (as well as licensing) of a DEMO would require a significant predictive capability to allow the projection to environmental conditions that are more aggressive than experienced in the preceding FNSF. The degree of projection would vary by subsystem depending on the FNSF mission and capabilities. *In other words, it is a viable technical goal to provide simulations capable of closing the gaps between FNSF and DEMO, in order to avoid the construction and operation of another FNSF device. However, it is not considered feasible, in general, to rely on simulations to eliminate experimental steps prior to a FNSF (with the exception of a few isolated circumstances)*. In fact, the FNSF provides a unique validation platform for virtually all sub-fields of fusion science.

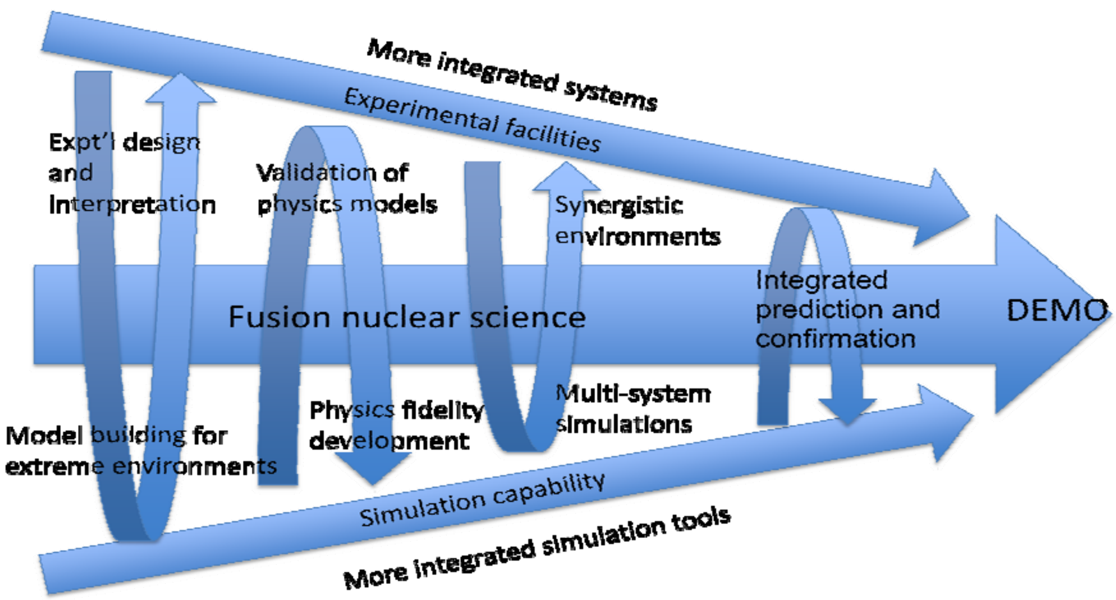

Figure 3.2.5. Illustration showing the evolution of simulation development and experimental validation as the demonstration power plant is approached.

3.3 Conquering Nuclear Degradation of Materials and Structures

Overcoming the performance challenges of materials and structures in the fusion energy environment is a daunting challenge, but yet is critically important to realize the promise of fusion as a practical energy source. Development of materials, components and structures for any complex engineered system generally occurs in a series of steps, proceeding from relatively simple single-variable experiments to very complex, fully integrated, multiple-variable tests. The first step on the development path begins with screening experiments that are performed under carefully controlled laboratory conditions. These experiments establish the basic mechanical and physical properties, chemical compatibility, and fabrication and joining technologies of candidate materials. If satisfactory results are obtained, more complex experiments are performed in order to identify materials that perform well in partially integrated or multiple-variable tests. The final stage of material development involves fully integrated experiments with prototypical test sections that are carried out in an environment combining the appropriate nuclear, chemical, thermo-mechanical and magnetic fields necessary to establish lifetime and performance limits. The ultimate goal is to develop an extensive material database, constitutive equations, and physics-based models to describe all aspects of material behavior required for fusion power system design and licensing.

This study confirmed the conclusion of many previous investigations that the single most important facility to evaluate the nuclear degradation on materials is a fusion relevant neutron source. While recent fusion materials research and development efforts have produced low activation ferritic/martensitic structural materials, as well as silicon carbide composites, with acceptable radiation resistance to doses on the order of 10 to 30 dpa, the performance of these materials with accompanying high levels of helium, hydrogen and solid transmutants in a fusion neutron spectrum remains non-existent. Thus, a fusion relevant neutron source remains a critically important linchpin in determining the feasibility of generating fusion power.

While an intense fusion relevant neutron source is clearly the overarching need for enabling progress towards a fusion demonstration reactor, it is also evident that critical materials degradation phenomena, including corrosion and high temperature deformation and fracture processes occur even in the absence of intense, high-energy neutron fluxes. Thus, it is also evident from this investigation that a number of non-nuclear facilities will be needed to test and qualify materials, components and structures for development of TBMs, and future plasma devices such as FNSF and DEMO. This section of the report therefore includes a discussion of essential work on corrosion/compatibility, advanced fabrication technology development, and magnet materials, and these topics are also discussed in Section 4 that follows.

Table 3.3.1 below highlights the facility needs identified during the analysis to determine the critical materials and structures for a DEMO fusion reactor. The columns identify critical facilities to address key scientific questions that are discussed in the rows of the table. The ordering of the facilities from left to right in the table also represents a rough timeline to DEMO. A number of these facilities involve rigorous testing of the materials in the thermal-mechanical environment anticipated for a fusion reactor, but without the neutron irradiation exposure. Thus, there are both individual testing facilities for thermal-mechanical degradation and corrosion identified as key near term needs, followed later by an integrated facility that can combine thermal-mechanical testing with a corrosive chemical coolant environment.

Table 3.3.1 also incorporates the neutron irradiation environment facilities required to address the most prominent fusion nuclear science degradation question related to the accumulation of displacement damage and transmutation products. Fission reactors and accelerator-based irradiation sources provide a good simulation of neutron-induced displacement damage and bulk-heating effects. Accelerator-based sources also permit simultaneous exploration of displacement damage and gas effects, and for specific situations enable investigation of multiple-effects such as *in situ* measurement of tritium production and release in ceramic breeders. On the other hand, it should be noted that large-volume plasma-based irradiation sources are needed to carry out fully integrated materials and component level testing. Non-plasma irradiation sources are not suitable for investigating the synergistic effects that occur in many other fusion components. Since non-plasma sources are comparatively low cost relative to plasma devices, they serve an important role in reducing the significant costs and risks associated with fully integrated, multiple variable tests performed in plasma devices.

Each row shows a scientific grand challenge or ancillary issue that must be addressed to prepare for FNSF and DEMO, and the relationship between those technical questions and the critical facilities. Estimates of the attainable Technical Readiness Level (TRL) for each grand challenge or ancillary issue are represented numerically and by color-coding of cells. The lowest TRLs (1-3) are coded red, intermediate TRLs (4-6) are yellow, and the highest TRLs (7-9) are shaded green. Culmination of materials development is TRL 9, which represents a commercial fusion power system. Note unshaded cells in the DEMO column span TRLs 7-8 indicating that FNSF contributes more strongly than DEMO toward retirement of certain issues. It is instructive to observe that none of the scientific grand challenge or ancillary issues advances beyond TRL 2-3 by conducting single-effects experiments in non-nuclear test stands or ion-beam/fission reactor facilities. None of the scientific questions can be fully resolved without performing experiments in partially or fully integrated test facilities. Another important conclusion is the limited benefit, from a materials science perspective, obtained from ITER TBM. The panel concluded that all the basic materials science needed to prepare a TBM for testing in ITER is essential for first-wall/blanket development for FNSF, but the low neutron fluence available in ITER is of limited value for resolving nuclear degradation of materials issues. The panel concluded that performing scientific experiments in partially integrated non-nuclear test stands coupled with essential work in a fusion relevant neutron source is the most effective pathway for reaching intermediate TRLs.

Significantly a fusion relevant neutron source, and a fusion nuclear science facility emerged from this analysis as the most important facilities for retiring the bulk of the scientific grand challenge and ancillary issues at intermediate TRLs on the road to a DEMO reactor. The panel noted that for FNSF to successfully perform integrated effects testing of materials and blanket concepts to permit design of a DEMO reactor, the prime candidate materials and blankets need to demonstrate safe, reliable and attractive performance to a goal fluence of ~50 dpa. Thus, to support FNSF design an irradiation effects database, derived from a fusion relevant neutron source to ~50 dpa is needed. Similarly, the panel concluded that for DEMO to demonstrate an economically attractive commercial fusion energy source, then materials, components, and structures must achieve a lifetime neutron fluence of ~150 dpa, with acceptably high levels of reliability and margins of safety. Thus an irradiation effects database to ~150 dpa will be needed to inform the design and construction of DEMO. An evaluation of possible irradiation sources that could provide the needed data is presented below.

The consideration of facility needs also illustrates a number of scientific grand challenge issues that must be addressed to enable development of the materials and structures to make fusion power a reality. These issues are described in more detail in the remainder of this section.

Table 3.3.1. Facility needs to resolve nuclear degradation of materials and structures grand challenges and ancillary issues.

Facility	Non-nuclear Test Stands (thermo-mechanical)	Non-nuclear Test Stands (corrosion)	Ion beams and Fission Reactors	ITER TBM	Non-nuclear Test Stands (partially integrated)	Fusion Relevant Intense Neutron Source	Fusion Nuclear Science Facility	DEMO
First-Wall/Blanket Structural & Vacuum Vessel Materials								
Science-based design criteria (thermo-mechanical strength)	2. Develop high temperature creep-fatigue design rules for nuclear components				4. Validate high temperature creep-fatigue design rules w/o irradiation	5. Validate irradiated high temp structural design criteria (50-150 dpa with He, stress)	7. Code qualified designs	7-8. Code qualified designs
Explore fabrication & joining tradeoffs	2. Conventional & advanced manufacturing technologies	2. Loop tests of joints & novel fabrication approaches	2. Rad. stability of joints & novel fabrication approaches		5. Validate near prototypic fabrication and joining technology w/o irradiation	6. Validate near-prototypic fabrication & joining technology (50-150 dpa with He, stress)	7. DEMO-relevant fab processes	8. Prototypic advanced fabrication
Resolve compatibility & corrosion issues		3. Basic and complex flow loops			5. Validate corrosion models w/o irradiation		7. Near prototypic operating environment	8. Prototypic extended operating environment
Scientific exploration of fundamental radiation effects in a fusion relevant environment			3. Up to 150 dpa/With He, stress (ion beams, fission reactors)			6. 50 - 150 dpa/With He and stress		
Material qualification: Structural stability in fusion environment (e.g., void swelling, irradiation creep)			3. Up to 70 dpa/no He (fission reactors)	2. Materials behavior in a low-dose, low-temp. env. (not DEMO-relevant matl, <2 dpa, low temperature)		6. 50 - 150 dpa/With He and stress	7. 10 - 50 dpa, DEMO prototypic environment	7-8. Prototypic operation, 50 - 150 dpa/With He/Fully Integrated
Material qualification: Mechanical integrity in fusion environment (e.g., strength, rad resistance, lifetime)	2. Unirrad. mech. prop. data (tensile, creep, fatigue, fract. toughness, da/dN, etc)		3. Up to 70 dpa/no He (fission reactors)	2. Materials behavior in a low-dose, low-temp. env. (not DEMO-relevant matl, <2 dpa, low temperature)	5. Qualify components w/o irradiation	6. 50 - 150 dpa/With He and stress	7. 10 - 50 dpa, DEMO prototypic environment	7-8. Prototypic operation, 50 - 150 dpa/With He/Fully Integrated
Fusion environment effects on tritium retention & permeation		2. Unirradiated diffusion and permeation data	3. Effect of radiation damage at DEMO-relevant temperatures			6. DEMO-relevant materials (up to 50-150 dpa with He at correct temp.)	7. System-scale tritium permeation and loss mechanisms	7-8. Prototypic permeation & losses

3.3.1 The lack of a fusion relevant neutron source for conducting accelerated single-variable experiments is the largest obstacle to achieving a rigorous scientific understanding and developing effective strategies for mitigating neutron-induced material degradation.

Radiation damage experienced by fusion materials shares much in common with structural materials for proposed advanced fission power systems in terms of the need to be resistant to high levels of displacement damage (displacements per atom or dpa) and to tolerate high operating temperatures. Therefore, experience gained from fission energy technology is highly relevant, but there are important differences. As noted by Stoneham, Matthews, and Ford [3.1], 14 MeV neutrons cause new features in radiation damage compared to fission or spallation neutrons. The primary knock-on atoms produced by 14 MeV neutrons possess energies up to 1 MeV. This is important because approximately half the energy losses occur from electronic scatter and half from nuclear scattering [3.1]. This is in contrast to energy deposition associated with fission neutrons in which the bulk of the energy loss is associated with nuclear scattering and therefore results in primarily displacement damage [3.1]. Furthermore, and perhaps more importantly, the interactions produced by 14 MeV fusion neutrons include inelastic scattering and threshold (n,α) and (n,p) transmutation reactions. For spallation neutrons, electronic energy losses dominate due to the higher energy PKA spectrum. In addition, there are other differences between the spallation and fusion environments such as electronic interactions of high-energy protons, spectral effects on the formation of displacement cascades, a different mix of transmutation products, and the effect of pulsing [3.2]. Considerable progress has been made in elucidating the basic mechanisms of materials degradation in an irradiation environment by utilizing a variety of irradiation sources (e.g. fission reactors, ion beams, etc.) coupled with a robust theory and modeling effort, but there remains a need for an fusion relevant neutron source that can enable accelerated testing of promising materials and subcomponents.

While scientific progress in the near term can continue to be made using nuclear reactors and ion irradiation facilities, the advances will be incremental at best. Such facilities lack the ability to perform experiments on materials and subcomponents in an environment that effectively simulates fusion nuclear conditions and provides an opportunity to conduct accelerated testing. Ultimately, significant progress towards development of materials and structures for the fusion environment requires an intense fusion neutron source that is capable of providing an energy-dependent neutron flux effectively equivalent to the first wall of a DT fusion power reactor including:

- Greater than 0.5 liter irradiation volume with greater than 2 MW/m^2 equivalent 14 MeV neutron flux;

- Greater than 70% availability to enable testing to exposures greater than 10 MW-y/m^2 in a reasonable time frame; and

- Flux gradients less than or equal to 20%/cm.

The panel carefully considered the requirements for data from a fusion relevant neutron source to enable the design and construction of next-step plasma devices such as FNSF and DEMO, and adopted the following descriptions of the irradiation effects database needs, which are also reflected in the Technical Readiness Levels presented in Table 3.3.1. A radiation effects database must be constructed that includes the following features:

- To advance the technical readiness level of fusion materials beyond levels 2 to 3 a neutron source that duplicates or effectively simulates the actual fusion neutron environment in terms of primary knock-on spectrum and production of transmutation products, most notably helium and hydrogen, and provides a mechanism for accelerated testing to permit evaluation of a wide range of materials is needed.

- The need for a fusion relevant neutron source is particularly acute because neutron-induced degradation such as volumetric swelling, irradiation enhanced creep, phase instabilities, helium embrittlement and solid transmutation effects become significant beyond ~10 dpa.

- For all candidate structural materials the irradiation effects data base in a fusion relevant environment does not exist, and there is essentially no information on behavior in a fission reactor environment (very low He) at neutron doses > 50 dpa.

- To ensure FNSF will meet mission goals it is essential to construct a database in a fusion relevant environment to 50 dpa in multiple specimens and test those specimens by the end of the FNSF design phase.

- To ensure DEMO will meet mission goals it is essential to construct a database in a fusion relevant environment to 150 dpa in multiple specimens and test those specimens by the end of the DEMO design phase.

