ADVANCED MATERIALS PROGRAM

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PROLOGUE

The development of a functional first wall and blanket system for D-T based fusion power plants may well represent the single greatest engineering challenge of all time. Even low-to-moderate performance power systems will place totally unprecedented and unexplored demands on materials and structures at all levels of integration. From a structural viewpoint, these staggering demands primarily relate to simultaneous: (a) high, time-varying thermal-mechanical loads on heat transfer structures with large, complex and difficult to fabricate geometries; and (b) degradation of a host of performance sustaining mechanical properties (e.g., strength, ductility and toughness), internal damage development and macroscopic cracking. These processes are driven by exposure to an incredibly hostile environment typically combining high temperatures and stresses, corrosive and reactive chemical interactions and intense high energy neutron radiation fields. The unique 14 MeV component of the fusion neutron spectrum is not currently available for materials development and testing.

The challenge is further amplified by the existence of a multitude of potential failure paths and processes, ranging from the mundane, like small and commonly ubiquitous pressure boundary leaks, to catastrophic, massive plant level failures. Completely new issues will also be present, like the inherent dimensional instabilities in large interconnected structures due to the unavoidable phenomena of irradiation swelling and creep.

It must be clearly understood that the materials and in vessel structural issues are not simply a matter of an orderly effort to characterize and extend material performance limits, although these certainly must be first-order objectives. Rather these issues relate fundamentally and explicitly to the basic feasibility of any approach to practically sustaining power production from a fusion reaction. The strong, concurrent relation of these 'materials-structures' issues to both general approaches to developing a burning fusion core and the design, construction and reliable operation of the surrounding breeding and energy conversion first wall, blanket and divertor systems cannot be challenged. Thus a sustained, focused and scientifically sound materials research and development program is paramount to demonstrating the inherent feasibility of fusion power.

If there are any questions about the seriousness of these challenges, it is only necessary to review the 'materials-structures' problems associated with the development of breeder reactors; or even the myriad of materials issues facing current LWRs that operate under an almost infinitely more benign set of conditions.

However, there is nothing that inherently dictates that meeting these challenges is beyond our reach. It is important that the materials development program establish very ambitious, and perhaps revolutionary, long term goals, with the recognition that this will also require evolutionary progress based on a program of the highest quality of science and engineering.

MISSION, SCOPE, AND LONG-TERM GOAL

The long-term goal of the Advanced Materials Program (AMP) is to develop structural materials that will permit fusion to be developed as a safe, environmentally acceptable, and economically competitive energy source. This will be accomplished through a science based program of theory, experiment, and modeling that (1) provides an understanding of the behavior of candidate material systems in the fusion environment and identifies limiting properties and approaches to improve performance, (2) undertakes the development of alloys and ceramics with acceptable properties for service in the fusion environment through the control of composition and microstructure, and (3) provides the materials technology required for production, fabrication, and power system design.

Historically, the AMP has focused on development of materials for the first wall and blanket structures in magnetically confined fusion power systems. The focus on structural materials stems from the fact that the characteristics and properties of the structural material by and large determine the economics, safety, and environmental attractiveness, and thus feasibility of fusion energy itself. It should be noted that questions relating to the performance of other non-structural materials that have implications for the feasibility of fusion (e.g., Radiation Induced Electrical Degradation of Ceramic Insulators) have been addressed in the AMP when they arise. Also, questions that relate to possible interactions of the structural materials with other materials present in the first wall/blanket system (e.g., coolants, tritium breeding materials) that would lead to degradation in the performance of the structural material are addressed in the AMP. The focus on magnetically confined systems was consistent with the mission of the Office of Fusion Energy Sciences until approximately two years at which time interest in inertially confined fusion energy concepts (IFE) began to increase and IFE became a larger part of the OFES strategy. There are many generic issues in materials science and technology that are common between MFE and IFE approaches which are being addressed in the present program. However, there are materials issues related to the pulsed nature of the radiation flux and thermomechanical loading characteristic of the IFE concept that will have to be addressed as IFE power system designs evolve and materials performance requirements are more clearly defined.

STRATEGIC PATHWAY

Background

For fusion to find its way into the energy marketplace it must compete economically with other energy options, and it must be developed as a safe and environmentally acceptable energy source, particularly from the view point of radioactivity. Achieving acceptable performance for a fusion power system in the areas of economics, safety and environmental acceptability, is critically dependent on performance of the blanket and divertor systems which are the primary heat recovery, plasma purification, and tritium breeding systems. Design and performance of these key components is critically dependent on the properties and characteristics of the structural materials. Temperature limits imposed by the properties of materials are the major limitation in the quest for high thermal efficiency. The major in-vessel systems will have a finite lifetime and will require remote maintenance and replacement. High reliability over the entire plant life-cycle is key to favorable economics and requires protecting against frequent minor functional failures and the imposition of severe operational limits. Along with lifetime, reliability is primarily determined by the performance of materials. Materials and structures must also provide acceptable safety margins for protection against off-normal events. Radioactive isotope inventory, and release paths are key considerations in designing for safety. The initial levels of radioactivity of materials on removal from service and the rate of decay of the various radioactive isotopes, dictate acceptable storage and disposal methods and the possibility of recycle of materials, both being major considerations in the environmental acceptability of fusion.

Approach to Materials Development - The Challenge

The properties of any alloy system are directly dependent on the composition and the microstructure at length scales from the interatomic to the continuum. Within inherent limits, the microstructures (or composite architectures) of structural materials can be manipulated to optimize some, but usually not all, relevant properties. Thus, along with the selection of a general material system, tailoring microstructures provides a viable approach to developing suitable materials for fusion. Structural alloys usually contain five or more constituents, two or more phases and complex dislocation and grain boundary defect structures. The detailed morphology and distribution of these features mediate the properties. Useful microstructures almost always represent a non-equilibrium thermodynamic state that evolves with time. These evolutions not only involve redistributions of the microstructures that exist at the start of service, but also the formation of new features and micro and macro damage which often leads to property degradation, cracking, increased failure probability and ultimately lifetime limits. Thus alloy design for sustained performance is an exceedingly difficult and time consuming undertaking, even for conditions that do not involve exposure to high energy, high fluence neutrons.

When irradiation damage processes are considered, the challenges are enormously amplified. Exposure to neutrons violently displaces atoms from their crystal lattice positions, up to hundreds of times over an operating lifetime and changes alloy chemical compositions by transmutation, including the introduction of damaging concentrations of reactive (e.g., H) and insoluble (e.g., He) gases,. Thermal degradation processes are often accelerated under irradiation, and entirely new types of microstructural and damage structures are introduced.

