Final Technical Report for DOE Award DE-FG02-99ER54513

Nuclear Technologies for Near Term Fusion Devices

01/01/1999 - 01/1/2017

Report No. DOE-UW-54513

Office of Fusion Energy Sciences
Office of Science

Principal Investigator: Mohamed E. Sawan

University of Wisconsin-Madison

Collaboratoring Investigators:

Gerald L. Kulcinksi
Paul P.H. Wilson
Andrew Davis
Tim D. Bohm

Contents

Chapter 1

Introduction

Over approximately 18 years, this project evolved to focus on a number of related topics, all tied to the nuclear analysis of fusion energy systems.

For the earliest years, the University of Wisconsin (UW)'s effort was in support of the Advanced Power Extraction (APEX) study to investigate high power density first wall and blanket systems. A variety of design concepts were studied before this study gave way to a design effort for a US Test Blanket Module (TBM) to be installed in ITER. Simultaneous to this TBM project, nuclear analysis supported the conceptual design of a number of fusion nuclear science facilities that might fill a role in the path to fusion energy.

Beginning in approximately 2005, this project added a component focused on the development of novel radiation transport software capability in support of the above nuclear analysis needs. Specifically, a clear need was identified to support neutron and photon transport on the complex geometries associated with Computer-Aided Design (CAD). Following the initial development of the Direct Accelerated Geoemtry Monte Carlo (DAGMC) capability, additional features were added, including unstructured mesh tallies and multi-physics analysis such as the Rigorous 2-Step (R2S) methodology for Shutdown Dose Rate (SDR) prediction.

Throughout the project, there were also smaller tasks in support of the fusion materials community and for the testing of changes to the nuclear data that is fundamental to this kind of nuclear analysis.

1.1 About this Report

This report summarizes the major findings over the duration of the project. For many sections, text has be excerpted from publications that described this work in more detail, with references provided for those publications. The following chapter describes a variety of nuclear analysis efforts, followed by a chapter on the development of improved predictive capability, and then one chapter each on nuclear data improvements and fusion materials support.

The references are divided into two sections: those that serve as research products of this project, in whole or in part, and those that are outside this project, but related in some way.

Chapter 2

Nuclear Analysis of Fusion Energy Systems

2.1 Advanced Power Extraction (APEX)

This project began as part of the APEX study that had the goal of identifying a first wall and blanket system that could withstand much higher power densities than conventional systems, while still meeting the shielding, cooling and tritium breeding needs for viable fusion energy. With neutron wall loads >10 MW/m^2, surface heat fluxes of >2 MW/m^2 and thermal efficiencies >40%, such concepts were expected to greatly increase the economic attractiveness of fusion energy. These concepts varied from flowing particulate systems, to flowing liquid first walls of various compositions, to solid walls with high heat capacity fluids flowing near the first wall.

A typical nuclear analysis considers a range of important engineering responses to the radiation flux, including tritium breeding, energy multiplication, local heating and radiation damage in the blanket system itself, and heating and damage to sensitive components such as magnets. With the addition of an activation calculation, it is possible to consider decay heat during off-normal conditions, dose rates during maintenance, and waste disposal ratings over the long term.

2.1.1 Advanced Plasma-facing Particulate Li_2O Evaluation (APPLE)

The first concept was a flowing Li_2O particulate blanket concept without a structural first wall. Neutronics calculations were performed for the Advanced Plasma-facing Particulate Li_2O Evaluation (APPLE) concept using a neutron wall loading of 10 MW/m^2. Steel structure was used in the shield and vacuum vessel. The Li_2O in the blanket had a packing fraction of 0.6. A minimum blanket thickness of 40 cm was required for the steel structure to be a lifetime component. The radial build that satisfied requirements for steel structure lifetime, vacuum vessel (VV) reweldability, and superconductor magnet shielding was determined. The achievable overall TBR was estimated to be >1.2 taking into account the difference in blanket thickness in the regions surrounding the plasma. Although the steel structure and VV were considered lifetime components, the SiC baffles and separation walls, which are not structure members, experienced a large damage rate and were designed for quick replacement. More than an order of magnitude reduction in decay heat and activity resulted from placing the structure behind the Li_2O particulate blanket. Using low activation ferritic steel structure behind the Li_2O particulate blanket allowed for near surface burial of the radwaste. It was concluded that the Li_2O particulate blanket concept without a structural first wall had the potential for achieving tritium self-sufficiency with lifetime structure and reweldable vacuum vessel. It had attractive safety features resulting from the significant reduction in radioactivity and decay heat generation in the structural material.[1]

2.1.2 EVaporation Of Lithium and Vapor Extraction (EVOLVE)

Another early concept relied on the evaporation of liquid lithium at the first wall to absorb the energy and protect the solid components. The neutronics performance of the EVOLVE concept was analyzed using two-dimensional calculations. Using tungsten alloys yields ~10% higher TBR than with tantalum and was used in the reference design. The overall TBR with tray zones having a radial thickness of 50 cm in the outboard (OB) region and 40 cm in the inboard (IB) region is 1.37 at a lithium enrichment of 40% ^{6}Li. This value assumes 75% outboard blanket coverage, 15% inboard blanket coverage, and no breeding in the divertor region. Tritium breeding has a comfortable margin that allows for design flexibility. The amount of nuclear heat deposited in the different regions was determined. Most of nuclear heating (~72%) is deposited in the high temperature front blanket. Adding the surface heat deposited in the first wall (FW) implies that ~76% of the total IB and OB energy is deposited as high-grade heat in the front evaporation-cooled zone (FW and trays) and carried by the Li vapor to the heat exchanger. Since this heat is extracted at a temperature of 1200°C, a thermal efficiency >60% in the power conversion system can be achieved by employing a closed cycle helium turbine system.