The panel assessed the ability to model materials degradation independently through the use of High Performance Computing (HPC) capability and came to the same conclusion as a 2004 International Panel convened by DOE that concluded that the field of computational materials science was too immature for this to be a practical reality [3.3]. Yet, over the years since that report, incremental improvements in both understanding and modeling of materials degradation have occurred at the same time as continual improvements in computing capability. Thus, the panel feels that computational materials modeling through HPC needs to be an integral part of the materials science research and engineering design analysis associated with fusion materials development, even though it is not feasible to short circuit the development path forward by computing alone.

The panel considered three options for obtaining the irradiation data needed to provide a rational basis for design of next-step plasma devices such FNSF and DEMO. Two simple figures of merit for evaluating the efficacy of a particular irradiation source to meet fusion materials development needs are helium production and displacement damage levels. A plot of these parameters for various irradiation sources is presented in Figure 3.3.1. The plot shows that bridging the substantial gap between ITER and a fusion power reactor will require a fusion relevant neutron source like IFMIF or MTS. A further important consideration is the DT fusion relevant irradiation effects data on ITER will only be achieved in the ITER test blanket modules during the DT burning plasma experiments that are currently scheduled to begin around 2028 and will require 2 or more years to achieve exposure levels approaching 1 dpa (depending of the ITER operating schedule). Facilities such as SNS or SINQ will permit exploration of scientifically interesting helium generation and displacement damage regimes in a significantly sooner time frame than ITER, but will not allow timely preparation of a materials qualification database for a DEMO due to flux limitations.

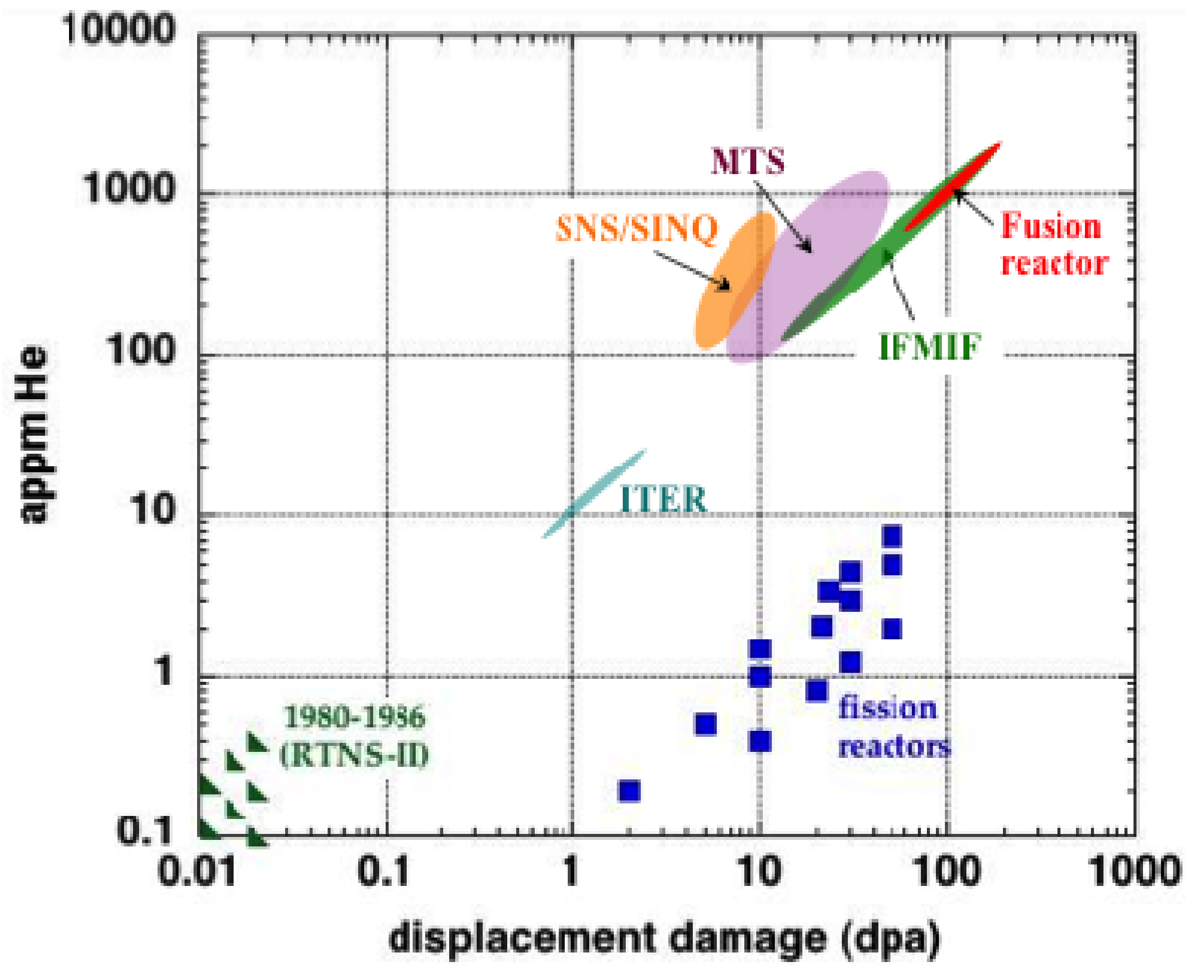

Figure 3.3.1. Summary of helium generation and displacement damage levels in ferritic steels for various neutron irradiation sources for ~2-4 year exposure times.

Options A through C given below are considered the most viable approaches to improving understanding of fusion neutron radiation effects and database development, but each has strengths and weaknesses. The advantages and disadvantages of each option are highlighted in Table 3.3.2 below. For Option A the bulk of the knowledge base is developed by performing experiments in the proposed International Fusion Materials Irradiation Facility (IFMIF). Fission reactors and ion beam facilities are used to explore basic science issues to optimize use of the limited irradiation volume in IFMIF, and computational modeling is used for data analysis and predictions. As noted in Table 3.3.2 this option comes the closest to simulating actual fusion reactor conditions, permits exploration of the effects of irradiation on bulk material properties, and offers the potential for testing of subcomponents to modest neutron fluence. The major concern with this approach is the high cost in comparison with the other choices. To go forward with this option probably requires creation of international partnerships in order to share the cost burden.

Option B was considered to be intermediate between Options A and C in terms of cost and risk. In this option a major upgrade to an existing spallation neutron source would be performed. The proposed Materials Test Station is an example of a possible upgraded spallation source that could significantly contribute toward development of the needed database. An upgraded facility would provide bulk property information at a range of fusion relevant helium to dpa ratios with the operational costs shared between several government agencies. Technical concerns with such a facility include a significant flux of high energy (>14 MeV) neutrons that produces higher than desired gas and solid transmutation rates, the ability to control temperature within a narrow range, the pulsed nature of the irradiation and the high flux test volume could potentially not be adequate to produce the entire regulatory database within the desired time frame. To offset these deficiencies an expanded theory and modeling effort would be needed.

The third option (Option C) relies almost exclusively on use of existing fission reactors (such as HFIR) and ion beam facilities to acquire fundamental radiation effects data with supplemental information obtained by making limited use of upgraded spallation neutron sources such as the SNS or SINQ facilities. This option represents the lowest overall cost, but is the highest risk and will likely take the longest to develop both fundamental predictive understanding of radiation effects in a fusion environment and an adequate regulatory database framework. Thermal and fast fission research reactors exist today and are available for irradiation services; however, the fast reactors are located in Russia and Asia. The BOR-60 Reactor is currently capable of producing 50 dpa in two years of operation [3.4]. Significant advantages of this source are its immediate ability for high-dose uniform irradiation over large samples with a well-characterized spectrum. Bulk properties can be evaluated. Technical limitations are the very low helium and hydrogen buildup, the long times to set-up agreements with non-U.S. facilities and receive samples back, transmutation gas production rates similar to mixed spectrum reactors, and handling activated samples. Fusion relevant helium generation can be achieved in fast reactors by isotopic tailoring of specimens. For example, irradiations in the Fast Flux Test Facility of Fe-Cr-Ni alloys doped with ^{59}Ni enhanced He production due to the ^{59}Ni(n,α)^{56}Fe reaction, resulting in production of helium to dpa ratios from 0.2 to 62 appm He/dpa [3.5]. Extensive utilization of theory and modeling would be required in order to develop predictive models based on limited experimental radiation effects data.

High damage regimes with variable helium to dpa ratios can be investigated in a very short time with ion beam facilities. With multiple beams (self-ion and gas ion beams), displacement damage and gas effects are separable, and different recoil energies can be explored. Ion beams have proven useful in basic science studies to examine unit processes. Furthermore, there is limited induced radioactivity in the samples during ion beam irradiation, which simplifies the post-irradiation characterization. Technical issues include the impact of damage rates that are 100 to 1000 times too high, and an inability to obtain bulk property data because of the small irradiation volumes. Correlation with neutron damage is also a challenge. When compared with the other alternatives, ion beam labs have the lowest capital cost. There are many facilities around the world with multi-ion-beams and ion-beam-TEM capabilities [3.6].

Option A: IFMIF provides the majority of the irradiation data, supplemented with fission reactor data, and basic science ion beam irradiation experiments. Theory and modeling is used to support data interpretation and analysis.

Option B: A major spallation source upgrade (e.g. MTS) provides the majority of the irradiation data, supplemented with fission reactor data, and basic science ion beam irradiation experiments. To correlate data with fusion neutrons an enhanced theory and modeling effort will be required.

Option C: Ion beam facilities, fission reactors, and an upgraded existing spallation source (e.g. SNS, SINQ) provide the majority of the irradiation data. To correlate data with fusion neutrons a significantly enhanced theory and modeling effort will be required.

The panel also examined the potential for plasma-based sources (e.g. Gas Dynamic Trap, and the FNSF itself) to provide the bulk of the irradiation data and concluded that these sources were not suitable, primarily because accelerated single-variable experiments were not feasible, or major investments would be needed to overcome significant technology issues.

Table 3.3.2. Comparison of irradiation source options

Irradiation Source	Advantages	Disadvantages
A - IFMIF	Closely matches fusion conditions in terms of primary knock-on atom (PKA) spectrum, displacement and damage rates Accelerated testing possible Bulk properties Correct He/dpa ratio Subcomponent testing in medium flux test module Provides regulatory databases required for FNSF and DEMO at lowest risk	Highest cost International collaboration needed Not anticipated until next decade Construction and operating funds completely provided by fusion Limited high-flux irradiation volume W spectral shifters used to match PKA spectrum attenuate neutron flux
B - MTS	Upgrade existing facility Intermediate cost Allows fusion relevant dpa & He/dpa irradiation conditions to be explored Bulk properties Operating costs may be largely provided by non-fusion agencies 10x higher neutron production per unit beam power than beam-target sources	Principal mission not fusion materials irradiations Shared ownership with DOE NE & NNSA Significant proportion of high energy (>14 MeV) neutrons Temperature control, solid transmutation, pulsed irradiation High flux test volume may not be adequate to provide entire regulatory database when needed
C – Upgraded SNS or SINQ	Lowest cost Upgrade existing facility Allows fusion relevant dpa & He/dpa irradiation conditions to be explored Bulk properties 10x higher neutron production per unit beam power than beam-target sources	Principal mission not fusion materials irradiations Shared ownership with DOE BES Significant proportion of high energy (>14 MeV) neutrons Temperature control, solid transmutation, pulsed irradiation Lowest rate of progress Slows development of regulatory databases required for FNSF and DEMO More reliance on theory and modeling to correlate with fusion neutrons

3.3.2 Identification of a prime candidate first-wall/blanket structural material is hindered by lack of an integrated engineering design and testing approach for materials development.

Design of full-power fusion devices, from an FNSF to a commercial plant, will require a Design Code that is, to the extent possible, specific to the fusion environment. Codes developed for fission reactors can be used as a starting point for development these Codes, but they fall short in several areas, including high temperature design, design with materials of limited ductility, helium effects, design for transient heat loads, etc. In addition, existing Codes emphasize simplified rules that are conservative to a degree that may not be suitable for commercial fusion power plants. Hence, fusion will require the development of modern, science-based design rules specific to its needs. Some of the research needs for this Code development include:

- Qualified materials with thoroughly documented constitutive laws for all relevant material properties

- Improved models for material damage and failure in a fusion environment

- Experimentally validated design rules for materials of limited ductility (with appropriate helium concentration levels)

- Experimentally validated design rules for high temperature failure mechanisms (with appropriate helium concentration levels)

- Qualified fusion power plant components suitable for use in DEMO or in a power plant.

This effort will require a substantial upgrade of failure models for fusion components. The models must use state of the art constitutive models for material behavior and be capable of addressing all possible failure mechanisms expected in fusion power plants, for both normal and off-normal loads. The primary benefit of this science-based approach is reduced design margins and, ultimately, higher performance components. Progress has been made in this area and captured in the ITER Structural Design Criteria [3.7], but much work remains, especially in the validation of the proposed, fusion-specific rules.

Once these individual failure mechanisms have been addressed, the design effort will require models capable of analyzing full-size components, including all relevant load conditions (transient loads, electromagnetic loads, seismic loads, etc.). In addition, these modeling tools will require coupled physics to address electromagnetic effects, and provide self-consistent heat transfer and thermal-hydraulic inputs to the failure analysis. Progress has been made in this area for fission and fusion concepts, but much more progress is required to enable a true, science-based design approach.

As discussed in the ReNeW report [3.8], this effort will require unprecedented collaboration between the materials science and design communities in order to properly account for all possible failure mechanisms and to optimize designs for high performance and long life. It will also require increased effort in tool development for incorporating materials data into failure analysis and carrying out coupled physics simulations, as well as substantial collaboration with research efforts in other areas, such as PFC and PMI.

3.3.3 Knowledge of the processes controlling tritium permeation and trapping in advanced nanostructured alloys designed to manage high levels of helium is inadequate to ensure safe operation of next-step plasma devices.

Considerable experimental and theoretical work has been performed to elucidate the mechanisms controlling permeation and retention of tritium in fusion reactor materials [3.9-3.12]. Most experiments have focused on single variable type experiments on unirradiated or ion implanted materials to characterize tritium effects. Despite the large amount of work that has been done, there are significant discrepancies between experimental results, which may be due to incomplete characterization of materials and/or experimental conditions [3.13]. While some experiments have been performed to explore the behavior of ion and neutron-irradiated materials much more experimental validation and code development will be needed before quantitative models are available to predict performance under the complex conditions present in the fusion nuclear environment [3.14]. As shown in Table 3.3.1 knowledge of the effects of the fusion environment on tritium permeation and retention is presently at a Technical Readiness Level between 2 and 3.

The focus of the research effort for the next five to ten years should be to develop improved understanding of tritium permeation and retention in three classes of fusion materials; 1) nanostructured alloys designed to manage high levels of helium, 2) materials that permit easy permeation to enable efficient extraction of tritium from liquid metal breeders/coolants, and 3) barrier materials or coatings that are designed to limit or prevent tritium permeation, while potentially contributing to corrosion resistance or electrical insulation. The panel noted that the recent development of novel experimental techniques to simulate the concurrent effects of neutron-induced displacement damage and helium injection [3.15,3.16] coupled with state-of-the-art computational materials science tools [3.17] affords an excellent opportunity to better understand and predict tritium permeation and retention in fusion materials. Many of the scientific questions posed in Section 2.3.3 can be effectively addressed without the need for significant investments in new or upgraded facilities. The recently developed *in situ* helium implanter technology used in existing mixed spectrum fission reactor irradiation experiments [3.15,3.16] provides an attractive means for assessing the effects of helium-displacement damage synergisms, while avoiding the confounding effects of other commonly used simulation techniques. When coupled with existing linear plasma devices such as the Tritium Plasma Experiment [3.13] a near-term opportunity exists to carefully explore tritium permeation and trapping in a variety of fusion materials and, in particular, advanced nanostructured alloys. The panel also found that materials testing in ITER, due to the low neutron dose expected, will not provide significant information for resolving tritium issues.