Characterizing and ultimately predicting these processes in a fusion environment and their consequences to the range of important properties is far beyond the scope of current science and engineering. The complexity of developing requisite mechanism-based material designs and performance predictions is clear when one considers the number of controlling variables, and synergistic variable combinations, the range of interacting physical phenomena and the enormous extent of time and length scales of importance. The latter are from that of femto scale neutron-atom interactions to the gigasecond lifetime of multi-meter structures. Steady progress will require a disciplined, long term experimental program buttressed with understanding provided by a close integration with theory and modeling.

Selection of Material Systems for Development

Selection of material systems for development as a fusion power system blanket structural material is based upon the general performance targets listed in Table 3.1.1. The conceptual determination of which material systems have potential to meet these performance targets is made through an interaction between the Advanced Design Studies and Advanced Materials Program tasks. Prior to the establishment of safety and environmental attractiveness goals, the program considered and eliminated Mo, Ni, Ti, and Al based alloys. The basic reason for elimination of these alloy systems was: for Mo, low temperature irradiation embrittlement and fabricability; for Ni, high temperature grain boundary embrittlement; for Ti, excessive tritium inventory and permeability; and for Al, limited temperature capability. Niobium based alloys were eliminated, and Mo, Ni, and Al alloys would be eliminated, on the basis of environmental impact considerations, e.g., any alloy containing more than a few parts ppm Nb could not be disposed of by shallow land burial. For designs utilizing bare metal walls, austenitic stainless steels are eliminated because of inadequate thermal-physical properties. At this point in time there are three material systems that remain and are judged to have potential of being developed as fusion power system structural materials: SiC composites, V based alloys and advanced ferritic steels. Copper alloys, because of their excellent thermal and electrical conductivity, are critically important in near term applications and will most likely find special applications in fusion power systems.

Opportunities for consideration of new material systems may arise in the future as a result of advances within the broad field of materials science, or new design concepts that permit additional choices of material systems that have potential to meet performance goals. The decision to undertake a research and development effort on a new material system would be made jointly by the Advanced Design and Advanced Materials Programs after a concept study had shown the potential of the material system and associated design concepts.

Major R&D Tasks and Interactions

The Advanced Materials Program is structured around the material systems that have the greatest promise to yield structural materials for fusion in-vessel components. Presently, the program has four major materials systems R&D tasks:

I. SiC Composites,
II. Vanadium Based Alloys,
III. Advanced Ferritic Steels, and
IV. Copper Alloys and Other Special Purpose Materials

An additional overarching task is

V. Mechanistic Research and Modeling Materials Behavior.

The integrated theory, modeling, experiment and data base development effort involves a number of key phenomena and issues, such as the effects of helium and its interaction with other irradiation and environmental effects, that cut-across a number of the material systems.
The materials development process or strategy can be viewed as consisting of three major overlapping and interconnected steps.

Feasibility Studies. Material systems with potential for fusion power system first wall and blanket (FWB) structural applications are identified through studies carried out in the Advanced Design and Advanced Technologies activities in collaboration with the Advanced Materials Program. Once identified, the AMP undertakes investigation of the basic properties of the material system and a 'best estimate' assessment of its response to the fusion operating environment, based on a combined application of the available tools. This is used to establish preliminary design windows and to identify feasibility issues and the properties that are most likely to be limiting. Research is also undertaken to begin to explore the relation between composition, microstructure, and properties. This provides preliminary assessments of the extent to which the critical and limiting properties might be improved. This information is fed back to the design studies. Most important, this research focuses the direction and possible approaches for materials development.

Materials Development. Materials development is the process of identifying more specific composition(s) and microstructure(s) that will result in a metallic alloy or ceramic with mechanical and physical properties sufficient to support design, construction, and operation. Fundamental understanding of materials systems and their response to the operating environment is insufficient to prescribe a composition and microstructure a priori. The process is experimental and iterative but builds upon understanding developed through interpretation of experimental results with state-of-the-art theory and models. In particular, practical engineering progress and the effectiveness of experimental efforts to develop a development level data base is greatly facilitated by understanding of common, cross-cutting phenomena obtained by the integration with theory, modeling and mechanism experiments. The development process initially focuses on obtaining maximum performance for those properties that clearly limit application. As development proceeds, however, all aspects of materials performance that are critical to success must be included. This would include primary production, fabrication, welding and joining, chemical compatibility with coolants, tritium breeders and all other operating environments, and the range of mechanical properties essential to acceptable performance. Since neutron irradiation can deleteriously affect most physical and mechanical properties it becomes the overarching consideration. The end product of the materials development activity is a composition and microstructure specification for the range of alloy and/or ceramic system that will support economic, safe, and environmentally attractive fusion power. Throughout the development process there is an interaction with the Advanced Design

Studies which leads to a focusing of materials choices and improvement in fusion power system designs. Along with a more sophisticated understanding of materials performance, interactions with the design community should take place that involve detailed component level benchmark studies using state-of-the-art computational tools of engineering science (e.g. , neutronics, thermal hydraulics, inelastic structural analysis and so on). This activity not only sharpens understanding of the real requirements and performance limits for materials systems, but also integrates the materials research effort with the development of advanced structural assessment methods, in areas such as fatigue lifing and fracture mechanics, that will be critical to reliable and practical designs.

Materials Engineering. Materials engineering will provide the extensive materials property data base required for design, licensing, construction, and operation of a large complex fusion power system. Production will be scaled up. Properties must be determined for each of multiple full commercial scale production runs for all pertinent reference materials and product forms required in the power system. All aspects of materials performance including response to a large number of synergistically interacting environmental/irradiation and pertinent metallurgical variables (including temperature and stress, neutron flux, fluence and spectrum, trace impurities and as-fabricated product forms). Consideration must also be given to realities of the operating environment such as varying temperatures and stresses and off normal and accident conditions. Data will be required for the prototypic operating environment and postulated off-normal events from the beginning to end-of-life conditions. The product of Materials Engineering is a specification for producing materials in the required product forms, an approved data base on properties and structural assessment methods to support design construction, and licensing, and a reliable basis to predict how materials will perform throughout the expected lifetime.

It is important to emphasize that the materials development and optimization will not end with the successful operation of the first, large, commercial-type, power producing fusion system. Clearly, it is of critical importance to achieve the most ambitious goals for that first step. However, consistent with all technological advances, , materials development can be expected to continue in a evolutionary sense, as experience of the real environment is gained, and new materials and designs concepts can be introduced in a modular sequence. The rate of that evolution will depend critically on the status of the fundamental materials knowledge base at that time.

Status and Critical Issues

SiC Composites. Silicon carbide composites are under development for aerospace and fossil energy applications for which high temperature strength, strength to weight ratio, and corrosion resistance are the most important properties controlling system design and performance. For fusion applications, interest in this material stems not only from its potential for low activation properties, but also from its potential for mechanical strength at very high temperatures (≥1000°C) which could, for example, make possible a very high efficiency direct cycle helium cooled concept. To date fusion related research on SiC composites has focused on the effects of irradiation on dimensional stability, strength, and thermal conductivity. The first work explored the response of state-of-the-art composites based on commercially available Nicalon-CG fibers™, carbon interfaces, and a chemical vapor infiltrated (CVI) SiC matrix. Significant degradation of strength, attributed to shrinkage of the Nicalon-CG fiber and of the carbon interface, has been observed at relatively low levels of displacement damage. Thus the first attempts to improve the performance of the SiC composite structure incorporated a more dimensionally stable fiber and interface. Low oxygen SiC fibers with higher crystalline perfection than Nicalon-CG (Hi-Nicalon, Nicalon-S, and Dow Corning Sylramic) were found to have improved dimensional stability upon irradiation. Alternate fiber/matrix interfaces; porous SiC and multilayered interface of alternating layers of SiC and C, have been developed. The second generation composites utilizing advanced fibers and interfaces have higher strength in the unirradiated condition and exhibit smaller reductions in strength than the Nicalon-CG/carbon interface composites after irradiation. Development of an increased database on the physical and mechanical properties of SiC/SiC composites will result from the continued measurement of baseline properties within the fusion and other programs. Some fundamental baseline properties include high-temperature fracture strength, creep and thermal conductivity.

Efforts to understand and improve the irradiation performance of SiC composite structures should also address other fundamental questions relating to the performance of SiC. Issues associated with dimensional stability and strength decreases with displacement damage plus helium must be measured and understood. All of the radiation damage studies conducted to date have investigated only displacement damage. In the fusion spectrum, in addition to displacement damage, significant amounts of He will be produced. At the first wall, the He production rate is approximately 1500 appm He/(MW a/m2). The production rate decreases rapidly with distance into the blanket structure. Using He implantation, Hasegawa et al. found that 150 to 170 appm He (the amount produced in ~ 0.1 MW a/m2) decreased the strength of a SiC composite by about 20%. At high temperatures SiC undergoes significant volumetric swelling. The temperature dependence of volumetric swelling, the effects of high damage levels on strength, and the effects of high concentrations of helium, up to at least 15,000 appm, upon the magnitude and temperature range of swelling and strength degradation, are critically important questions with answers that could significantly influence the approach to development of SiC composites for fusion.

Another critical issue is the reduction of thermal conductivity that occurs in monolithic SiC and in SiC composites as a result of neutron irradiation damage. Thermal conductivity in SiC is primarily by phonon transport (unlike metals where thermal conductivity is dominated by electron transport) and the irradiation-produced point defects and small point defect clusters act as phonon scattering centers. The reduction of thermal conductivity relative to the unirradiated value is particularly large at low irradiation temperatures (e.g., 100°C) and decreases as the irradiation temperature is increased, reflecting the size and concentration of irradiation produced defects. The reduction of thermal conductivity from point defect production saturates at very low damage levels (at low irradiation temperatures at less than a dpa). The effects of fusion-relevant helium generation rates in combination with displacement damage have not been investigated. After irradiation at 1000°C the thermal conductivity of high purity monolithic SiC is approximately 20 W/m-K and the highest value for a SiC composite structure is ~ 10 W/m-K. With highest priority the AMP should carry out the research required to provide fundamentally sound estimates of the thermal conductivity of composite structures of the thermal conductivity of composite structures of the absolute highest quality that might be achieved in the irradiated condition. An assessment should then be made with the design community of the feasibility of using this material as the primary structural material in a large structure with significant thermal loads.

In addition to the development of a SiC composite that will retain adequate mechanical and physical properties in the fusion environment, there are several technological problems that must be solved. The CVI manufacturing process produces a microstructure that has ~10% porosity. In-service microcracking is inherent and associated with the SiC composite toughness, but may make the structure becomes permeable to gas.. Methods must be developed and demonstrated to seal the material and reduce the permeability to acceptably low levels. Methods of joining SiC to itself and to metals must also be developed. Joints will be subjected to the same operational conditions as the SiC structure itself. It should be anticipated that a significant amount of field fabrication and joining will have to be accomplished. Technology for manufacturing components or structures much larger than present state of the art must be developed.

Possibly the most difficult challenge is the development of designs that permit the use of an inherently brittle material like SiC composites. While this is more straightforward in simple geometries, including those that involve thermal loading, there is no analytical or experimental assessments to demonstrate if this can be extended to large interconnected thermally and inelastically loaded structures that will have significant gradients in dimensional changes due to thermal expansion, creep, and, potentially, swelling. If this possibility can reasonably be demonstrated in principle, then a major effort will be needed to develop the associated design methods and codes. One must bear in mind that the these issues were a major challenge in the design of breeder reactor piping systems, based on the use of highly forgiving, ductile stainless steels. Today’s design codes for metal structures (e.g., the ASME design codes for boilers, pressure vessels, nuclear systems) have evolved over many decades and are the product of literally thousands of man-years of engineering, construction, and even many more years of operational experience.

Vanadium Alloys. Vanadium alloys have several inherent properties or characteristics that make them attractive candidates for fusion reactor blanket structural applications. The combination of relatively high thermal conductivity coupled with relatively low thermal expansion and low elastic modulus yields low thermal stresses and thus high heat flux capability. V-Ti-Cr alloys have excellent compatibility with pure lithium making them the material of choice for a liquid lithium coolant/breeder blanket concept. The elements V, Ti, Cr, and Si which make up most vanadium alloys offer the potential for low activation. In fast reactor irradiations at temperatures in the range ~ 400 to 600°C, vanadium alloys, exhibit low irradiation induced swelling and adequate tensile ductility.

Loomis and coworkers at ANL carried out investigations of the effects of Cr and Ti content on the tensile and Charpy impact properties of V-Cr-Ti alloys. Additions of Cr and Ti, separately or together, increase the yield strength, ultimate tensile strength, thermal creep strength, and the DBTT as measured in a Charpy impact test. A trade off in the desire for a low Charpy DBTT (the assumption being that a low DBTT in the unirradiated condition would yield a lower DBTT after irradiation) and high strength led to the selection of an alloy with nominal composition V-4Cr-4Ti as a reference composition. Several small (~ 15 kg) and two large (500 kg and 1200 kg) heats have been melted and fabricated by the U.S. Fusion Program. Mechanical and physical properties have been determined.