The panel concluded that computational materials science is currently not sufficiently advanced to eliminate the need for carefully controlled experiments. The proper role of computation is to provide the means to re-evaluate existing data, optimize the design and execution of new experiments, and effectively interpret the results from those experiments. While recent advances in multi-scale materials modeling should be utilized to the maximum extent possible, the research emphasis should be on experimental characterization of tritium effects.

Full development of quantitative models of tritium permeation and trapping behavior for plasma devices beyond ITER, such as FNSF or DEMO, will require a fusion relevant neutron source so that material behavior can be studied to high neutron dose at appropriate irradiation temperatures with the correct levels of transmutation produced helium. It is anticipated that a fusion relevant neutron source will not be available for at least ten years due to its technical complexity and relatively high cost compared to non-nuclear test facilities.

3.3.4 Current understanding of the thermo-mechanical behavior and chemical compatibility of structural materials in the fusion environment is insufficient to enable successful design and construction of blankets for next-step plasma devices.

Considering the current limited scientific understanding of corrosion mechanisms in fusion-relevant coolants and materials at elevated temperatures, the suggested path to understanding the more complex corrosion behavior involves generating baseline experimental data under well-controlled conditions that can be used for the development of corrosion science models. Once the baseline behavior is established, then the more subtle complications can be addressed and understood. Proceeding immediately to complex corrosion tests at this stage is not recommended until a robust understanding of corrosion mechanisms is obtained for model systems. A particular near-term priority is improved understanding of corrosion mechanisms in PbLi coolants.

No corrosion information is currently available above 550°C in flowing PbLi coolants. Single-material (ferritic steel and SiC) thermal convection loop testing is needed, followed by investigations of mixed material effects including SiC, coatings, etc. since the corrosion kinetics in many real-world systems is strongly influenced by impurity and interstitial solute transfer which often act as chemical catalysts. For liquid metal corrosion, data from a thermal gradient is needed to understand the underlying dissolution/precipitation kinetics. A progression in maximum loop temperature would create important baseline information on corrosion kinetics and thereby determine the maximum allowable DCLL temperature based on temperature-controlled mass transfer considerations. Currently, helium cooled PbLi concepts are designed below 500°C because none of the structural steels can operate above that temperature without significant dissolution. New strategies such as the use of SiC flow channel inserts (high temperature DCLL) or new alloys and/or coatings need to be evaluated to determine if there is a path forward to higher blanket temperatures with a liquid breeder like PbLi. In parallel, studies to investigate the role of coolant turbulence (including magnetic field effects) and velocity (solute transfer through the coolant boundary layer) should be initiated using pumped loop experiments. In the longer term, the potential role of complex stress states and fusion neutron irradiation on enhancement of solute diffusion and localized corrosion at features such as grain boundaries (irradiation assisted stress corrosion cracking) should be investigated, but only after an improved understanding of corrosion mechanisms in well-defined model environments has been obtained.

The experimental testing and accompanying model development should be approached in a systematic manner in order to facilitate understanding. An operating blanket is an extremely complex compatibility environment including the potential for magnetohydrodynamic (MHD) effects and radiation to alter compatibility. However, each complicating factor must be evaluated in a controlled manner for understanding to develop. Initially, well-controlled experiments are needed to understand mechanisms and relate results to fundamental theory. For example, it should be possible to relate the dissolution rate of Fe and Cr in PbLi to their temperature-dependent solubilities. Once the basic mechanisms are understood, predictive models can be used to design next-generation experiments to validate the models. The subsequent steps would involve investigation of more complicated issues such as radiation and magnetic fields, perhaps in a joint facility where other issues (e.g. tritium transport/removal, MHD effects) are addressed simultaneously. With a well-understood baseline, these additional effects on compatibility will be easier to understand and quantify in more sophisticated predictive models.

3.3.5 Transformational advances in fabrication and joining technologies may have the potential to provide high-performance materials with properties that enable construction of fusion power systems that fulfill safety, economic and environmental attractiveness goals.

Additive manufacturing is a promising new methodology for fabricating high-performance near-net shapes, and could favorably impact the development of fusion energy components and systems. In additive manufacturing, material is deposited only where needed and then fused by laser, electron, or other power sources in accordance with computer aided design 3-dimensional fabrication drawings. In contrast, conventional manufacturing is the process of fabricating a mill product and then machining the component from the material. The benefits are high yields, specific material properties where needed, and the ability to develop shapes and geometries that cannot be machined.

Applications in fusion include high-temperature coatings and other hybrid systems, complex geometries for cooling, intricate structural members where material yields are low, prototyping, and affordable methods to rebuild components after degradation or consumption. Candidate materials for various applications in fusion reactor design have both advantages and challenges to performance; for example, the material that is the best selection for thermal loading may also be susceptible to irradiation damage. Additive manufacturing allows for the potential of metallurgical bonded coatings to use the materials' benefits but reduce or eliminate the disadvantages via a hybrid approach. Additively manufactured parts are usually built to geometry with minimal surface machining required. A recent manifold assembly built for a hydraulic system via additive manufacturing was able to properly function with less than five percent of the structure machined away; conventional manufacturing would have had a yield below 50%. Rapid prototyping using additive manufacturing enables a reduction in development times and provides the ability to precisely explore a range of engineering design options, with fabricated parts often being available in hours or days rather than weeks. Many functional problems that were not evident in computational design can be realized in prototyping. Lastly, damaged or degraded large costly components may be rebuilt or fixed using additive manufacturing practices without complete fabrication of a new part.

A large ongoing effort in additive manufacturing is the development of closed loop systems that could provide real time information on the deposition quality and temperature layer by layer. This approach is completely different in thought to conventional manufacturing where non-destructive evaluation is limited and most testing must occur after fabrication. The ability to evaluate material as deposited could allow for immediate repair (decreasing scrap rate), the potential of "metallurgical arrangement" or changing microstructures for preferred properties in specific areas by changing power level, raster rate, etc., and/or the ability to qualify components for fusion applications while being built.

Exploratory fusion materials research to investigate the potential benefits of solid-state and molten-metal additive manufacturing techniques should be initiated in the near term to examine the potential benefits and limitations of this emerging fabrication technology. In particular, analyses to explore potential tradeoff benefits in hybrid fabrication methods (e.g., utilizing additive manufacturing for some particularly demanding high heat flux regions on a component and conventional manufacturing methods on a tailored radiation-resistant bulk material for the remainder of the component). The role of additive manufacturing techniques to reduce the number of weldments in blanket components should also be investigated.

3.3.6 High Temperature Superconductors and Magnet Systems

It is well understood that the performance and economics of Magnetic Fusion Energy is highly leveraged by magnet technology. Thus there will be a compelling motivation to continuously explore improvements in superconducting magnet capability (e.g. higher fields, higher tolerance to neutrons, lower manufacturing costs, etc.) as well as adapting the latest improvements in strand technology and new high temperature superconducting material, HTSC [3.18].

The Nb_3Sn technology used in International Thermo-nuclear Experimental Reactor (ITER) is operating close to the limits of current and field it can survive due to the enormous Lorentz force exerted on the strands, which is evident in the persisting issues of degradation in conductor performance with field cycling. Although DEMO could be built with Low Temperature Superconducting (LTS) technology, an economical commercial reactor would still require the development of an alternative technology that would expand the range of magnetic field, current density and temperature. To this end, the ReNeW report [3.8] proposed an "integrated program of advanced magnet R&D focuses on developing High Temperature Superconductor (HTS) materials and magnet systems, which offer enormous potential for magnetic fusion energy research experiments, and potentially transformative technological innovation." The major question posed by the ReNeW report in the area was "can high-temperature superconductors and other magnet innovations be exploited to advance fusion research?" Recent advances in the application of HTS to high field magnets since the publication of these reports indicates that robust progress is being made towards this goal.

In ReNeW [3.8] the limits to the application of HTS technology that reflected their early stage of their development were listed as:

1) Cost
2) Performance
3) Piece length
4) Strength
5) Production capacity

To these we should also add a sixth limitation of current HTS technology:
6) Availability of cable

1). *Cost* continues to be as issue for the wide-scale application of HTS technology to fusion (in 2012 REBCO coated conductor ranges from $400-600/kA·m, 77 K self field, compared to ITER Nb_3Sn at $15/kA·m for 12 T, 4.2 K). It is very clear, however, that production costs for HTS conductors, Bi-2223 ($(Bi,Pb)_2Sr_2Ca_2Cu_3O_{10-x}$), Bi-2212 ($Bi_2Sr_2CaCu_2O_{8-x}$) and REBCO ($REBa_2Cu_3O_{7-x}$) are improving significantly as the conductors go into applications and scale up occurs. HTS conductors have unique capabilities that can offset additional costs, for instance in ITER, where 60 HTS (Bi-2223) current leads will be used to provide the cold to warm transition on the magnet power distribution system [3.19], an application which follows their successful installation in the Large Hadron Collider (LHC) at CERN.

2). The key *performance* capabilities of HTS that surpass Nb-based LTS materials are an upper critical field at 4 K, which is 4 times that of Nb_3Sn (~120 T vs. 30 T) and the ability in the case of REBCO conductors to allow ITER-strength magnetic fields at temperatures perhaps as high as 50 K, thus greatly reducing the cryogenic power load. The last 5 years have established several benchmarks for HTS coils that have shown their capability to exceed the maximum field of any Nb-based magnet (1 GHz NMR system operating at 23.5 T), first 27 T [3.20], and then more recently a layer-wound REBCO coil producing 35.4 T [3.21]. These coils, like the 32 T all-superconducting, user magnet being constructed at the National High Magnetic Field Laboratory [3.22] have all used 4 mm wide, single-strand REBCO tape, which has a very strong Hastelloy substrate that supports axial stress very well. The most recent advance has come from a set of two solenoids made at BNL , one of the coils achieved almost 10 T in an all-HTS solenoid, matching the previous record. The second coil achieved 16 T in an all-HTS solenoid, 50% higher than the previous record [3.23]. The continued advance in high field HTS magnets is illustrated in Figure 3.3.2 [3.24].

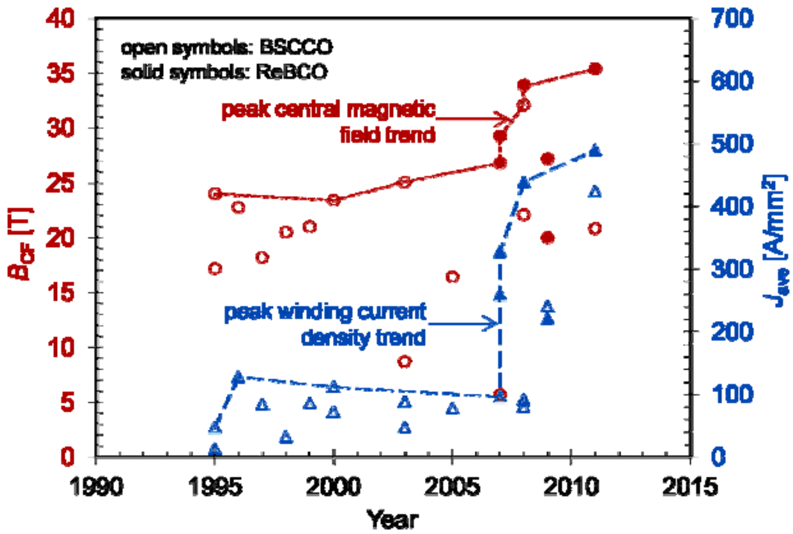

Figure 3.3.2. Highest fields and overall winding current densities generated by HTS R&D magnets during the last 16 years. The two conductors featured here, REBCO and round wire Bi-2212, complement each other, one being more suitable for certain applications than the other. The ideal HTS conductor would in fact have the round wire attributes of Bi-2212 and the high flux pinning capability and high irreversibility field of REBCO, properties which would allow the very high current cables needed for DEMO or other large fusion machines [3.24].

3). The *piece lengths* for HTS tapes are typically 150 m or less (although at least one manufacturer is currently selling 300 m lengths). Development of a robust >1 km lengths production capability will be necessary for before HTS technology can be applied to fusion magnets.

4). The *strength* of HTS tapes compares favorably with Nb_3Sn (the UTS of typical REBCO tapes is of the order of 750 MPa to 1 GPa). The strain dependence of critical current density in REBCO tapes is also lower than for Nb_3Sn but operation at higher fields and currents will increase the Lorentz force on the tapes and will need to be mitigated.

5). The *production capacity* for HTS is likely to be scalable to the demands of fusion magnets in the 10+ year timeframe, especially if a market for transmission lines develops.

6). A key issue for large fusion magnets is the need to produce very high current carrying cables (e.g. 60 kA as specified for ITER) that allow fast ramp and protection under fault conditions. Because the emphasis of HTS conductor manufacturing so far is still at the strand level, cable work is less far advanced. Four kinds of cable concept dominate the thrust of HTS cable development. One is the “classical” Rutherford cable used in accelerator magnets, now being pursued strongly for future high energy physics machines like an LHC upgrade and a Muon Collider. Such cables require a round (or perhaps a very low aspect ratio) wire, a requirement that can be met by Bi-2212 round wire, but not by Bi-2223 or REBCO tape conductors. For fusion applications there may be much greater interest in any case in REBCO than either Bi conductor because neither offers the possibility to generate >12 T fields at 20 K or higher. Three types of cable concepts are being pursued for REBCO, one being a nested barber-pole concept (Conductor On Round Core Cable or CORCC) [3.25], the second being the Roebel style cable [3.26], complemented by the Twisted Stack approach pursued at MIT [3.27]. These designs demonstrate that cables of up to 10 kA can be made and, for the CORCC, that small coils can be made with it reliably operating at 4 K and 20 T.

3.4 Harnessing Fusion Power

3.4.1 Overview of key fusion technology aspects

Fusion power is captured in an integrated first-wall/blanket system that surrounds the plasma. This system must operate at a high temperature to efficiently convert fusion power into electricity or other possible end uses. Furthermore, tritium fuel has to be bred by capturing fusion neutrons in lithium. In addition to the in-vessel first wall/blanket, systems associated with harnessing fusion power include additional shielding of various components (e.g., superconducting magnets), heat transport loops, coolant chemistry control, heat exchangers, and systems to recover and process bred tritium from the blanket and tritium in the plasma exhaust.

This area is illustrated by Figure 3.4.1 which shows fuel cycle and energy capture streams and functions. The functions are organized into the following three areas:

- **Tritium breeding and energy capture** (red): Utilize and manage fusion neutrons by breeding tritium to replace burned fuel, convert neutron energy to heat, and shield equipment outside the reactor from nuclear damage. This region is exposed to high neutron fluence.
- **Blanket tritium and heat extraction** (green): Extract tritium and heat from the breeding blanket—send tritium to the purification and recycle loop, and convert the heat to electricity. This region is characterized by high temperatures and amounts of tritium.
- **Tritium purification and recycle** (blue): Purify unburned tritium from reactor exhaust and tritium extracted from the breeding blanket in preparation for fueling the reactor. This region must handle large tritium inventories and must prevent tritium release to the environment.