A major focus of research and development of vanadium alloys is to understand and develop alloys with acceptable radiation damage resistance. For irradiation temperatures of 400°C and below, the yield strength is increased to levels approaching the ultimate tensile strength and the uniform strain is reduced to values below 1%. The reduced ductility is associated with deformation patterns that evolve from homogeneous strains in the unirradiated condition to highly localized flow in the hardened conditions. For irradiation temperatures below 400°C, irradiation hardening causes a significant upward shift in the DBTT. For example, the 500 kg heat of V-4Cr-4Ti irradiated at 400°C to 4 dpa has a DBTT in 1/3 size blunt notch Charpy samples that is above 290°C. Fracture toughness testing of the V-4Cr-4Ti alloy irradiated at low temperatures also suggests a severe reduction in the toughness at higher testing temperatures normally associated with stable crack growth. Over the approximate range, 400 to 600°C, the magnitude of the radiation-induced increase of yield stress decreases and the yield stress approaches the unirradiated level. A critically important temperature regime is 400 to 500°C. In this temperature range a transition from low tensile ductility and poor fracture toughness to higher ductility and fracture toughness occurs.

At high irradiation temperatures (>~650°C) the primary concern relative to the effects of irradiation on mechanical properties is helium embrittlement which causes a loss of tensile and creep ductility and creep rupture life resulting from the growth and coalescence of helium bubbles at grain boundaries. There have been no investigations of the effects of helium on creep rupture properties of vanadium alloys and no investigations of the effects of helium on the candidate V-4Cr-4Ti alloy [except in the dynamic helium charging experiment (DHCE)]. At helium concentrations typical of those produced in a fusion spectrum, helium embrittlement could reduce high temperature creep rupture life and ductility which would reduce allowable stresses and/or impose lower temperature limits. This is an area that should be addressed in near-term fusion materials research; if helium embrittlement occurs in the V-4Cr-4Ti alloy it may have a significant impact on the alloy development strategy.

A liquid Li-cooled/Li-breeder concept is the most attractive fusion reactor blanket for use with a vanadium alloy structural material. In a liquid metal cooled magnetically confined concept an electrically insulating coating is required on the surface of coolant channels/ducts to reduce large pressure drops associated with magnetohydrodynamic (MHD) forces. The coating must be chemically compatible with the vanadium alloy and with the lithium coolant at reactor operating temperatures and for blanket lifetimes. It must be assumed that the coating will be damaged during operation, e.g., by thermal cycling, and thus there must be a mechanism for self healing during service. The requirement for chemical compatibility with Li limits the practical choices for insulating coatings; A1N and CaO are considered the two leading candidates. It has been demonstrated that these materials can have adequate insulating capacity in thin layers (a few μm). Practical methods of applying the coatings, achievement of adequate mechanical integrity and adherence, retaining electrical insulating capability, compatibility with the Li coolant and the vanadium alloy, and a practical approach to produce a self healing system are still to be developed. This is a critically important area relative to the feasibility of a V alloy-liquid Li coolant/breeder concept.

Interstitial impurity elements such as oxygen, carbon, and nitrogen can significantly affect the mechanical properties of vanadium and its alloys. These impurities can be introduced into the alloy during melting, fabrication, welding and joining processes, service in partial vacuum or gaseous environments, and by mass transfer in liquid metal cooled systems. Effects include an increase in yield strength, loss of tensile ductility, and low toughness cleavage or grain boundary fractures. The magnitude of the effect is strongly dependent upon the distribution of the impurity element(s) in the microstructure. Contamination with nonmetallic impurities has been a continuing major concern relative to the technological use of vanadium alloys. Contamination by minute levels of impurities (e.g., oxygen, water vapor, etc.) in a helium coolant is the key feasibility issue for a helium cooled/vanadium structural material system. It is also known that interstitials can significantly effect basic mechanical behavior and play a crucial role in many aspects of microstructural evolution during irradiation. Indeed, interstitial impurities may have beneficial effects on high temperature creep strength. Research on the kinetics of nonmetallic element contamination, their behavior and effects on properties of the alloy, should continue with high priority. This research should have two fundamental goals. First, to determine and understand the effects of interstitial impurities on the range of important mechanical properties including low temperature fracture behavior, and creep rupture; and second, to determine the feasibility of using a V-Ti-Ci alloy structure in any partial vacuum or gaseous cooled (He) system.

Advanced Ferritic Steels. Ferritic steels have the most advanced technology base of the three materials presently being considered for fusion blanket structural applications. High temperature ferritic steels (having a tempered martensite structure), e.g., Fe-9Cr-1MoVNb and Fe-12Cr-1MoVW, were developed for fossil boiler and steam generator applications and were investigated for use as cladding and duct materials in liquid metal reactors (LMR). The data base on irradiation induced swelling, irradiation creep, and effects of irradiation on tensile and Charpy impact DBTT developed in the LMR programs throughout the world is extensive and shows that the alloy class generally exhibits favorable irradiation response at least for LMR conditions characterized by high displacement damage, (but little helium), and temperatures above about 370°C. The lower irradiation temperature (< 400°C) response of these alloys share many of the characteristics of the vanadium based alloys discussed above. These include large increases in yield stress that diminish with increasing irradiation temperature and a severe loss of tensile ductility and flow localization at high yield stress levels. The irradiation hardening and ductility reduction is also manifested in upward shifts in DBTT and reductions in upper shelf toughness.

In the development of low activation ferritic steels the international fusion materials programs have narrowed their focus to steels having Cr in the range 7-9%, 1-2% W and V and Ta as carbide forming elements. The lower Cr avoids severe hardening and embrittlement problems associated with the formation of fine dispersions of the α' phase. The behavior of low activation alloys is generally similar to the Modified 9Cr-1Mo steel from which they were derived. One aspect of performance that is potentially different, however, is the DBTT. The reduced activation steels generally have shown a lower DBTT and the shift induced by irradiation is generally less than most of the existing data for the standard Modified 9Cr-1Mo steel (note, however, that some recent data on a standard French T91 alloy shows similar or better DBTT behavior than reduced activation F82H).

A refined assessment of the suitability of using advanced steels as fusion reactor blanket structural materials should focus on two important areas. The first relates to possible effects of helium upon the fracture properties. Some experiments conducted in the United States, Japan, and European Community indicate that helium, produced by (n,α) transmutations, can act in addition to, or synergistically with, hardening caused by displacement damage and precipitation to cause a shift in DBTT that is larger than that attributable to displacement damage and precipitation alone. It is critically important to establish if, and by what mechanisms, helium affects low temperature fracture behavior, to obtain estimates of the magnitude of the effect(s) at helium concentrations relevant to fusion blanket lifetimes and to define the lower bound on acceptable operating temperature.

The second key area of research relates to the upper temperature limit for use, which is usually estimated to be 550°C based on creep properties. Increasing this temperature to 600 or 650°C would expand the design window and make the alloys much more attractive for helium and liquid metal cooled concepts. Development of dispersion strengthened alloys, metallurgical approaches to creating a tempered martensite structure that is stable to higher temperatures, or development of new precipitation strengthened ferritic alloys are possible approaches. The concept of a advanced high temperature nanocomposited ferritic materials (NFM) appears to be particularly attractive from the perspectives of both high temperature strength and irradiation stability. Indeed, recent developments suggest that early studies of NFM systems are warranted and should focus on the potential trade-offs in toughness. Other issues associated with the use of advanced ferritic steels under high temperature irradiation relate to the stability of the coarse scale substructures, formation of intermetallic embrittling phases and reductions in creep rupture properties at high levels of helium. As with V alloys, use of ferritic steels with a liquid metal coolant will require development of a MHD insulator.

<u>Copper Alloys.</u> High strength copper alloys with high thermal conductivity provide the most effective means of transferring high heat fluxes in plasma facing components. These materials are used extensively in the design of the ITER first wall, limiter, divertor, and baffle components. Because of their unique capacity to handle high heat fluxes, it is important to include these materials in the program even though they do not meet currently adopted criteria for low activation. In addition, high strength copper alloys are the primary candidate material for the center-post magnet in low aspect ratio tokamak designs.

During the ITER project, several problems emerged which impact the performance of both precipitation-hardened and oxide-dispersion-strengthened alloys. For example, the precipitation-hardening alloys (e.g., Cu-Cr-Z) are susceptible to loss of strength during brazing or HIPping cycles and during irradiation at temperatures >300°C. The oxide-dispersion-strengthened alloys (e.g., GlidCop Al25) are less susceptible to these phenomena but suffer from anisotropy in fracture properties and undergo severe radiation-induced loss of fracture toughness following low dose irradiation at temperatures <300°C. It is clear that further modifications to alloy compositions, microstructures, and fabrication methods are needed to address these issues and to tailor one or more of these alloys for the fusion environment.

<u>Mechanistic Research and Modeling Materials Behavior</u>. The goal of this task is to develop an increased emphasis on modeling, primarily on the context of improving the integration of theory, experiment, and database development as part of the long-term effort

to build a knowledge base. The expanded computational materials effort will (a) focus on the key issues in fusion materials, (b) be closely integrated with the experimental program, (c) be as collaborative as possible and be leveraged by other activities, (d) support the production of tangible useful results (e.g., more reliable predictions of materials properties), and (e) provide the scientific basis for improved design and methods of structural integrity assessment.

In-service performance will be mediated by a combination of factors, including temperature, stress, the chemical environment and the neutron flux, fluence, and spectrum. It is well known that there are synergistic interactions among this multitude of variables; and that complexity is further amplified by the overall temporal history of the service environment. Thus, as a reasonable and necessary expedient, this effort will start with relatively simple models for selected properties, with ongoing development leading to more robust and comprehensive models as the knowledge base is improved. Finally, the integrated modeling program should also provide information useful for advanced design and structural integrity assessment method development. For example, a better understanding of the micromechanics of the ductile-to-brittle transition is needed to develop better embrittlement models, as well as improved engineering-code-based structural integrity assessments such as the master curve method.

Generally, the underlying physical basis of these models will be provided by first understanding microstructural evolution, including the mechanisms that lead to the production, transport, and fate of defects and key constituent species as a function of the material and irradiation-service variables. The microstructure must be linked to basic structure-sensitive properties, like the yield stress, and such basic properties must in turn be linked to other engineering properties, like fracture toughness. Microstructural information and mechanical property data developed within the experimental component of the AMP, as well as relevant data generated by other programs such as Basic Energy Sciences (BES), the Nuclear Regulatory Commission (NRC), and the wider materials community, will provide the basic information needed for model development and verification. This implies developing and applying:

- integrated multiscale models of microstructural evolution and materials properties,
- mechanistic models of key phenomena,
- an appropriate mix of mechanical property testing and microstructural characterization,
- experiments to investigate specific mechanisms (e.g., to explore helium transport, fate, and consequences),
- the verification of model predictions with results of integral database experiments, and
- physically based, engineering correlation models from evolving databases.

A key product commitment is the development of mechanical performance maps (similar to Ashby maps) by systematizing and condensing the information in the knowledge base. This will also help to provide an interface between materials research and engineering design studies.

The principal features of a fully integrated program are summarized in a set of technological challenges in Table 3.2.2. of the Roadmap section. These activities span all length and time-scales from the atomistics of primary damage production and the development of integrated models to predict structural performance.

In order to fashion a highly leveraged and well-coordinated program, a number of key phenomena have been identified that crosscut multiple materials systems. These crosscutting phenomena are presented in the following table. Several of these phenomena play a critical role in the areas identified as feasibility issues for each materials system (Table 3.2.1). To maximize overall impact on program directions, the integrated theory-modeling-experimental program will focus its attention on these crosscutting phenomena.

Key Crosscutting Phenomena for Fusion Reactor Materials *

FM	V	Cu	SiC	Phenomena, Issues, Comments
W	**W**	x	-	hardening and nonhardening embrittlement including underlying microstructural causes and the effects of helium on fast fracture
X	X	X	-	flow localization, causes and consequences including underlying microstructural causes
X	**W**	x	x	helium transport and fate
x	**W**	-	x	coatings, multilayers, functionally graded materials
X	X	x	x	helium effects on high temperature deformation fracture and improved multiphase alloys
X	X	X	X	thermal and irradiation creep
X	X	X	X	swelling and general microstructural stability
X	**W**	X	**W**	welding, joining and processing issues
X	X	X	X	fatigue
X	**W**	X	**W**	hydrogen and interstitial impurity effects on deformation and fracture
			W	physical properties, e.g. thermal conductivity
			W	permeability of gases
X	X	X	X	erosion, chemical compatibility, bulk corrosion, cracking, product transport

***** X -highly relevant, x - probably relevant, **W** - related to feasibility issues

Critical Facilities

Fusion materials must function in a demanding environment that includes various combinations of high temperatures, chemical interactions, time-dependent thermal and mechanical loads, and intense neutron fluxes.