These functions must be performed with the following high-level requirements:

- **Effective**: Each function must perform with demanding operational requirements using, in many cases, first-of-a-kind technologies
- **Safe**: Besides normal industrial safety needs, DEMO will have the added challenge of operating with large quantities of nuclear material
- **Reliable**: Each of these functions must have high reliability since they must be operational for DEMO to perform successfully

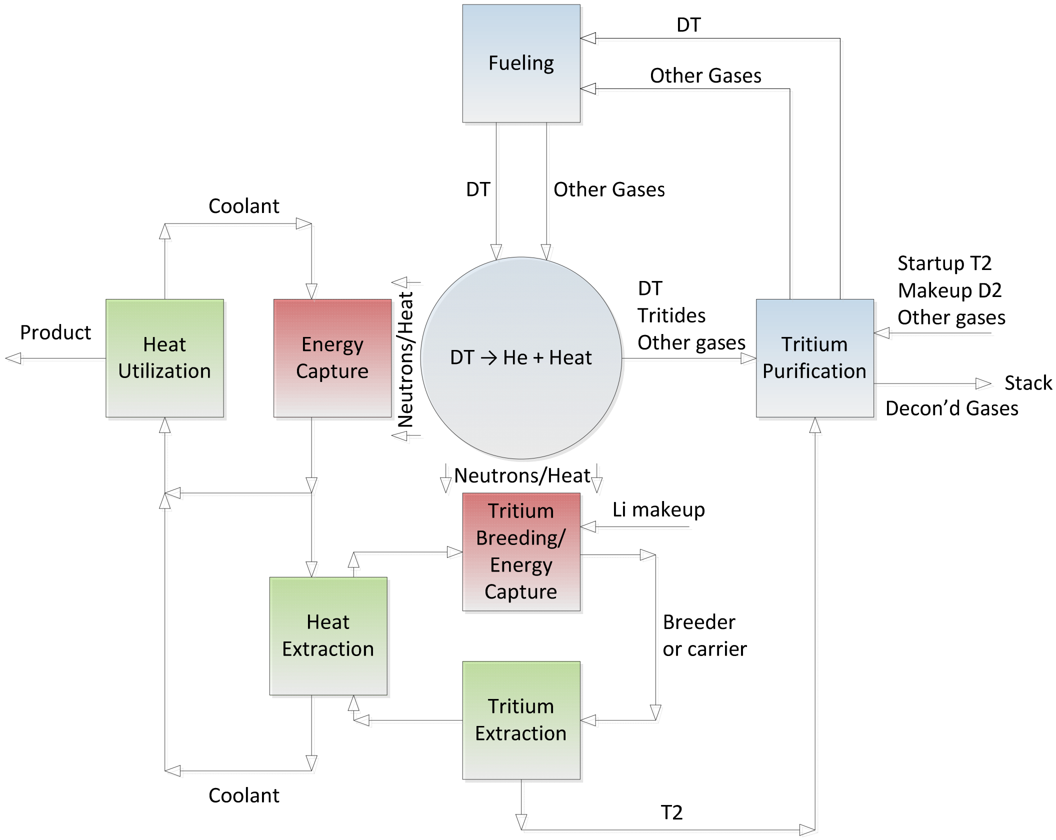

Figure 3.4.1: Key Harnessing Fusion Power functions including (red) tritium breeding and neutron energy capture in the blanket, (green) extraction of fusion energy and bred tritium, (blue) purification and recycle plasma exhaust

3.4.2 State of Development

The attractiveness of a fusion system (economics, safety and environmental features) is mainly determined by materials and design of systems associated with harnessing fusion power. But, at present these systems are at a low technical readiness level with high uncertainty as to the performance of envisioned solutions and material systems.

We also note that US does not have an integrated technical plan and strategy to develop the scientific and engineering basis for power extraction and tritium breeding and processing systems.

The significant challenges in ultimate development of power extraction and tritium breeding components for fusion are numerous ranging from lacks of fundamental data (single-effect experiments) to the need to develop validated, integrated simulation capabilities meeting requirements for design and analysis of nuclear systems. The efforts to meet these challenges are hampered due to a lack of resources and test facilities.

To date, no fusion blanket or power extraction system has ever been built or tested. Fundamental experiments and modeling on the thermofluids of liquid metal and helium coolants have been performed that provide some basis for the conceptual design of blanket systems. Use of prototypic liquid metal coolants, reduced activation steels and silicon carbide in heat transfer, MHD, and even corrosion test mockups is rare.

Fundamental tritium breeding rates are known from basic cross-section type experiments and neutron transport simulation capabilities are quite advanced, lending confidence in the potential of fusion to breed sufficient tritium. The uncertainties in tritium breeding rather stem from to a greater degree from the uncertainty in the blanket design itself, for instance, the quantity of structural material necessary to adequately support the blanket against the significant and not well-characterized thermal, mechanical, particle and electromagnetic loads.

While we expect heat transport systems in DEMO will utilize relatively conventional materials and technologies to the largest possible degree, the fusion environment will likely result in many non-conventional challenges such as liquid metal chemistry and isotope control and high temperature, liquid metal compatible heat exchanger systems. Such systems have not been developed or tested. Similarly, there is no tritium extraction technology yet tested which has high tritium recovery efficiency from PbLi.

Development in this area or plasma exhaust processing has benefited from several fusion prototypic experiments and related weapons tritium experience. This has laid a good technological basis for the current ITER tritium system design work. When ITER begins operations with tritium in the latter half of the 2020's, valuable DEMO-relevant information can be collected, though it is noted that ITER will not directly scale up to DEMO requirements in regards to reliability, duty factors and controllability. Furthermore, ITER will not perform substantial work on processing of bred tritium.

3.4.3 Development Paths

3.4.3.1 Analysis

Development Path Analysis Methodology: To determine purposeful development paths and potential roles for key facilities for each of the three areas listed on Figure 3.4.1, the major performance phenomena associated with each area were identified. These are listed on the left-hand side of Table 3.4.1. On the right-hand side of the table the DEMO requirements for each parameter are listed. In the middle of the chart, the columns summarize parameter values associated with major phases or facilities associated with fusion development. As much as possible, performance requirements were quantified although qualitative descriptions were sometimes necessary. Columns were arranged so that a reasonable set of parameters were advanced from one program phase to the next. Highlighting was used to indicate key regions of parameter advancement, with red shading to denote TRL 1-3, yellow for TRL 4-6, and green for TRL 7-9. Advancement from left to right proceeds from basic experiment through development phases to DEMO with parameters progressing by modest increments in TRL values (capability jumps that require improvement in TRL of more than one or two levels are considered to be high risk activities). Performance parameters (rows) are ordered so that early advancement is at the top and later advancements are at the bottom. The shading then serves to illustrate development progression. Computation is used throughout, as will be explained later, to progress development from left to right.

Tritium Breeding and Energy Capture: The first section of Table 3.4.1 describes the development pathway for a lead-lithium based breeder blanket called the Dual Coolant Lead Lithium (DCLL) system. In the DCLL, neutrons are predominantly multiplied by and then captured in the PbLi breeder/coolant that is flowing sufficiently fast to remove both the bred tritium and the majority of the fusion energy from the reactor. A second helium coolant is used to cool the structures especially the first wall.

Table 3.4.1 columns show a progression from separate-effect experiments on the nature of PbLi as both a heat transport fluid and tritium breeder. Separate-effects tests include thermophysical properties and nuclear cross-sections. This work has been largely completed. A notable exception is aspects of tritium chemistry in PbLi such as solubility and diffusivity which have been difficult to measure accurately and consistently.

Development experiments include aspects of flowing PbLi and helium such as the interactions with magnetic field and heat transfer behavior. Flow rate, pressure and temperature parameters of current experimental test facilities (PbLi flow in MTOR, UCLA; and helium coolant flow PMTF, SNL) are shown in the table with quotes to indicate the degree to which necessary DEMO conditions are currently being simulated. These experiments are being used to help develop fundamental MHD and heat transfer modeling tools as well as performing tests on material system corrosion including silicon carbide and ferritic steels in PbLi. The temperature of flowing tests is currently limited in both the PbLi and He test facilities. Also, the magnetic field and magnetic volume in these laboratory facilities are also limited to that typical of iron core gap magnets—about 2T and 15 cm gap distance.

In order to study the effect of more integration, both in terms of the loading conditions and blanket functions and materials, a multi-effect Blanket Thermomechanical, Thermofluid MHD Test facility (BT3F) is envisioned. This facility should bring together simulated surface and volume heating and reactor like magnetic fields with test mockups having prototypical size, scale, materials operated at prototypical flow rates, pressures and temperatures for extended periods. The experimental program should be centered around understanding performance and failure rates as well as modes and effects under increasingly integrated conditions. An out of pile facility such as BT3F can be much more flexible and easily instrumented than tests in the nuclear environment. However, there does not appear to be a way to fully simulate prototypical nuclear heating in the volume of flowing PbLi coolant. Internal immersion heaters can disturb MHD flows in significant ways, and so clever experimental methods and testing strategies will be needed to elucidate DEMO-relevant behavior.

There is also an opportunity to utilize the nuclear field in ITER to perform blanket module experiments as part of the ITER Test Blanket Module (ITER-TBM) program. There are currently test spaces reserved in 3 midplane ports to allow deployment of TBM experiments, with space reserved in the port cell, cooling water building and tritium building for independent helium and lead lithium coolant systems. The magnetic field in ITER is DEMO-relevant, and the pulse length and neutron wall load in the later DT phase is sufficient for the blanket system to come to thermal equilibrium in a single pulse. This presents the opportunity to use ITER to do limited studies of prototypic volumetric nuclear heating and blanket thermofluid MHD behavior augmenting studies in the BT3F. Studies of beginning of life irradiation damage effects in ceramic insulators in the blankets, as well as tritium production and transport will also be possible but of more limited value due to the pulsed nature of ITER and the limited total fluence.

The need for an FNSF facility for the full testing of in-vessel blanket systems is clear. FSNF will bring together relevant neutron wall load, volumetric heating, and sufficient fluence to study the synergistic effects of material degradation together with overall blanket operation. FNSF will allow a significant increase in the number of blanket modules, availability targets and continuous operating time. While such tests over the life of FNSF will be necessary to fully establish the engineering feasibility and tritium self-sufficiency of fusion, it should be emphasized that blanket systems must be fully tested to the highest possible degree of confidence prior to proceeding to the more challenging and expensive nuclear environment of FNSF.

As described in the preceding section, DEMO is the goal of the other development steps. DEMO will combine all of these technologies into a production environment where increases in neutron wall load and inboard magnetic field must be accommodated, in addition to duty factor and availability.

Blanket Tritium and Heat Extraction: The principles of this development path are essentially the same those described above. This area is included in the first section of Table 3.4.1. It is important to note that, even at the early development phases, to-date there has been very limited progress in blanket tritium and heat extraction research. Considerable work is needed in these phases to develop the experimental foundation for a potentially attractive system. The goal will be to ready one system concept with a number of possible variations that are ready for further development.

Addressing gaps in this area will require a Tritium Breeding and Extraction Facility (TBEF). This facility is envisioned to include a neutron source and a breeding module coupled with a tritium extraction system. A potential configuration is a liquid metal breeder loop in a fission test reactor, but other configurations might be superior. Components would be as small as possible while being large enough to test key performance parameters which can ultimately be scaled to DEMO. The compact size of this facility would make it practical to test multiple variations and operating conditions for the leading breeder/extraction concept. Test campaigns of approximately a month are envisaged with the goal of measuring steady state performance. Both tritium breeding and extraction will be integrated to test overall system behavior. It is anticipated that basic data on heat extraction systems will also be collected on TBEF. TBEF data will, however, not be taken in a fusion reactor relevant environment.

The ITER TBM will provide an opportunity for testing in a fusion reactor relevant environment. By deploying blanket tritium and heat extraction systems on ITER, integrated operations data can be collected. But the ITER TBM environment will have considerable restrictions on experimental flexibility and on the ability to collect precise data.

The next phase of development will be on FNSF. In moving to FNSF, TBEF and ITER TBM experience will require a large scale-up. This is expected to be quite challenging and will place considerable pressure on the quality of experiments and data from TBEF. On FNSF, the tritium extraction system will be integrated with the tritium purification and recycle system for the first time in a practical environment. This environment will be very important for collecting heat extraction data at prototypic temperatures and with relevant materials to lay the basis for DEMO.

Tritium Purification and Recycle: The second section of Table 3.4.1 shows tritium purification and recycle development starting with “basic experiments” which are small, single-effects tests using hydrogen and perhaps a small amount of tritium. Experiments of this type address fundamental phenomena such as chemical reaction rates, separation factors, hydride storage capacities, materials compatibility, analytical techniques and confinement effectiveness.

The second phase is “development experiments” in which tritium processing sub-systems are conceptualized and tested. For instance, the “fuel cleanup” is a sub-system of the overall tritium processing system. It must recover tritium from impurities such as water and methane. All hydrogen isotopes must be highly purified prior to sending gas to the next sub-system, i.e. isotope separation. In this phase many concepts are tested and those that show promise are developed further. Testing the first two phases do not challenge practical concerns such as duty factor and availability, so these columns in the table are coded with “NA” for “not applicable.”

The third phase of development is “integrated experiments.” This phase conducts tests of interconnected sub-systems. The Tritium Systems Test Assembly collected these type of data with almost all tritium processing sub-systems as an integrated system. This was performed without the production demands of an actual fusion reactor, so there was ample opportunity to test various sub-systems and configurations. Other data of this type have been collected at JET and TFTR in tritium systems connected to DT fusion machines. These experiments collected data in the fully integrated fusion environment. This phase of experiments not only developed understanding of fundamental scientific parameters, but also measured performance of duty factors and availability. But, as shown in the table, all of this work was performed at scales much reduced from DEMO.

In this area, the next phase of development involves ITER. As shown in the table, this will involve a major scale-up of flow rate, tritium inventory, facility size and other parameters. It will be a major undertaking to meet these requirements. Notably, at this scale, tritium processing sub-systems become quite expensive, and this will likely result in strong incentives to develop alternate, lower-cost technologies that have potentially superior performance. Additionally, the nature of ITER operations will afford little opportunity for technology development and refinement. To address these issues, a Fuel Cycle Development Facility is envisioned. This would not be a tritium facility. Rather, it would keep construction and operations costs and complications to a minimum by only operating with protium and deuterium. In this flexible environment, two technologies would be developed and partially qualified. The final tritium qualification would occur on ITER or FNSF.

The next phase of development involves FNSF. This is the point where tritium purification and recycle will be integrated with tritium breeding and tritium/heat extraction. Operated at near-DEMO conditions, this will be a major step in Harnessing Fusion Power development. It is envisioned that FCDF would also be used at this point to provide a development path optimized for time and cost.

DEMO, the right-most table column, is the goal of the other development steps. DEMO will combine all of these technologies into a production environment where challenging parameters such as duty factor and availability will be of high importance.

3.4.3.2 Experimental Approach

Given the state of technical readiness level of power extraction and tritium breeding and recovery systems, the US should first develop an integrated technical plan for this area. Such a plan should address this challenge through a mix of single-effect experiments, multi-effect experiments in non-nuclear environment, fast neutron sources and fusion devices together with a comprehensive simulation capability.

Technical challenges in this area depend strongly on the choice of material, first wall/blanket concept, and operating environment (temperature, stresses, etc.). As such, the development plan for harnessing fusion power systems is concept specific. The US has developed a potentially attractive family of first wall / blanket concepts based on the use of lead-lithium eutectic alloy as a breeder/coolant, separate gas cooling of reduced activation ferritic steel first wall and structure, and the use of thermal / electrical insulating inserts based on silicon carbide. But considerable feasibility issues remain due to the lack of some fundamental data and technologies. It is essential that the development strategy focuses on resolving feasibility issues in logical progression. This is realized though experiments performed in the following order.

Single-Effect Experiments: These are small experiments to address fundamental scientific issues. Example of such experiments include test of tritium systems using hydrogen and perhaps a small amount of tritium to address fundamental phenomena such as chemical reaction rates, separation factors, hydride storage capacities, materials compatibility, analytical techniques and confinement effectiveness. Similarly, experiments that are designed to uncover complex heat transfer in the first wall, mass and heat transfer in LiPb in a magnetic field, chemical compatibility and corrosion of LiPb/steel/SiC systems, etc. These type of experiments are specially suited for the university environment and help train the next generation of scientists and engineers. Priority should be given to those experiments that address the feasibility of the concept.

Multiple-Effect Experiments: These are "proof-of-principle," medium-size experiments to address partially integrated effects in a realistic geometries and prototypical material (test articles). Examples include 1) integrated experiments of a blanket and/or fist wall "unit cells" under relevant heat loads and flow rates to uncover thermo-physical effects 2) separate experiment with such unit cells to validate chemical compatibility and corrosion, 3) small loop to test tritium extraction from the breeder, 4) a "fuel cleanup" experiment to recover tritium from impurities such as water and methane while producing high purity hydrogen isotopes.

Partially-Integrated Experiments: These are more complex "proof-of-principle" experiments with test articles having prototypical size and scale with materials operated at prototypical flow-rates, pressures and temperatures for extended periods. Experiments are carried under simulated load conditions (e.g., simulated heat and particle loads, neutrons from fission and/or accelerators) and without the production demands of an actual fusion reactor, so there was ample opportunity to test various sub-systems and configurations. Experiments of this type needed for Harnessing Fusion Power would utilize midscale facilities such as BT3F, BTEF and FCDF which are described below.

Fully-integrated experiments: These are validation experiments performed in a realistic fusion environment. The need to perform integrated validation of the predicted performance of all systems in a true fusion environment is the last step prior to proceeding to a fusion DEMO. ITER is likely to be the first fusion device capable to produce significant nuclear heating, but has limitations in pulse length and total fluence. While there is disagreement as to the benefits of performing first test blanket experiments in ITER, there is a consensus in the US that a facility such as the Fusion Nuclear Sciences Facility (FNSF) will be necessary to bring together larger scale integration, multi-module interactions, uniformity of nuclear field, and middle to end of life irradiation damage. A major element of the technical plan for harnessing fusion power is to delineate the US position on ITER-TBM. In addition, developing a detailed mission and requirements for the FNSF facility is essential and should receive special attention.

Theory, modeling and computation: Modeling and simulation is a core activity necessary for progress along the development path. One motivation is to minimize testing and development costs. This is especially true for Harnessing Fusion Power development, as a fully relevant fusion environment cannot be produced in the laboratory or in test facilities prior to FNSF. Furthermore, not only is the cost of experimentation on an FNSF high, experiments in such complex environments are difficult to understand and interpret and impossible to fully instrument. This requires an integrated theory/modeling and experimental R&D program aimed at developing the best predictive capability before the operation of FNSF. In this case, a central role of a FSNF facility is the validation of the comprehensive simulation capability, as well as the discovery and resolution of un-anticipated synergistic effects.

The development plan for Harnessing Fusion Power should incorporate such as "science-based engineering" approach with theory and modeling playing a major role in guiding and interpreting the R&D. Accurate understanding of fundamental physics principles through careful single-effect experiments is an essential first step. Integrated models can then be used to plan multiple-effect and partially integrated experiments such that they would highlight multi-physics interactions.

A major part of developing such a predictive capability, detailed sub-system design and analysis is necessary to understand and resolve conflicts among component constraints. Periodic integrated power plant modeling is also needed to ensure that "problems" are not transferred to another system. Lastly, modeling results used in the licensing process for a future nuclear system will need to come from codes validated and verified by the regulatory agency involved.

Significant investment is needed to develop, validate and verify, and maintain these computational capabilities.

3.4.3.3 Key Facilities Summary

The development path for each of the three areas was summarized above. Following these paths will require key facilities which were introduced above and are summarized in this section. Value engineering principles were used in selection of the facilities. For instance, 1) maximum use is made of already planned facilities such as ITER, 2) non-nuclear facilities are used whenever possible, and 3) as appropriate, facilities serve more than one development area.

BT3F (Blanket Thermomechanics Thermofluid Test Facility): This facility brings together simulated surface and volume heating and reactor like magnetic fields with test mockups having prototypical size, scale and materials operated at prototypical flow rates, pressures and temperatures for extended periods. The experimental program will be centered around understanding both performance and failure rates, modes and effects under more and more integrated conditions.

TBEF (Tritium Breeding and Extraction Facility): This facility will irradiate breeding blanket modules with neutrons and extract the resulting tritium. This facility will be as small as possible, but large enough to demonstrate the essential integrated features of these functions. Tritium extraction technologies will be developed to TRL level 2 - 4. It will be this facility's job to bring tritium extraction to the level that TSTA brought tritium processing. This facility can also be used to collect earlier TRL information on power extraction.

FCDF (Fuel Cycle Development Facility): This is a hydrogen/deuterium-only facility. Thus, it is relatively cheap to build and operate. It runs alongside ITER and provides a flexible, non-production environment for "tritium" science and technology (but again, without tritium). Sub-systems are integrated so that higher level TRL issues can be addressed. This facility is used to develop and qualify new technologies and approaches which leads to lower-risk deployment in other facilities such as ITER where final qualification with tritium is done. The flexible development environment of this facility (including the "luxury" of breaking things) together with ITER/TBEF/FNSF tritium data will provide an optimized environment to reach TRL levels 5/6. This facility can also be used as an operator training facility.

ITER: to develop/demonstrate/qualify core tritium processing technology, but in a production environment, so development opportunities will be limited. Nonetheless, "hard tritium" experience (confinement, beta catalyzed effects, isotopic effects, detritiation performance, chemical analysis, accountancy, authorization basis) will be substantially extended beyond present experience and take things close to where this area needs to be for FNSF/DEMO. ITER will take many aspects of tritium processing from TRL 3/4 to 5/6.

FNSF: This facility will do for tritium extraction what ITER does for tritium processing. i.e. it will bring tritium extraction from TRL 2-4 to 6/7. It will also have a full tritium processing system which will bring the tritium processing TRL from 6 to 7. Thus, both tritium extraction and processing will reach TRL 7. FCDF will be run "alongside" (not necessarily at the same location) FNSF. FCDF will serve the same role for FNSF that it does for ITER. i.e. it will serve as an integrated, flexible, non-tritium test bed for development of tritium extraction technologies. FNSF will be used for the "hard tritium" development/demonstration/qualification of both tritium extraction and tritium processing. This facility will be responsible for developing power extraction and utilization systems from TRL level 4 to 7.

Existing Supporting Facilities: There will be a number of test facilities supporting these development paths. Some will run with and others without tritium. They will utilize blanket relevant breeders and heat transport systems. Examples of such national and international facilities include the Safety and Tritium Applied Research facility at INL, Tritium Laboratory Karlsruhe (KIT), Tritium Processing Laboratory (Tokai), Hydrogen Processing Laboratory (LANL) and the Active Gas Handling System (JET). Beyond these examples there are many others that are making and will make important contributions. These facilities will address specific, focused, mostly non-integrated topics.

3.4.4 Near-term next steps

In the near term, the following tasks and research and development activities should be pursued:

1. Develop a detailed technical plan and strategy to develop the scientific and engineering basis for power extraction, and tritium breeding and processing systems. In particular,
    1. Develop a detailed mission and requirements for the FNSF facility,
    2. Identify associated science and technology risks, and
    3. Delineate necessary specific R&D including the role of ITER-TBM.
2. Perform single-effect and multiple-effects experiments primarily aimed at resolving feasibility issues of the DCLL blanket.
3. Initiate a comprehensive theory/modeling program aiming at developing an integrated predictive capability.
4. Start planning for partially-integrated facilities

Specific technical research tasks with near-term and high-impact opportunities are described in detail in Chapter 4.

Table 3.4.1: Harnessing Fusion Power-- Performance Parameter Progression toward DEMO

Tritium Breeding & Energy Capture/Blanket Tritium & Heat Extraction

Key Experiments	Basic Experiments	Development Experiments (e.g. PMTF, MTOR, others)	Partially-Integrated (e.g. BT3F, TBEF)	ITER-TBM	FNSF	DEMO
Performance Requirement						
Helium Temp/Flow Rate/Pressure per blanket module	2) gas bottle discharge or air simulant	3) 600C, 0.1 kg/s, 4 Mpa	6) 500C, 1.5 kg/s, 8 MPa	5) 500C, 1.5 kg/s, 8 MPa	8) 500C, 3 kg/s, 8 MPa	9) 500C, 3 kg/s, 8 MPa
PbLi Temp/Flow Rate/Pressure per blanket module	2) PbLi properities, static testing, MHD	3) 400C, 5 kg/s, 0.2 MPa	6) 700C, 30 kg/s, 1 MPa	5) 700C, 30 kg/s, 1 MPa	8) 700C, 50 kg/s, 2 MPa	9) 700C, 50 kg/s, 2 MPa
Surface Heating, module area	2) Contact heaters on small samples	3) 1 MW/m2, 1 m2	5) 0.5 MW/m2, 2 m2	6) 0.3 MW/m2, 1.5 m2	8) 0.5 MW/m2, 4 m2	9) 0.5 MW/m2, 4 m2
Tritium extraction rate	2) Property measurements	3) Concept tests	6) ~0.002 g/hr	4) ~0.002 g/hr	8) 2 g/hr	8) 13 g/hr
Fraction tritium recovered	2) Property measurements	3) 30%	6) 99.9%	5) 90%	8) 99.99%	9) 99.99%
Magnetic Field	NA	3) 2 T	5) 2 T	6) 4 T + poloidal field	8) 5 T + poloidal field	9) 8 T + poloidal field
Volumentric heated (as NWL), volume	NA	2) simulated w/ heaters, 0.01 m3	4) sim. w/ heaters, 2 m3	6) 0.7 MW/m2 wall load, 0.5 m3	8) 1-2 MW/m2, 2 m3	2-3 MW/m2, 2 m3
Max Operating time to replacement	NA	3) months	4) months to years	5) 2 years	7) 2-10 years	9) 5 years
Tritium production per module, TBR	1) Cross-sections	NA	6) Coupling to fission or accelerator source	4) 0.002 g/hr, 0.5	7) 0.04 g/hr, >1.0	9) 0.07 g/hr, > 1.05
Tritium containment	NA	NA	3) Concept tests	5) Collect data	7) Fully Integrated	9) Fully integrated
No. Modules	NA	NA	4) 1	5) 6	8) 50	9) 200
Availability (on demand)	NA	NA	5) 70%	6) 85%	8) 90%	9) 95%
Duty factor (annual)	NA	NA	5) 50%	4) 5%	7) 30%	9) 50%
Pulse length w/ integrated conditions	NA	NA	6) weeks	5) 1 hr	7) weeks	9) months
Fusion Fluence per component	NA	NA	NA	6) 0.1 MW.yr/m2	7) 1-6 MW.yr/m2	9) 7.5 MW.yr/m2

Tritium Purification and Recycle

Key Experiments	Basic experiments	Development experiments	Partially-Integrated (e.g. TSTA/TFTR/JET)	ITER, FCDF	FNSF, FCDF	DEMO
Performance Requirement						
Supplied T Conc	2) Few%	4) 10's of %	5) 50%	7) 50%	9) 50%	9) 50%
DT Flowrate	2) Low sccm	3) high sccm	4) 6 SLPM	7) 120 SLPM	8) 80 SLPM	9) 120 SLPM
Tritium inventory	2) ~1 gm	3) ~10 gm	4) 100 gm	7) 4000 gm	8) 4000 gm	9) 6000 gm
Tritium containment	1) Hoods	3) Standalone confinement and detritiation	4) Partially integrated confinment and accident response	7) Fully integrated confinment and accident response	8) Fully integrated confinment and accident response	9) Fully integrated confinment and accident response
Recycle time requirement	NA	NA	3) 24 hr	6) 1 hr	8) 1 hr	9) 1 hr
Degree of integration	NA	NA	3) 70%	6) 80%	8) 100%	9) 100%
Overall Facility Size	NA	NA	3) 3000 m3	6) 40000 m3	8) 35000 m3	9) 30000 m3
DEMO relevance	NA	NA	3) 30%	6) 60%	7) 85%	9) 100%
Availability (on demand)	NA	NA	3) 70%	6) 85%	8) 90%	9) 95%
Duty factor (annual)	NA	NA	3) 5%	6) 5%	7) 30%	9) 50%

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Chapter 4: Analysis of Opportunities for High-impact Research

4.1. Overview of Compelling Research Opportunities

Considering the broad range of R&D activities outlined in Chapter 3 that are intended to address the grand challenges and key scientific proof of principle issues for fusion nuclear science, the panel identified a small number of particularly compelling research opportunities. The evaluation was based on opportunities for US leadership in combination with R&D issues that ranked high according to the prioritization metrics outlined by the Greenwald FESAC panel [4.1]: Importance for fusion energy (and gap beyond current state of knowledge), urgency, and generality (applicability to multiple concepts). In order to quantify the gap beyond current state of knowledge, values of the TRLs that have been and could be achieved in various facilities were estimated by the panel members for a broad range of scientific phenomena (see tables in Chapter 3). Additional factors involved consideration of how existing facilities could be best utilized (perhaps with moderate modification), what significant new facilities would be needed during the near-, medium- or long-term, and whether there was any opportunity for the same facility to resolve proof of principle issues for multiple technology areas. High impact research opportunities that could be performed using existing facilities (or after moderate modifications) were considered to have a higher priority compared to research opportunities that could only be performed in newly constructed facilities.

Recommendation: As fusion nuclear science matures from concept exploration studies (TRL 1-3) to more complex proof of principle studies (TRL 4-6), it is appropriate to focus R&D on front-runner concepts.

Research within the multiple scientific disciplines that comprise fusion nuclear sciences is approaching a major transition. For most of the history of this research field, the vast majority of research activities were devoted to exploring attractive and innovative scientific hypotheses by performing exploratory single effects studies to determine their fundamental feasibility (equivalent to TRL 1-3 R&D activities). As the international fusion energy research community moves forward in the ITER era, in most technological areas there is sufficient information available to identify the most promising material, concept or technology. Therefore, the most expedient path as fusion nuclear science moves into more complex (and typically more costly and time-consuming) multiple-effects R&D studies is considered to involve focusing the majority of research activities on the front-runner approach. In some cases, the difference between the front-runner and back up option may be very close; in such cases a substantial minority share of the available funding might be applied to the backup concept. In other cases where the front-runner option is clearly identified, a large fraction of the available funding would be applied to the front-runner. It is anticipated a moderate (minority fraction) of funding might be applied to fundamental cross-cutting activities within the scientific R&D activity.

Looking to the future, the majority of the panel members considered the main priority for next-step major fusion nuclear energy device after ITER (e.g., FNSF) should be to extensively study the scientific basis for operational limits of the front-runner plasma-facing and blanket concepts (due to complexity, cost and time for testing multiple blanket concepts, particularly regarding tritium handling/extraction). However, additional community discussion regarding the scientific vision for a next-step device such as FNSF should be solicited.

Recommendation: Numerous fusion nuclear science feasibility issues can be effectively investigated during the next 5 to 10 years by efficient use of medium-scale facilities.

During the panel deliberations, it was evident that numerous scientifically important activities could be initiated within the near-term (0-5 years) using existing or modifications to existing facilities. Specific examples of recommended R&D opportunities are described in Chapter 3.2-3.4, with the most compelling opportunities summarized in Chapter 4.2-4.4.

Recommendation: The key mission of the next step US device should be to explore the integrated response of tritium fuel, materials and components in the extreme fusion environment in order to provide the knowledge bases to contain, conquer, harness and sustain a thermonuclear burning DT plasma at high temperatures.

This longer-term scientific mission will be an important driving force for prioritization of the most compelling scientific opportunities in the near- and medium-term. It may be prudent to minimize the electricity production mission for FNSF, due to large engineering science challenges associated with the exploration of synergistic fusion nuclear science phenomena in the extreme environment of a multi-hundred megawatt scale fusion energy device.