One of the major materials issues in developing attractive fusion power is the effect of the intense neutron fluxes associated with deuterium/tritium fusion concepts. The first-wall neutron spectrum that contains a large 14-million-electron-volt (MeV) component not only results in very high displacement rates (about 20 dpa per year at 2 megawatts per square meter), but it also causes much higher transmutation rates than those experienced in fission reactors. The elements helium and hydrogen are of particular concern, but other impurities can also be important. The fact that these transmutation products can have a significant influence on mechanical and physical properties is well established; the obvious examples are the role of helium in swelling behavior and high temperature ductility and fracture. Thus, neutron irradiation is a particularly important issue because of its effects on physical and mechanical properties, as well as the production of radioactive materials, and is the most difficult to investigate with currently available facilities.

At present, fission reactors, such as the High-Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory, and the BOR-60 Fast Reactor at the Research Institute of Atomic Reactors (RIAR) in Dimitrovgrad, Russia, are the primary means of investigating the effects of irradiation on fusion materials. Significant resources are required for the design and fabrication of specialized assemblies for irradiation of test specimens and for operation of hot cell facilities for disassembly of experiments and remote testing of the irradiated specimens. Techniques such as tailoring of the neutron energy spectrum, or tailoring the isotopic composition of the alloy to control transmutation rates are being used to more nearly reproduce the fusion environment. In addition, specialized studies using ion beams can be conducted. However, the response of materials to these various radiation fields can be quite different from that which is due to a fusion neutron spectrum and a 14-MeV neutron source will ultimately be needed to develop and qualify fusion materials.

A suitable neutron source must reproduce the fusion spectrum, particularly in terms of the ratio of transmutation products to atomic displacements. In addition, the neutron source must have flux and fluence capabilities that are sufficient to allow accelerated testing of fluences up to end-of-lifetime. The international community has proposed an accelerator facility based on the deuterium/lithium interaction ƒ the International Fusion Materials Irradiation Facility (IFMIF) ƒ to fill this role. The characteristics of the IFMIF are summarized in the table below. Development and qualification of structural materials in such a device would be required before embarking on a high-fluence fusion-based system.

Leveraging the Broad Field of Materials Science Research

Fusion materials research is an extremely small fraction of the broad field of materials research. Our strategy is to focus on those areas of technological need that are unique to fusion and to leverage other areas of materials science research to the maximum extent possible. Examples of areas of materials science in which we rely entirely on other programs for developments include: advanced techniques for structural characterization such as analytical electron microscopy, atom probe-field ion microscopy, synchrotron x-ray facilities, development of advanced materials synthesis and processing techniques, and basic studies of the relationship between structure, composition and properties of metals and ceramics. Funding for research in these and related areas comes primarily from DOE-Basic Energy Sciences, NSF, and DOD.

Characteristics of the IFMIF Neutron Source

Characteristic	Specification	
Neutron flux/volume	>5 MW/m^2	100 ml
	>2 MW/m^2	400 ml
	>1 MW/m^2	>6000 ml
Particle	D+	
Beam current	2 @ 125 mA	
Beam energy	32, 36, or 40 MeV	
Beam spot	5 cm × 20 cm	
Duty factor	100% (cw)	
Plant factor	70%	

Chemical interactions between the structural materials and other materials in a power system, such as corrosion of the structural materials by the coolant, oxidation in gaseous environments, mass transfer within the system via the coolant, interaction with tritium breeders, must be determined and understood, and the impact of such phenomena on design and performance of the fusion power concept must be assessed. Determination of what chemical interactions are potentially important, understanding the chemical thermodynamics and kinetics, and providing the basic information for design studies is a fundamental function of materials science and engineering and within the scope of the Advanced Materials Program. In the feasibility and materials development steps of the AMP this research requires specialized facilities such as high temperature gas reaction systems for oxidation studies, thermal convection and small pumped loops to investigate corrosion and mass transfer, phenomena with liquid metal coolants, and in some instances specialized mechanical test systems to investigate dynamic effects of chemical interactions on mechanical properties.

There are areas of materials science which the advanced materials program leverages by support of research, often to use directly or to modify and adapt developments in the field for use by fusion. To a very large extent the broad topic of modeling of materials behavior is in this category. Similarly, the DOE Energy Efficiency and Fossil Energy Programs have significant programs for the development of SiC composites that are leveraged by the Advanced Materials Program.

Working with Others

International collaborations are the hallmark of the Advanced Materials Program. Formal bilateral collaborations that involve joint planning, funding, testing and data analysis, and reporting of materials research and development have been in existence with Japan Atomic Energy Research Institute and Monbusho (Japanese Universities) for over 15 years. These collaborations have been crucially important to the vitality of the U.S. Fusion Materials Program. An important bilateral collaboration exists with the Russian Federation which provides access to the RF fast reactor BOR-60 for radiation effects research. The international fusion materials research and development effort is coordinated through the IEA Implementing Agreement on a Programme of Research and Development on Fusion Materials. Canada, Japan, the EU, Switzerland, the PRC, the Russian Federation, and the United States are participants. Fusion materials research activities coordinated through the IEA include SiC composites, vanadium alloys, advanced ferritic steels, ceramic insulators, conceptual design of IFMIF, test specimen miniaturization technology, irradiation damage theory and modeling. Within the national laboratories and universities the fusion materials program is generally embedded in the organizational structures (divisions, departments, etc.) in which other materials R&D is carried out. This co-location provides extensive interactions with materials scientists and engineers working on materials problems ranging from basic to applied.

Program Quality, Innovation, and Focus

Success in the development of materials for fusion in-vessel structural applications requires a program that is focused on the critical issues, seeks innovative approaches in understanding how the fusion operating environment will affect the properties of materials and in developing materials that will function in this environment, and maintains the highest levels of quality in its research and development endeavors. The Advanced Materials Program will continue to maintain quality, innovation, and focus by combining approaches that have worked in the past with new approaches and techniques that address issues brought about by the Virtual Laboratory for Technology concept.

The Fusion Materials Program Leaders, consisting of representatives from the largest national laboratory and university program elements, will continue as a group to plan and coordinate the Advanced Materials Program.

The Task Group structure for planning of activities within the individual program elements will be maintained. At the present time, the following task groups have been established and are functioning:

Vanadium Alloy Task Group
Ferritic Steels Task Group
SiC/SiC Composites Task Group
Ad Hoc Group for Modeling Planning and Workshop
Alloys for High Heat Flux Applications
Materials/Design Interface, Data Base, and Handbook

The Advanced Materials Program will continue to publish a semi-annual report of progress. This serves as an excellent record of the many important details and findings of the program that do not find their way into the open literature.

The very successful effort to develop and foster innovation with US industry through the SBIR program will continue.

The Program will establish a standing review committee and conduct formal program wide reviews on a periodic (18-24 month) schedule. The review committee will be drawn from the broad field of materials science and will have national laboratory, university, and industry representatives.