Since the FNSF and/or DEMO would be an experimental device, the ability to replace major components with notionally similar (modified) components is considered to be an essential feature. It is recommended that the next-step device after ITER should focus on exploring the myriad scientific and engineering issues associated with operation of the front-runner first wall and blanket concepts at DEMO-relevant conditions (high radiation, particle and heat fluxes, high coolant operating temperatures, significant tritium transport and continuous extraction, etc.). An alternative approach discussed by the panel involves a more flexible operational vision that would enable testing of significantly different first wall or blanket concepts (e.g., liquid vs. solid wall divertor concepts, or solid breeder versus DCLL blanket concepts). The majority of the panel felt there was merit in continuing to maintain focus on the front-runner concept for the next-step fusion energy device after ITER, unless the concept proved to be unviable. Due to the importance of this topic on the research vision for fusion energy, it is recommended that additional community input be sought in order to have a broad basis for making a decision regarding the testing mission for an FNSF.

4.2 Taming the Plasma Materials Interface

4.2.1 Significant confinement plasma science initiatives are required to provide any confidence in the extrapolated steady and transient power loadings of material surfaces for a FNSF/DEMO. The mechanisms governing the steady-state perpendicular power width on open-field lines must be determined. Integrated plasma scenarios, and operating

techniques must be developed that eliminate or mitigate transient heat loading from tokamak disruptions and intermittent edge instabilities such that surface damage to solid PFCs does not compromise their viability.

Power handling on the first wall, divertor, and special plasma facing components (PFCs) is challenging in steady state, and is severely aggravated by non-steady loading. Power handling at the plasma material interface must endure steady state, transient, and large off-normal heat fluxes, but at present, these fluxes cannot be predicted with sufficient accuracy. Efforts to eliminate or mitigate transient and off-normal loads are critical. We also know that "silver bullets" do not exist and that the ultimate solution will be a compromise between loading conditions, plasma operating modes, material properties optimization, design solutions, and component lifetimes.

The steady heat loads to PFCs are characterized by a very narrow scrape-off region just outside the plasma's last closed flux surface that concentrates the power into a small region on the target plates of the divertor. The standard approach to handling this power is a "detached" regime where the plasma and neutral particle densities are high and entrained in the divertor, and the majority of the power is radiated before striking the target plate. For a fusion power plant this approach would that that the fraction of radiated power in the divertor be very high. However our present understanding of the scaling of peak heat load with major design parameters such as linear size give large uncertainty in projecting the peak power loading for an FNSF/DEMO. This situation is unacceptable because the requirement for quiescent power exhaust must always be satisfied even if all transient heating instabilities are removed.

The extensive efforts by ITER over the last 15 years to characterize the heat loading during both disruptions and edge localized modes (ELMs) are impressive and but still inadequate. Presently, only empirical correlations exist for these processes and the correlations have large variations in associated parameters, which involve large energy deposition (3-100% of the plasma's stored energy) in very short time frames (400 microseconds to 1-2 seconds). In addition, the projected impact on materials, from the standard discharge, involves melting and ablation of tungsten or sublimation of graphite. Apart from the loss of material with each energy impact, which itself is an issue, thermal cycling can promote a number of crack formation mechanisms. Projections for ITER show that these PFCs could have very short operational lifetimes, requiring frequent and lengthy maintenance periods, if plasma regimes or approaches are not found to eliminate or reduce the transient loads.

In particular, the occurrence of disruptions in tokamaks presents a potential flaw for its development as a fusion energy source, due to the impact of these events on the blanket, divertor, and other surrounding structures. This type of event causes large thermal loads and large electromagnetic loads. Although present tokamaks, and ITER, are designed to accommodate disruptions through structural engineering without significant damage, a FNSF and DEMO cannot likely accommodate this due its impact on the power extraction and tritium breeding requirements. Aggressive elimination or amelioration of disruptions in the present US tokamaks is a high priority. The similarly aggressive development of operating modes with no ELMs, small ELMs, or amelioration of ELMs is also a high priority. Finally a need that permeates these studies is vastly improved measurements of the region outside the plasma's last closed flux surface to provide the needed data to predict accurately the effects and physical dependences of heat loading and a number of other PMI processes.

4.2.2 The leading FNSF/DEMO candidate solid material to meet the variety of PFC material requirements is tungsten due to its projected erosion resistance, high melting temperature and high thermal conductivity. Initiatives with the following objectives are required: 1) Identify and characterize suitable tungsten-based materials in appropriate plasma, thermal and radiation damage environments; and 2) Develop engineering solutions for tungsten PFCs with high-pressure helium gas coolant. The majority of PFC material research should be oriented towards tungsten, however due to open questions on tungsten melting and microstructural evolution; a parallel effort should be maintained in carbon-based solid materials with similar objectives.

The PFCs of a fusion device will experience the combined effects of plasma and nuclear loading. The material choices are severely limited, their performance in the fusion environment is highly uncertain, and the uncertainty in establishing PFC solutions is high. Important considerations are the impact on the core plasma via impurities, their response to plasma particle bombardment, their nuclear damage response, their thermal performance under high heat flux and operating temperatures above 500^{o}C, and their implications for safety and nuclear waste. Establishing material and design candidates will require significant efforts in experimentation and multi-scale simulation, and the coupling of plasma science, materials science, and advanced engineering and manufacturing technology.

Tungsten, or tungsten alloys, are the front-runner plasma facing material for the divertor and first wall (although the base material will likely be a ferritic steel), and the specialty components like launchers and diagnostics. This may be a solid component, multi-piece, or coating depending on the application. Although tungsten has a number of favorable properties as a PFC, its database is lacking, particularly in the regimes of high temperature and nuclear irradiation. Its brittle nature, and the aggravation of these properties by irradiation, creates an uncertainty in design to accommodate all loading and operational environments. An initiative is required to establish in detail the thermomechanical properties of tungsten, including unirradiated and irradiated at the relevant/expected operating temperatures above 500 C for FNSF/DEMO. In addition to the basic material properties, the manufacturing and joining of tungsten must be established. The operating window for tungsten within a range of plasma loading conditions which includes transients (heating and particles) is needed. The needed initiatives will include understanding of how to mitigate against fracture through development of tungsten-based materials as well as an understanding of failure modes of PFCs as components in an integrated system.

Due to uncertainties in the performance of tungsten, carbon-based materials should be considered a backup option as a solid PFC. Carbon has excellent thermo-mechanical properties (heat conductivity, etc.), and does not melt, which provides a large operational advantage to W. However carbon, which has been used extensively in confinement devices, has limitations; its erosion rate is substantially higher than W due to its low mass and therefore the total rate of atomic removal and migration will be higher, the deposits that carbon forms can hold substantial amounts of tritium fuel, and the carbon-based materials have a relatively poor neutron response, in particular its thermal conductivity decreases and volumetric changes tend to limit its lifetime. However, much like W, carbon has not been tested at relevant FNSF/DEMO operating temperature (T_{op})above 500 C. Based on laboratory data, the high T_{op} is expected to mitigate the retention issue and alter/reduce the erosion rates (due to dependence of chemical erosion on T_{op}). Temperature will likely affect neutron response as well through annealing processes, although data are scarce to non-existent in a fusion-relevant neutron field. Thus while the projected ideal lifetime of carbon-based PFCs may be worse than tungsten, it seems prudent to evaluate carbon at some level in a relevant environment.

The possibility of liquid metal surfaces for PFCs should be maintained at a secondary level, due to the uncertainty in the performance of solid materials. Feasibility assessments are required to determine if the basic concept works at all, even under steady heat loads. Successful outcomes would suggest further research in more integrated environments, particularly with plasma, magnetic fields, and transient loads. Design concepts would ultimately be required to determine the viability of integrated PFCs in the divertor, and possibly the first wall.

Modifications to the magnetic field geometry in the divertor region has been proposed as a method to reduce the heat flux on the PFC targets, such as the snowflake and Super-X configurations. Although the flux expansion aspects of these concepts is well understood, and in some cases demonstrated on present tokamaks, the more integrated function of the divertor to pump (control) particles is not. The resulting impact on the scrape-off layer plasma in the divertor, or nearby regions, requires more study. This is also considered a secondary level activity.

The integrated PFC design requires material properties, and their variations in the fusion neutron and plasma environments, thermal performance criteria for steady and any transient loading, high temperature operation, as well as serious boundary conditions associated with the plasma exposure. These include material loss and/or re-deposition and tritium retention in surfaces and bulk material (trapping at damage site). The complexity of this design space is highly unique and challenging.

4.2.3 A combined initiative for both extensively diagnosing the region outside the plasma's last closed flux surface and learning about material responses to plasma exposure in real-time and during operation (in-situ) is necessary to develop and validate the physics understanding over a wide range of processes ranging from power scrape-off width to material migration.

This includes determination of the radial power scrape-off width, operating conditions and stability for divertor detachment, flows of both main plasma ions and eroded materials throughout the SOL to other regions and into the plasma, behavior of fusion fuel and helium ash, and integration of solutions for PFCs that are consistent with the high performance core plasma. Pursuing high temperature PFCs representative of FNSF/DEMO should be pursued due to the strong temperature dependence of the associated physical chemistry.

The boundary plasma continually rearranges plasma-facing materials through sputtering, plasma transport and redeposition. The modifications to materials, such as the erosion depth, become macroscopic for pulses of months-to-year duration. It will be necessary to reduce the impact of this material erosion and migration to an acceptable level for plasma performance and continuation. Particular issues of concern are the complete removal of PFC materials through net erosion, the reduced thermo-mechanical response of thick plasma deposits which may lead to perturbing macroscopic removal of materials that perturbs the plasma, the microstructural evolution of surfaces, tritium fuel retention and the production of dust. Achieving this understanding requires that we develop approaches to perform measurements in the plasma edge and in PFCs *in-situ* and in *real-time* in a relevant plasma environment.

The retention of tritium in co-deposited layers on PFCs or in dust creates a safety issue both for release and for deflagration. The dust will also contain other radionuclides which can be volatilized in the event of an accident. Both processes, co-deposition and dust generation, will require a period without plasma operation while remedial action is taken to remove the material. Control and strategies for mitigation will require knowledge of SOL plasma physics and material response to plasma exposure.

The physics associated with the SOL plasma and material interactions with this plasma is complex and virtually always occurs as collective processes containing several underlying physics mechanisms. In order to understand this it will be necessary to advance from our present capability in the theory and simulation, and embark on an aggressive program of experimental validation. The extensive diagnostic development suggested by this research activity can only meet its potential if there is a coordinated computational component.

4.3 Conquering Nuclear Degradation of Materials and Structures

The panel identified five compelling opportunities for conducting high-impact research to conquer nuclear degradation of materials and structures. Emphasis is given to impactful research that can be performed in the next five years and will significantly advance the Technical Readiness Levels of the issues described in Table 3.3.1. Longer-term research that must be performed to prepare for FNSF and DEMO design activities is described, but the timeframe when this research will be needed depends on many variables and was difficult to precisely determine. The role of key facilities to retire the R&D risks highlighted in Table 3.3.1 is presented. Rough assessments of research priority are given along with judgments on U.S. leadership opportunities.

4.3.1 Re-engagement in the IFMIF Broader Approach Engineering Validation and Engineering Design Activity (EVEDA) should be initiated, in parallel with limited-scope neutron irradiation studies in upgraded existing spallation sources such as SINQ or SNS.

The key recommendation related to nuclear degradation is that the U.S. should to re-engage with the International materials irradiation community (i.e., participate in several working meetings per year) to determine the status and prospects for the construction and operation of the International Materials Irradiation Facility (IFMIF). This re-engagement is desired to facilitate the path forward of this critical neutron irradiation facility. Simultaneously, the U.S. should continue materials irradiation studies using available nuclear reactors and ion beam facilities, while moving forward with intense spallation source facilities to serve as a bridge until the IFMIF facility is operational for obtaining materials nuclear degradation data with high levels of transmutation.

This is a high priority activity, which should begin as soon as possible because the scientific knowledge and database that will be needed to support the design, construction and licensing of future plasma devices FNSF and DEMO cannot be developed without such a facility. Re-engagement with IFMIF-EVEDA at an observer level would constitute a relatively low-cost activity that will enable the U.S. to keep in touch with and significantly influence the future direction of this critical facility. At present there is no decision to proceed to IFMIF construction, but re-engagement by the U.S. could significantly boost the credibility of IFMIF and provide momentum toward its eventual construction.

Establishment of a limited-scope irradiation capability in an upgraded existing spallation source such as SINQ or SNS will enable important basic science studies in a neutron environment that more closely approximates fusion nuclear conditions. Again this is a relatively low cost effort in comparison to a major spallation source upgrade or commitment to construct IFMIF. This activity will partially validate microstructural development experiments performed in mixed spectrum fission reactors, and the physical models that have been developed based on those experiments, while yielding a low dose assessment of the effects of transmutation produced gases on mechanical and physical properties of materials. Implementation of this recommendation will likely place the U.S. in a leadership position on quantitatively assessing the synergistic effects of displacement damage and gaseous transmutants.

4.3.2 A detailed engineering design activity should be initiated that is closely integrated with materials research activities and is supplemented with limited neutron irradiation data (~10 to 20 dpa) from spallation neutron sources to permit selection of a prime candidate reduced activation steel for FNSF.

A fusion reactor must meet all design, qualification, safety and licensing requirements typically applied to fission reactors. This requires all component design and fabrication to meet strict nuclear-based design codes and a rigorous quality control and assurance process. However, an adequate knowledgebase does not exist and extensive operating experience for fusion components, as compared to fission reactors is completely lacking. Furthermore, ASME code rules do not cover design, construction, operation and inspection of magnetic confinement fusion energy devices.

Development of new design codes and rules for plasma-facing components in fusion energy devices is complicated further due to the need to develop high-temperature structural design criteria for safe operation in high-neutron irradiation environments. Further, the knowledgebase on the expected thermal-mechanical operating environments is hindered by incomplete analysis of the structural design of fusion reactor components. Therefore a science-based understanding of high temperature synergistic effects of multiple deformation processes (e.g., thermal creep, creep-fatigue, cyclic mechanical fatigue, ratcheting) need to be developed and validated in confirmatory tests (e.g., in neutron irradiation environments). The methodology used by governing engineering bodies, such as the ASME for qualifying operation of structures at high temperatures is based on empirical testing. Due to lack of fusion test reactors the development of science-based methods for qualifying structural material for high temperature operation will be necessary to avoid very costly testing-based empirical approaches.

This is a high priority activity because the process of developing high-performance materials that are able to meet design goals while maintaining adequate margins against failure is significantly enhanced. Enhancement of current design activities will permit more rigorous evaluation of various design options, and can be accomplished with at relatively modest cost. No new facility investments will be needed in the next five years. For the longer term as noted in Table 3.3.1 substantial investment in non-nuclear structural integrity testing and benchmarking facilities will be needed. Implementation of this recommendation will keep the U.S. on par with the most advanced materials research efforts underway in the European Union and Japan.

4.3.3 A robust experimental and theoretical effort should be initiated to resolve scientific questions associated with the permeation and trapping of hydrogen isotopes in neutron-irradiated materials with microstructures designed to mitigate transmutation produced helium.

As noted above tritium permeation and retention in fusion materials could be a significant safety issue. The scientific understanding of tritium effects is not sufficient to accurately predict the performance of large surface area, geometrically complex, thermo-mechanically loaded structures of an FNSF or DEMO reactor. Obtaining better quantitative understanding is crucial because; 1) effective radiation resistant, high-temperature tritium barrier materials will be needed to prevent tritium from accumulating in unwanted places; 2) conversely high permeability materials are needed in selected areas of a fusion reactor to permit efficient extraction of tritium from liquid coolants and breeders; and 3) microstructures designed to manage helium may have the unintended consequence of storing unsafe levels of tritium.