The Program will continue its move towards a more science based program by redirecting a portion of its budget into an initiative on Experiment, Theory, and Modeling to provide the foundation of understanding upon which a long term materials development effort can be built.
The two major international collaborations, US/JAERI and US/Monbusho, will be coordinated and directed through formal steering committees consisting of senior U.S. and Japanese materials scientists.

ROADMAP AND DELIVERABLES

Performance Targets

The Advanced Materials Program is focused upon the development of structural materials for blanket systems and other in-vessel components of fusion power reactors. The primary reason for this focus is that it is the blanket system more than any other system in the fusion power plant that impacts the attributes of economics, safety, and environmental attractiveness. Further, within the blanket system it is the structural material along with the breeding material that dictates the design concept and performance characteristics. From concept evaluation and design studies that have been carried out over the past 2-3 decades, fundamental performance targets for the blanket structural material can be established. These targets are listed in Table 3.1.1.

Technological Challenges

Since the scientific and technological knowledge and experience base for the three candidate material systems that have been identified (SiC composites, vanadium alloys, and advanced martensitic steels) are radically different, they will have markedly different development paths. However, it is most important to realize that fundamental feasibility issues that could preclude use in fusion power systems remain for all three material systems. The major remaining technological challenges for establishing feasibility are listed in Table 3.2.1.

In addition to the feasibility issues listed in Table 3.2.1 it is necessary to provide a fundamental understanding of the behavior and response of these materials in the fusion environment to guide materials development activities which are in the early stages and to support the future materials engineering activities. The core of this understanding will be physical based theories and models that describe the key aspects of materials behavior and phenomena. The roadmap for this activity is presently being developed. Table 3.2.2 identifies some key areas in which improved understanding and predictive capability is needed.

Although, as discussed in section 2.d, it is convenient to view materials development as consisting of three steps: (1) Feasibility Studies, (2) Materials Development, and (3) Materials Engineering, there is no sharp line of demarcation between these steps. As problems/limitations are identified in the feasibility step we begin to address them through materials development. The technological challenges identified at the present time that are being addressed through Materials Development are shown in Table 3.2.3.

TABLE 3.1.1

PERFORMANCE GOALS FOR
FUSION REACTOR STRUCTURAL MATERIALS

· Develop materials that will support economically attractive fusion power systems
  - fabricability, weldability, joining technology for field assembly
  - compatibility with operating environment (coolant, tritium breeder, ambient atmosphere, special purpose coatings, hydrogen plasma)
  - adequate performance design window (operating temperatures limits, stress limits, irradiation damage)
  - Reliability and maintainability
  - Component lifetime

· Develop materials that will achieve the potential of fusion as an environmentally attractive energy source
  - materials not requiring long term geological disposal
  - potential for recycle of materials to minimize environmental impact

· Develop materials that will achieve the potential of fusion as a safe energy source from the viewpoint of the worker and the public
  - acceptable levels of heat from radioactive decay and chemical reactions
  - limited dispersability of radioactivity
  - acceptable biological hazard potential (BHP)

TABLE 3.2.1

TECHNOLOGICAL CHALLENGES TO ESTABLISH FEASIBILITY

Silicon Carbide Composites

- Develop understanding of radiation damage in monolithic SiC and SiC fibers with and without gaseous transmutation products, He and H
- Improve radiation stability in SiC fiber/interphase/SiC matrix structures which currently leads to degradation of mechanical properties
- Develop and demonstrate joining technology applicable in large fusion systems
- Overcome adverse impact on allowable heat fluxes due to low thermal conductivity of current composites and reduction of thermal conductivity as a result of irradiation
- Identify major compatibility issues with coolants and breeders and establish R&D plan to quantify temperature limits based on compatibility
- Develop design methodology or experience for large structures that are thermally, hydraulically, and mechanically loaded
- Demonstrate techniques for production of large components
- Improve hermeticity of composite structures and quantify permeability of gases
- Develop a database on fundamental properties such as high-temperature fracture strength, creep and thermal conductivity

Vanadium Alloys

- Develop and demonstrate self-healing MHD insulator coating for Li cooled/breeder systems
- Develop welding methods applicable to manufacture and field construction of large systems that yield weld structures with acceptable properties and radiation resistance
- Establish potential design window for alloys ¾ focus on effects of radiation on constitutive and fracture properties in the temperature range of 400-500°C and creep-rupture properties and effects of irradiation (neutron damage and helium transmutation) on creep rupture properties at high temperatures (>650°C)
- Develop a fundamental understanding of the role of impurities (including solid and gaseous transmutation products) in the mechanical behavior and in radiation response
- Improve understanding of the kinetics of pickup of interstitial impurities and their effects on properties to assess feasibility of using He coolant

Advanced Martensitic Steels

- Verification that ferromagnetic structural materials are acceptable for MFE concepts
- Expansion of the low temperature-dose operating regime through (a) experimental measurements integrated development with physically-based modeling of the combined role of displacements, helium, and hydrogen on toughness and constitutive properties, and (b) development and integration of advanced mechanism-based design methods (master curve-shift procedure)

- Expansion of the high temperature-dose operating regime through investigation of concepts involving (a) improved tempered martensite stability, (b) new precipitation-strengthened ferritic alloys, and (c) nano-composited ferritic materials
- Resolution of system-specific compatibility issues (e.g., T barriers for He or PbLi coolants)

Copper Alloys

- Development of an isotropic high toughness GlidCop material with improved resistance to radiation-induced degradation
- Development of a high-temperature precipitation-hardened alloy system that will resist thermal and radiation-induced softening

TABLE 3.2.2

TECHNOLOGICAL CHALLENGES IN THE DEVELOPMENT OF A FUNDAMENTAL UNDERSTANDING OF THE BEHAVIOR OF STRUCTURAL MATERIALS IN THE FUSION ENVIRONMENT

Fundamentals of Radiation Damage Production

- Development of calibrated interatomic potentials
- Properties of primary defects
- Primary damage production

Damage Accumulation and Microstructural Evolution

- Atomic transport
- Nucleation and growth of extended defects (helium bubbles, voids, dislocations)
- Radiation induced solute segregation
- Alloy phase stability during irradiation

Fundamental Behavior of Candidate Material Systems

- Thermodynamics and phase equlibria
- Influence of alloying elements and impurities on microstructure and important properties
- Corrosion and materials compatibility

Understanding and Predicting Properties Critical to Performance

- Swelling and irradiation creep
- Deformation and fracture behavior
- Integrated models for predicting structural performance
- Integrated models for predicting corrosion and compatibility in complex systems

TABLE 3.2.3

TECHNOLOGICAL CHALLENGES FOR THE
MATERIALS DEVELOPMENT STEP

Silicon Carbide Composites

- Develop composites structures with advanced SiC fibers/alternate interphases/SiC matrixes with improved thermal conductivity and irradiation damage resistance

Vanadium Alloys

- Develop metallurgical approaches to expand the design window

Advanced Martensitic Steels

- Explore oxide dispersion strengthening as an approach to improving the high temperature strength of martensitic steels while maintaining acceptable low temperature toughness characteristics

Copper Alloys

- Develop high strength copper alloy with improved high temperature fracture toughness

Focus of Research in the Advanced Materials Program

From the Performance Targets and the Technological Challenges that have been identified in achieving these targets, the focus of research activities in the Advanced Materials Program from the present to 2005 have been identified and are listed in Table 3.3.1.