In the next five to ten years considerable progress can be made toward development of science-based models of tritium permeation and retention in a wide variety of fusion structural and functional materials. The facilities and technical expertise already exist to perform fundamental microstructural and permeation investigations that will lead to better understanding of 1) how tritium diffuses and is trapped in neutron irradiated materials; 2) how barrier materials perform effectively in the laboratory but much less effectively in an irradiation environment; and 3) how to design microstructures that selectively permit tritium permeation. This can be achieved by using existing mixed spectrum fission reactors coupled with *in situ* helium injection techniques to produce relevant microstructures that can be permeation tested in either simple permeation cells, ion sources, or in linear plasma sources, such as PISCES or TPE. This is a high priority activity because significant information can be gained at relatively modest cost and it addresses a ubiquitous scientific issue.

For the long-term validation of models and materials performance will need to be conducted in a fusion relevant neutron source to permit exploration of behavior at FNSF and DEMO fluence levels and helium concentrations. Ultimately model validation and verification will be needed in a full-scale facility such as FNSF to explore the potential for synergistic effects that are not revealed in simpler single-variable experiments or limited multiple-variable studies. Integral tests of this type are essential to validate models of materials performance and to provide insight and understanding of possible synergistic phenomena that compromise system reliability and performance.

4.3.4 Experimental and theoretical investigations to develop science-based high-temperature design criteria, and understanding of the fundamental mechanisms controlling chemical compatibility in the fusion environment should be significantly enhanced.

As described above, substantial research is needed to develop appropriate design Codes needed to design fusion components. Progress towards this outcome will require experimental validation of the proposed rules. Some of the key experimental facilities and approaches needed here include the following:

- Single coupon testing of materials is required to develop the needed constitutive laws. Essential results include unirradiated properties that are not yet available (such as crack growth rates for tungsten), irradiated properties using existing facilities (primarily ion beams and fission reactors), and irradiated properties from a source with a fusion neutron spectrum.

- Sub-component facilities to carry out high heat flux testing and validate high temperature design rules. In addition to providing more flexibility for validating the design rules (beyond those possible with coupon tests), these tests will identify synergistic effects that may not have been anticipated by prior modeling efforts.

- Full-size component testing to explore the possibility of synergistic effects not identified in previous facilities and to fully qualify components to be placed in a DEMO.

For the near term, it would appear to be particularly useful to develop an initial science-based model of high temperature creep-fatigue (potentially leveraging small activities that might be pursued by fission reactor and other programmatic efforts). As a first step, this model could be compared and validated against the rather substantial experimental test results on creep and fatigue of unirradiated ferritic steels and austenitic stainless steels. As the model becomes more robust, specific radiation effects experiments could be performed to investigate the potential impact of deformation flow localization and radiation induced solute segregation on the controlling deformation mechanisms at high temperature.

4.3.5 High-Temperature Superconductor Development

The ReNeW report outlined a research program that covers the major issues for development of magnet materials that would be required for development of an HTS-based confinement system and offer the advantages of demountable coils [4.1]. This program remains a valid and comprehensive program for the development of HTS for Fusion that addresses the issues raised above:

- HTS wire and tape development program.

- High current conductors and cables development program.

- Development of advanced magnet structural materials and structural configurations.

- Development of cryogenic cooling methods for HTS magnets.
- Development of magnet protection devices and methods specific to HTS magnets.
- Development of advanced radiation-tolerant insulating materials.
- Integration of conductor with combined structure, insulation, and cooling.
- Development of joints for demountable coils.
- Coil fabrication technology incorporating the unique features of all of the preceding elements.

Timeframe for Development of HTS Fusion Conductor in 15 years

In Europe a timeline for the development that would be required for an HTS-based approach to DEMO was proposed by the F4E/CCE-FU Ad Hoc Group on DEMO activities [4.2]; even over the 15 year time span, the required development of multiple new technologies requires an aggressive schedule (see Table 4.3.5.1).

U.S. Leadership in High Field and HTS Development

The U.S. has a strong leadership position in the development and use of high field superconducting magnets, including the highest magnetic field for a continuous field magnet - the 45 T Hybrid superconducting/resistive facility at NHMFL (NSF funding) the highest field (16 T, Nb_3Sn) dipole magnet at LBNL (DOE-HEP funding). The U.S. also holds and the records for high fields in REBCO tape magnets described previously. The U.S. has also been the leader in the development of the REBCO coated technology that made those HTS magnets possible, and that leadership can be attributed primarily to major support by OE (Office of Electricity Delivery and Energy Reliability) of the Department of Energy (DOE), which ended in FY10; thus both the rate of conductor advancement and the U.S. technological leadership in this and related areas are likely to be greatly diminished in the years to come. High energy physics has recently funded a small program aimed at demonstrating key aspects of HTS technology required to make a Muon Collider feasible (see the BNL coil). The National High Magnetic Field Laboratory under National Science Foundation (NSF) support has placed all-superconducting user magnets of >30 T in its next 5 year plan. Thus there are appropriate vehicles for collaboration both with DOE and NSF for future fusion efforts.

Table 4.3.5.1. Superconducting Magnet Development Timeline from the Report of the European Fusion for Energy and the CCE-FU Ad hoc Group on DEMO Activities [4.2].

What needs to be done?	Priority	Start	Relationship to existing R&D programs	Timescale (Years)
Clarify DEMO objectives and evaluate the impact of HTS magnets with respect to the scope for system simplification.	1	Immediate		2
Evaluate magnetic field strength effects on feasibility of use of HTS magnets and clarify development needs	1	Following clarification exercise.	National activities for investigating HTS material properties with respect to Fusion requirements have started recently in Germany, Japan and U.S.	1-2
Develop suitable cabling concept for HTS Fusion magnets taking into account loss, cost and manufacturing	2	Following from clarification and evaluation		5-10
Demonstrate sub-size model coils as proof of principle	2	Once cabling concept development at a mature stage		5-10
Demonstrate full prototype or model coil	3	Once sub-size model coil passes tests		10-15

4.4. Harnessing Fusion Power

Fusion development, including all aspects of plasma physics research, requires authoritative information on many aspects of fusion nuclear science to evaluate technological readiness and identify paths toward a successful DEMO. In the near term, emphasis is placed on R&D that 1) enables basic understanding, 2) rapid advancement towards key decision points and evaluations of currently envisioned solutions, and 3) opportunities for innovation and success. This R&D phase requires building both core test facilities and a trained work force capable of advancing the program to the multiple-effect and integrated testing stages.

The research opportunities described here are based on the fact that the US has developed a potentially attractive family of first wall / blanket concepts based on the use of a lead-lithium eutectic alloy as a breeder/coolant, separate cooling of reduced activation ferritic steel first wall and structure with helium, and the use of thermal / electrical insulating inserts based on silicon carbide. Currently, the lead-lithium based Dual-Coolant Lead Lithium (DCLL) blanket is the lead US option based on design and R&D that has occurred over the past decade. The DCLL concept has been developed by the power plant studies program and studied as a candidate US ITER-TBM [4.6-4.8]. The DCLL uses flowing PbLi as both breeder and coolant for the breeding zones, while utilizing high pressure helium to cool all structures including those surrounding the breeding zone. Flow channel inserts made of a SiC-composite in all liquid metal ducts serve as electrical and thermal insulator, enabling a liquid metal exit temperature about 200K higher than the maximum temperature of the steel structure. By this method the thermal efficiency in the power conversion system can approach 45%, compared to values of ~40% for entirely He-cooled blankets. R&D on aspects of the DCLL, on lead-lithium as a generic liquid metal breeder, multiplier and coolant, and on helium cooling are applicable to a number of concepts currently proposed in the international community [4.3], and can serve as fallback options should some aspects of the high temperature DCLL not prove practical or feasible. Strong research programs on PbLi as a breeder exist in the EU and China. Research cooperation on fundamental issues can help bolster US R&D work with US efforts focusing on the unique issues of the high temperature DCLL. The US fusion program specific research strengths include SiC fabrication and thermomechanical testing, modeling and experiments for liquid metal magnetohydrodynamics and related transport phenomena; surface heat flux handling; tritium processing, permeation and control; and neutron transport safety / accident event analysis.

Another option, a stationary, helium-cooled, pebble bed, ceramic breeder, beryllium multiplier based FW/Blanket concept, is the current focus of the majority of international breeder blanket systems R&D programs. Such blankets use lithium ceramics typically in pebble bed form with a circulating purge gas to remove the tritium. Ceramic breeders have markedly different feasibility issues, and so represent a strategic alternative to the liquid breeder systems. Ceramic breeder blankets have been investigated in many reactor systems studies and for ITER-TBM. The opportunity here is to capitalize on the large international R&D programs by maintaining a low level of R&D activities in the US on selected critical areas associated with ceramic breeder and beryllium multiplier thermomechanics and tritium release.

The US fusion program retains some niche capabilities in specific research areas including modeling and experiments for liquid metal magnetohydrodynamics and related transport phenomena; surface heat flux handling; tritium processing, permeation and control; power plant modeling; and neutron transport, safety / accident event analysis, as well as power plant design and modeling. The opportunity in the near-term is to utilize these capabilities to make progress on feasibility and design issues that most affect the primary candidate blanket systems, while recognizing the need for a coherent longer term development strategy including multi-effect and integrated fusion environment testing and a comprehensive predictive capability. The main recommendations are summarized below, and they are described in more detail in subsequent sections.

RH1 A fully integrated strategy to advance the scientific and engineering basis for power extraction and tritium breeding systems. should be established. Such a strategy should address this challenge through a mix of single- and multi-effect experiments in non-nuclear environments, fast neutron sources and fusion devices together with a comprehensive simulation capability.

RH2 Several key feasibility issues for the lead-lithium based blanket concepts should be examined as soon as possible in order to provide confidence in successful development of these concepts. These feasibility issues include tritium extraction from hot PbLi and He; liquid metal MHD effects on flow control and heat, tritium and mass transfer; chemistry control and compatibility of PbLi with, and thermomechanical loading of, ferritic steel structures and ceramic flow channel inserts.

RH3 The development of coupled models and predictive capabilities that can simulate time-varying temperature, mass transport, and mechanical response of blanket components and systems should be emphasized. These predictive capabilities should be validated against the experimental database; used to explore the coupling between disparate phenomena and loading conditions; and used to extrapolate beyond testing conditions to help guide and interpret further experimentation.

RH4 The performance and reliability of blanket and tritium extraction systems must be understood, demonstrated and made predictable with prototypic geometry, and in multi-material unit cells and mockups under combined loads where phenomena studied in separate effects tests can produce interactions that may lead to unanticipated synergistic effects. Planning for multiple-effect test facilities combining simulated thermal, mechanical, chemical, and electromagnetic conditions should begin in earnest as multi-effect experiments are essential prerequisites to any integrated testing program in fusion devices.

4.4.1 PbLi Based Blanket Flow, Heat Transfer, and Transport Processes

The DCLL is identified as a primary US liquid metal based FW/blanket option, with PbLi itself being a generic liquid breeder and coolant medium applicable to a family of liquid metal blanket concepts as well [4.4]. PbLi in the blanket will absorb the majority of the nuclear heating in the blanket system. This energy will be either transported by the PbLi flow to the heat exchanger and will be transported via conduction and convection to the helium used to cool the structure. In either case the fluid dynamics of the PbLi plays a crucial role in the transport and recovery of this energy and therefore on the temperature and temperature gradients in the blanket structures. In addition the flow of PbLi dominates the transport of tritium and activated corrosion products throughout the system. PbLi in liquid metal blankets, whether flowing slowly for tritium removal, or more rapidly for power extraction will experience magnetohydrodynamic (MHD) forces at least 3-5 orders of magnitude greater than viscous and inertial forces of ordinary hydrodynamic flow, dominating their flow physics.

There is currently one PbLi loop in the US with capability to perform MHD and transport related experiments, but its operating temperature is limited to 400C due to material restrictions from the use of austenitic steel. There are several PbLi flow facilities in the EU but only one small one in Riga geared towards studying MHD effects (on corrosion). There has been recent construction of PbLi facilities in KO and CN but parameters and research programs are not well known. To date, there has been no SiC flow channel inserts experiment done in a flowing LM test facility and none of the experimental facilities noted have a magnetic field in excess of 2 T.

Thermofluid MHD modeling for liquid metal blankets has been slowly improving in various areas including the attainment relatively large Hartmann number (10^3-10^4) in fully 3D laminar MHD calculations and some flexibility to study complex geometry effects and multiple materials (FCIs) [4.5]. At the same time various research codes aimed at understanding flow fluctuations and quasi two dimensional MHD turbulence resulting from unstable buoyancy driven forces and shear flows has also been made, pushing towards relevancy on other important parameters such as Grashof and Magnetic Interaction Parameter. Transport models for calculating the corrosion and deposition using relevant MHD velocity profiles have also been developed of late to help understand corrosion behavior.

The following near-term research tasks that can contribute to the evolution and evaluation of the DCLL concept are recommended:

- Pressure drop, flow instabilities, and distribution in PbLi blankets with prototypic heating, temperature, geometries, and materials (especially with SiC based flow channel inserts)
- Extraction of tritium from PbLi at high temperature, tritium recovery efficiency and longevity with typical impurities
- PbLi corrosion, transport, deposition, and impurity control with prototypic temperatures and materials (RAFS/SiC/PbLi and tritium extraction and heat exchanger tube materials)

Each of these subtasks will require upgraded or new laboratory scale facilities and modeling efforts, and a mechanism to feed results into an evolving DCLL design concept with significantly reduced performance uncertainty and robustness to fluctuations in heating, irradiation, and magnetic field conditions.

4.4.2 Robust high heat flux Blanket/FW structures

In designing fusion power plants, high-pressure helium coolant is typically employed to remove heat deposited in the first wall (FW), divertor and blankets structures. In the DCLL blanket concept, this same structure must contain the flowing molten PbLi which captures the majority of fusion neutrons and their energy. The FW/blanket are one integrated structure, and so it is not useful to consider them separately. The typical FW heat flux assumed for various power reactor design studies and a fusion DEMO is ~0.5 MW/m^2 over an area hundreds of square meters in size.

For steady state operation, the FW/blanket has to have a robust design so that it will withstand the high coolant pressure (~8 MPa), have a surface material compatible with plasma interactions, surface heat removal, and fusion neutrons. The coolant and its peak temperature must be suitable for use in a system with high power conversion efficiency. It is necessary to minimize the volume fraction of RAFM steel and any other heat sink or armor materials such that the necessary tritium breeding ratio can be obtained. The component must have adequate lifetime and be maintainable and replaceable. In addition, the FW components will need to function properly during all the non-steady-state operational phases of a tokamak, including startup, shutdown and all expected transient events.

The complex heat-transfer problem in a gas-cooled first wall is exacerbated by coolant channel complexity and the need for a relatively thin first wall which can withstand transient events. In addition, many physical phenomena limits the life of the first wall including large temperature gradients and associated stress and strain, creep (thermal and irradiation), fatigue and their non-linear synergistic effects. It is not clear that standard models for turbulent heat transfer (k-epsilon) and/or wall functions can be applied for fusion systems with normal flow impinging jet cooling and other complicated cooling arrangements.

As such, research in the following areas is proposed:

- Experimental study of helium heat transfer enhancement and flow stability, of strongly heated, prototypic RAFM steel FW/blanket mockups.
- Analysis of first-wall life-limiting phenomena to guide material research.

We also expect that plasma physics research will aim at a better understanding and predictive capability of transient heat and particle loads on the first wall during transients.