TABLE 3.3.1

FOCUS OF RESEARCH IN THE ADVANCED MATERIALS PROGRAM

Silicon Carbide Composites

- Determine the effects of displacement damage (fission reactor irradiations) on the critical properties of monolithic SiC and advanced SiC fibers at temperatures in the range 500 to 1000°C
- Explore the effects of high concentrations of helium on the microstructure and mechanical properties of monolithic SiC and state of the art SiC composites
- Synthesize and evaluate the mechanical behavior (e.g., high temperature fracture and creep properties) and irradiation response of SiC composites using advanced fibers and interphases
- Select and evaluate candidate joining methods
- Explore intrinsic and extrinsic thermal conductivity concepts for improving the thermal conductivity of SiC/SiC composites

Vanadium Alloys

- Develop an MHD coating system that has adequate electrical properties, is stable in the V alloy-Li coolant system for relevant times and temperatures, and has adequate self healing characteristics
- Determine properties of welds made by e-beam, laser, and GTA techniques as a function of welding environment. Establish "best achievable properties for V-4Cr-4Ti alloy and investigate irradiation response of these welds
- Determine high temperature (>600°C) creep rupture properties of V-4Cr-4Ti alloy and investigate the effects of irradiation and helium on these properties
- Determine research on irradiation effects in V alloys at temperatures in the range 300 to 700°C with particular focus on: mechanisms and role of impurities in microstructural evolution and fundamental mechanisms for changes in mechanical properties
- Establish the kinetics of impurity pickup from vacuum, He and Li coolant environments at conditions relevant to fusion power systems and the effects of these impurities on properties that are critical in design and performance of a structure
- Explore alternative alloy compositions and approaches for the V system to address problems associated with interstitial effects, irradiation embrittlement, and improve high temperature strength

Advanced Martensitic Steels

- Determine the effects of displacement damage concurrent with helium on the low temperature deformation and fracture behavior of F82H and closely related compositions
- Explore oxide dispersion strengthening as an approach to improving the high temperature strength of martensitic steels while maintaining acceptable low temperature toughness characteristics

Determine effects of high concentrations of helium on creep-rupture properties

Copper Alloys

- Develop metallurgical approaches to improving the fracture toughness of the ODS alloy GlidCop CuAl-25
- Investigate the relationship between flow properties and the fracture properties of Cu-Cr-Zr and GlidCop Al25 irradiated to low doses

Fundamental Understanding of Materials Behavior

- Development of master curve approach for characterization of fracture behavior and design for fracture resistance
- Development of multiscale models of damage evolution under irradiation

PROPOSED ACTIVITIES/DELIVERABLES

FY 1999 Activities/Deliverables

In the latter part of FY 1998 and in FY 1999 the materials program will make adjustments in priorities and direction as recommended in the FESAC review, in the VLT workshops, and in program planning meetings. The first order deliverables for FY 1999 are listed below.

- A workshop will be held (late FY 1998 or early FY 1999) for the purpose of roadmapping and integrating an expanded theory and modeling effort into the materials development program.
- The Master Curve approach for correlation of fracture toughness, test, and irradiation parameters will be experimentally verified for a vanadium alloy and a ferritic/martensitic steel.
- The unirradiated creep characteristics of a V-4Cr-4Ti alloy will be determined in the range 600 to 800°C uniaxial and biaxial test methods
- Kinetics of oxygen and hydrogen interaction and their effects on mechanical properties will be characterized
- Investigations will be completed on the effects of irradiation on the tensile strength and fracture behavior of V-4Cr-4Ti alloys for irradiation temperatures to 425°C levels of ~5 dpa
- The irradiation stability of monolithic SiC, advanced fibers, and interface materials will be determined using low fluence, HFIR rabbit tests
- The effects of neutron irradiation on the thermal conductivity and mechanical properties of advanced (second generation) composites will be determined using low fluence, HFIR rabbit tests
- An assessment of low activation joining technologies applicable to SiC composites will be initiated
- The ferritic steel activity will be focused on an experimental program to determine the role of helium and reactor spectrum on the low temperature embrittlement of advanced ferritic steels. Preliminary results will be reported by the end of FY 1999

FY 2005 Deliverables

Deliverables for the period FY 2000 to FY 2005 are listed below.

<u>Theory and Modeling</u>

- Create a framework of theory, modeling, and experiment, for damage production and microstructural evolution involving irradiation and its effect on physical and mechanical properties as part of the advanced materials effort

Vanadium Alloys

- Establish acceptable weld technologies for V alloys
- Model kinetics of H and O interactions and effects on properties to establish environmental operational limits
- Characterize and model displacement damage and He effects to damage levels of ~10 dpa (350-750°C)
- Complete an assessment of the viability of generating a self-healing, electrically insulating coating on vanadium alloys for liquid lithium systems with Advanced Technology Task

SiC Composites

- The effects of neutron irradiation to 5 dpa on microstructure, strength and fracture properties of 2nd generation composites will be determined
- Mechanical properties and irradiation effects will be modeled and understood in terms of fiber, matrix, and interface behavior
- The effects of irradiation on the thermal conductivity will be determined and modeled including materials optimized for high thermal conductivity
- Reach a preliminary conclusion on the feasibility of SiC composites for structural applications based on advanced design performance goals
- Evaluate the radiation response of optimum low activation joining methods
- Determine the radiation creep behavior of advanced fibers and model the creep response of composites
- Determine the chemical stability of 2nd generation material in fusion relevant environments

Ferritic Steels

- Establish the low temperature limit for advanced ferritic steels based on experiment and modeling of the He-dpa effects on fracture properties
- Establish the potential of ODS ferritic/martensitic steels as fusion blanket structural materials

The General Correlation of the FY 1999 and FY 2005 Deliverables with the Roadmap

Technical Challenges is shown in Figure 1.