4.4.3. Plasma Exhaust and Blanket Effluent Tritium Processing

A commercial fusion system will require proper handling of DT fuel and reaction products. Only a fraction of the tritium burns on each pass through the plasma, so most of the tritium must be processed and fed back into the reactor. As seen in the Harnessing Fusion Power Development Path tables (Chapter 3), substantial experience already exists from the Tritium Systems Test Assembly at Los Alamos National Laboratory [4.6], the tritium systems at the Tokamak Fusion Test Reactor at the Princeton Plasma Physics Laboratory [4.7], and the Active Gas Handling System at the Joint European Torus [4.8]. TSTA at Los Alamos National Laboratory was constructed and operated as a fusion fuel processing integrated prototype. Tritium systems at the TFTR and JET were integrated with DT fusion reactors. And a number other facilities in the US, Canada, Japan, the EU and elsewhere have contributed important information to this area. However, these systems were typically tested at 1/20th scale or less of ITER, so considerable additional work remains to meet the requirements of ITER, DEMO and other future facilities. ITER itself represents a large step forward from these facilities towards DEMO-relevant tritium processing systems.

But significant extension of the knowledge base will be necessary to realize fusion power, in particlar in processing rates and the time to produce on-spec product as well as significantly increased duty cycle requirements will be required. In addition, some systems will have to be adapted to cope with operations in a nuclear environment. It should also be noted that the ITER tritium systems will largely be a production system with little opportunity for experimentation outside what is needed for operations.

The need for development in this area stems from:

- Certain plamsa exhaust processing technologies, while shown to be feasible, have received very little development toward a practical system
- Tritium extraction from blanket materials is only beginning and technology that has been tested shows low extraction efficiency compared with DCLL goals
- Tritium containment at DEMO-relevant conditions especially includes a breeding and power extraction system has never been tested
- The integration of plasma exhaust and tritium extraction has not been demonstrated and has only received little design attention
- As tritium processing systems are being scaled up, the cost and size of certain systems is being unmanageable, necessitating the need to develop alternate technologies
- Tritium handling technologies receive exceptional regulatory requirements to demonstrate reliable and effective performance which results in the need for more-thorough-than-usual testing

Meeting these needs will require both development of new technologies and extensions and refinements of existing technologies. R&D tasks are recommended in the following areas.

- Detailed DEMO fuel cycle modeling study. A complete study of a DEMO tritium system has never been done and will be extremely useful to consolidate assumptions and R&D needs beyond ITER in terms of plasma exhaust processing, fuel processing, vacuum, and tritium accountability
- There are a number of important tritium separation operations that need attention such as removal of tritium from He, removal of impurities from hydrogen isotopes and separation of tritiated water from a variety of streams. A series of small-scale tests with H/D would provide valuable design data. This effort would have high synergy with the fuel cycle modeling effort.

4.4.4 Medium term Research Needs – Focus on Multiple Effect Experiments and Simulation

In a fusion power extraction and tritium fuel cycle R&D program where the above activities are being actively pursued, the medium term 5-15 year time frame plays an important role in the overall fusion development strategy. This period serves as a buffer time to complete and digest R&D begun in the near-term, especially as unanticipated findings may require repeating or redirecting some efforts. But it is also the point where various separate effect experimental and verified modeling capabilities can make a turn towards addressing multiple-effect phenomena in multi-material and geometrically complex unit cells and component mockups. The experimental activity should be designed to provide data to support these modeling, safety and reliability studies; and to eventually serve as prototypes that precede fully integrated testing in a representative fusion environment.

The following multi-effect campaigns are recommended:

- **Blanket/ FW Thermomechanical/Thermofluid MHD integration** – (1) acquire precise measurements of thermomechanical, thermofluid MHD performance of mockups for comparison to and validation of simulation capabilities, (2) gain failure modes, frequencies and effects data for representative blanket systems with prototypic materials, temperatures, and under simulated fusion loading conditions.

- **Integrated Bred Tritium Extraction and Processing –** (1) **test and develop technologies for** tritium extraction from PbLi with prototypic tritium concentrations, transport, impurities and containment; (2) determine impact of nuclear reaction tritium and helium production on PbLi on tritium extraction efficiency

To undertake such studies modeling efforts must continue and become even more integrated, and a new class of multi-effect test stands will likely be required. These facilities (BT3F and BTEF) are included in the Harnessing Fusion Power development path tables in Chapter 3, and a more detailed description of possible facility parameters and research program is provided in Ref. [4.9]. Both facilities are anticipated to be significant undertakings, bringing together for the former:

- High temperature PbLi flow and He flow loops
- Megawatts of simulated FW heat flux and internal heating (likely via embedded heaters, surface heaters, or induction heaters, this will require careful study and integration with the experimental mockups themselves)
- High magnetic field with variable field direction, field gradients
- Mechanical loads: weight, pressure, vibration, impulses (for example via pulsed current in magnetic field)

and for the later:

- High temperature PbLi flow loop
- Tritium extraction system
- Control of radiological hazards
- Coupling to a neutron source (fission reactor or other source) for tritium breeding

The mission and planning for these multi-effect research programs and test facilities should begin in earnest, so that they can be constructed in medium-term.

4.4.5 Validation in an integrated fusion environment

The need to perform integrated validation of the predicted performance of blanket/FW, divertor, shield components and materials in a true fusion environment is the last step prior to proceeding to a fusion DEMO. ITER is the first burning plasma fusion device but has limitations in pulse length and total fluence. While there is disagreement as to the benefits of ITER test blanket program, there is a consensus in the US that a facility such as the Fusion Nuclear Sciences Facility (FNSF) will be necessary to bring together larger scale integration, multi-module interactions, uniformity of nuclear field, and middle to end of life irradiation damage.

We recognize that the international ITER TBM program is proceeding, and that the US intentions regarding utilizing or collaborating on ITER-TBM are not clearly defined. Furthermore, the lead time to develop and build an FNSF facility is long. As such, we re-iterate the need to develop a fully integrated strategy to develop the scientific and engineering basis for power extraction and tritium breeding systems. A major component of such a strategy is the US position on ITER-TBM as well as detailed mission and requirements for the FNSF facility.

Chapter 4 References

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Appendix A: Charge Letter

Department of Energy
Office of Science
Washington, DC 20585

Office of the Director

July 22, 2011

Dr. Martin Greenwald
Plasma Science and Fusion Center
Massachusetts Institute of Technology
77 Massachusetts Avenue, NW16
Cambridge, MA 02139

Dear Dr. Greenwald:

I request that FESAC address the opportunities for the U.S. plasma and fusion research communities presented by new or soon-to-be commissioned fusion facilities outside the US. In addition, I would like FESAC to elucidate the research needed to fill the gaps in materials science and technology required to sustain fusion plasma operations and to harness fusion power.

There are two reasons to investigate the international research opportunities now. First, plasma dynamics and control may well be defined by the capabilities of facilities that use superconducting magnet technology – currently all overseas. These facilities are at the forefront of advanced tokamak and stellarator research, and they present significant new opportunities for U.S. engagement. In fact, in some cases the U.S. has been invited to participate in setting program direction. Second, budget realities make it unlikely that the U.S. will construct a major new domestic facility for some time, and certainly not during the period of ITER construction. Regarding materials science, technology, and harnessing fusion power, you are already aware of the gaps that have been identified in the world program that must be filled if ITER is to be the penultimate step to a DEMO, and the opportunities for U.S. leadership through well-posed initiatives in these areas of research.

With this in mind, I ask FESAC to consider the following.

1. What areas of research on new international facilities provide compelling scientific opportunities for U.S. researchers over the next 10 – 20 years? Look at opportunities in long-pulse, steady-state research in superconducting advanced tokamaks and stellarators; in steady-state plasma confinement and control science; and in plasma-wall interactions.

2. What research modes would best facilitate international research collaborations in plasma and fusion sciences? Consider modes already used by these communities as well as those used by other research communities that have significant international collaborations.

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3. What areas of research in materials science and technology provide compelling opportunities for U.S. researchers in the near term and in the ITER era? Please focus on research needed to fill gaps in order to create the basis for a DEMO, and specify technical requirements in greater detail than provided in the MFE ReNeW (Research Needs Workshop) report. Also, your assessment of the risks associated with research paths with different degrees of experimental study vs. computation as a proxy to experiment will be of value.

I look forward to receiving your assessments by January 31, 2012.

Sincerely,

W. F. Brinkman
Director, Office of Science

Appendix B: Subcommittee Members

Name	Institution
Steve Zinkle*(chair)	Oak Ridge National Laboratory
Farrokh Najmabadi*	University of California-San Diego
Rich Callis*	General Atomics
Kathy McCarthy*	Idaho National Laboratory
Dennis Whyte	Massachusetts Institute of Technology
Richard Nygren	Sandia National Laboratories-Albuquerque
George Tynan	University of California-San Diego
Brian Wirth	University of Tennessee-Knoxville
Rick Kurtz	Pacific Northwest National Laboratory
Chuck Kessel	Princeton Plasma Physics Laboratory
Jake Blanchard	University of Wisconsin
Neil Morley	University of California-Los Angeles
Scott Willms	Los Alamos National Laboratory /on ITER assignment
Peter Lee	Florida State University

*FESAC member

Appendix C: Summary of panel evaluation process and meeting schedule

Due to the relatively short period of time (6 months) from the date the Charge was delivered to FESAC (July 28, 2011) and the requested delivery date of January 31, 2012, most of the panel business was conducted via teleconferences and email exchanges. A total of 15 conference calls involving the full subcommittee were held from October 3, 2011- Feb. 10, 2012. In addition, two face-to face meetings involving the full subcommittee were held on Nov. 7-8, 2011 in Gaithersburg, MD and on Jan. 5-6, 2012 at the University of California-San Diego.

In response to a request for community input, 21 contributed white papers and 5 email comments were received from community researchers; these were discussed at the UCSD face-to-face meeting. The community input is summarized in Appendix D.

In addition, three invited teleconference presentations to the full panel were solicited on the topics of fusion safety (Brad Merrill, INL), Technology readiness levels as applied to fusion energy sciences research (Mark Tillack, UCSD), and Lessons learned from the NGNP fission reactor project as applicable for fusion energy research (Dave Petti, INL).

The following solicitation was sent to a wide distribution network in early November, 2011 and was also presented at a University Fusion Association (UFA) session at the 53rd American Physical Society-Division of Plasma Physics (APS-DPP) meeting on Nov. 14, 2011 in order to attract diverse community input. The solicitation was forwarded to members of the following organizations: American Nuclear Society-Fusion Energy Division, APS-DPP, UFA, Burning Plasma Organization, and the fusion energy sciences Virtual Laboratory for Technology.

FESAC materials sciences subcommittee seeks community input

In response to a charge from the Office of Science to assess "what areas of materials sciences and technology provide compelling opportunities for US researchers in the near term and in the ITER era", a FESAC subcommittee consisting of 14 scientists is evaluating research needs to bridge current knowledge gaps in order to establish the scientific basis for a Demonstration power plant. The subcommittee evaluation is scheduled to be completed by January 31, 2012.

Research community input is solicited on key scientific challenges that need to be resolved, particularly in the following topical areas: Plasma-materials interactions, nuclear degradation of materials and structures, and fusion power conversion and tritium fuel cycle technologies. The contributions should focus on the scientific issue(s) to be resolved, rather than technical specifications of facility(ies) that might be important for resolving current engineering science barriers.

Short white papers or suggested scientific questions or issues to be considered by the subcommittee can be submitted to the FESAC materials sciences web site (http://aries.ucsd.edu/fesac_mat/) by sending the contribution to Farrokh Najmabadi (fnajmabadi@ucsd.edu). Questions regarding the scope of issues to be evaluated can be submitted to the subcommittee chair, Steve Zinkle (zinklesj@ornl.gov).

Appendix D: Overview of community white paper solicitation process

Requests for community input was distributed to a broad range of fusion energy science mailing lists (American Nuclear Society-Fusion Energy Division, American Physical Society-Division of Plasma Physics (APS-DPP), University Fusion Association (UFA), Burning Plasma Organization, and the fusion energy sciences Virtual Laboratory for Technology), and was also presented at a UFA session at the 53rd APS-DPP meeting on Nov. 14, 2011. A total of 21 contributed white papers and 5 email comments were received from community researchers, and were discussed by the full subcommittee during teleconference calls and during a face-to-face subcommittee meeting in early January. In addition, three invited teleconference presentations to the full panel were solicited on the topics of fusion safety, Technology readiness levels as applied to fusion energy sciences research, and Lessons learned from the NGNP fission reactor project as applicable for fusion energy research prioritization.

All contributed contributions (white papers and emails) and teleconference presentations to the subcommittee are posted on the FESAC materials science subcommittee website: https://aries.ucsd.edu/FESAC_MAT/

Community Input

Presentations

1. D. Petti, "Prioritizing R&D Activities in Construction Project: NGNP Lessons Learned" (1/23/2012)
2. M. Tillack, "Technology readiness evaluations for fusion materials science & technology" (12/20/2011).
3. B. Merrill, "Fusion Safety Scientific Challenges Leading to a FNSF" (12/7/11)

Papers

1. S. Jitsukawa, "Needs for methodology development to estimate deformation and fracture of intensely irradiated components" (received 12/31/2011).
2. D. Ryutov, "Importance of assessing thermal fatigue of fusion blanket components caused by the normal variability of neutron flux" (received 12/23/2011).
3. D. Babineau, "Questions to be considered" (received 12/22/2011).
4. D. Babineau, "The Need for a Method for Processing Highly Tritiated Water" (received 12/22/2011).
5. M. Mauel, "Fusion Technologies for Tritium-Suppressed D-D Fusion" (received 12/20/2011).
6. R. Goldston, "A Note on the Relationship Between Neutron-Material Interaction Issues and Plasma-Material Interaction Issues in Magnetic Fusion Development" (received 12/15/2011).
7. K. Young, "Radiation-effects Studies for Instrumentation for Future Magnetic Fusion Devices" (received 12/14/2011).
8. F. Meyer, "Suggestions for FESAC" (received 12/10/2011).
9. R. Buttery, "Understanding and Mitigating the Challenge Posed for Fusion Materials by Testing New Divertor Concepts on Tokamaks" (received 12/07/2011).

10. B. LaBombard, "The scientific challenges at the plasma-material interface in a DEMO will be extreme...." (received 12/05/2011).
11. J. Sienicki, "Advanced Power Conversion Working and Heat Transport Fluids and Cycles for Harnessing Fusion Power" (received 12/05/2011).
12. W. Blink, "Uncertainties in Predictions of Material Performance using Experimental Data that is Only Distantly Related to the System of Interest" (received 12/05/2011).
13. M. Ferraris, "Nuclear degradation of materials and structures" (received 12/05/2011).
14. P. Stangeby, "Research required to develop the option of using carbon PFCs for application to high duty cycle tokamaks" (received 11/22/2011).
15. J. Lew, "Necessity of numerical modeling for pebble bed ceramic breeders in fusion reactors" (received 11/22/2011).
16. M. Fluss, "Enhanced synergistic swelling due to hydrogen: A critical issue for fusion energy materials" (received 11/22/2011).
17. H. Neilson, "International Collaboration in Fusion Material Development" (received 11/10/2011).
18. N. Ghoniem, "Surface Dynamics Of Plasma-Facing Materials" (received 11/8/2011).
19. L. Elguebaly, "Need for Online Adjustment/Control of Tritium Bred in Blanket" (received 11/8/2011).
20. T. Weaver, "Structures for Liquid Lithium Walls" (received 11/4/2011).
21. C. Wong "Misc Comments" (received 11/3/2011).

E-mails

1. R. Andreani (received 12/26/2011).
2. R. Iotti (received 11/4/2011).
3. K. Lee (received 11/4/2011).
4. J. Corelli (received 11/4/2011).
5. F. Chen (received 11/4/2011).