Nuclear heating and radiation damage profiles were calculated. No significant poloidal peaking is observed. The peak damage rate in the OB secondary blanket and IB shield is about a factor of ~6 lower than in the FW for which a damage rate of ~35 dpa/year has been calculated. This implies that they are expected to have a factor of 6 longer lifetime than the FW and trays. The lifetime of the OB shield is about an order of magnitude longer than for the OB secondary blanket and the IB shield making it a lifetime component. The radial build required for VV reweldability and magnet shielding was determined.[2, 3, 4]

2.1.3 Solid Wall with FLiBe Coolant

Another design option considered in the Advanced Power Extraction (APEX) program featured a solid nano-composite ferritic steel (NCF) alloy FLiBe (Li_2BeF_4) acting as a combined breeder and coolant. Work done under this award included both the structural design of at least two such concepts, as well as the analysis of neutronics performance.

The spiral blanket is an innovative design of a solid FW concept utilizing NCF steel in combination with FLiBe coolant. Thermal hydraulic evaluation has shown that it is capable of dissipating an average neutron wall loading of 6.4 MW/m^2, a peak neutron wall loading approaching 10 MW/m^2 and surface heating of 1.3 MW/m^2, while satisfying the temperature limits ascribed to NCF steel of 800°C maximum and 700°C interface with FLiBe. The pressure drop is a modest 0.56 MPa and the stresses for 12 YWT NCF steel are well within limits with time effects taken into account. The overall tritium breeding ratio (TBR) is 1.33 using natural Li and the energy multiplication is 1.26. The amount of tritium bred in the Be over the estimated blanket lifetime is 1.5 kg. However, at the operating temperature of the Be, most of the tritium will diffuse out. Fabrication of the blanket using forming techniques and diffusion bonding appears to be possible. Using an inlet FLiBe temperature of 500°C, an outlet from the primary blanket of 590°C and after traversing the rear blanket, exiting at 600°C, and assuming a supercritical steam power cycle with double wall heat exchanger, an efficiency of near 48% can be expected. Finally, this unconventional innovative design shows that solid wall blankets have a legitimate place along with liquid protected walls as a first line of defense facing an intense neutron plasma.[5]

Another concept relies on a simpler geometric design, with a lead (Pb) neutron multiplier. A description of the geometric and engineering aspects, and preliminary stress analysis of the re-circulating blanket was developed. It was shown that the time-dependent and time-independent primary stresses as well as the primary and secondary stress limits were satisfied at all points in the blanket. Material, and material compatibility issues were addressed and solutions offered. Fabrication possibilities and coolant circuits were presented. Even though the re-circulating solid wall blanket is somewhat complicated, its aimed to maximize nuclear parameters to achieve high conversion efficiency.[6, 7, 8]

Neutronics calculations were performed to determine the relevant nuclear performance parameters for the blanket. With an enrichment of 40% ^{6}Li the overall tritium breeding ratio is expected to be ~1.16 and the blanket energy multiplication is 1.13. Nuclear heating profiles were determined for the different components of the blanket and used in the thermal hydraulics analysis. The peak structure dpa and helium production rates are 77.6 dpa/FPY and 955 He appm/FPY, respectively, implying a lifetime of ~3 FPY. Calculations were performed to determine the radial build in both the inboard and outboard regions required to provide adequate shielding for the vacuum vessel and toroidal field (TF) magnets. The NCF structure dominates the total activity and decay heat. The Mo content in the NCF needs to be reduced from 0.02% to <0.01% for the structure waste to qualify as low level class C waste. While the waste disposal rating (WDR) of the FLiBe is well below unity, the Pb has to be circulated at a very small flow rate (<1 cm^3/s) to remove the generated ^{208}Bi and allow Pb disposal as low level waste.[9]

2.1.4 Tritium Breeding Ratio

In comparing different potential molten salts, a key performance criteria is the TBR. The molten salt LiF–NaF–BeF_2 (FLiNaBe) with ratio 1:1:1 has been suggested as the front flowing liquid layer (FFLL) in the convective liquid flow first wall (CLiFF) high power density concept. FLiNaBe has intrinsic properties that favor its use as the FFLL. The concern raised with regard to its potential for tritium breeding has been examined in this paper and comparison was made to the FLiBe (LiF:BeF_2 , with ratio 2:1) in two arrangements: (1) as a FFLL and breeder in the blanket; and (2) as only the FFLL with LiPb/SiC conventional blanket following the FW. Two types of structure were also considered, ferritic steel (FS) and SiC. In the first configuration and with FS structure, the local TBR in FLiNaBe is less than in FLiBe by ~15% and by ~9% at natural Li and 90% Li-6 enrichment, respectively. The TBR maximizes at 50% Li-6 (FLiNaBe) and at 25% Li-6 (FLiBe) with values ~1.04 and ~1.16, respectively. Using SiC as the structure material reduced TBR even further (~8%). A front beryllium multiplier must be introduced in the blanket for both breeders. Significant improvement in TBR is achieved with the Be zone. To achieve a TBR of 1.4, the required Be zone thickness is ~6 (FLiBe) and ~10 cm (FLiNaBe) with SiC. This is equivalent to an effective thickness ~4 (FLiBe) and ~6 cm (FLiNaBe) of solid Be zone. If a conventional LiPb/SiC blanket is employed, the local TBR with FLiNaBe FFLL is less than FLiBe by ~3%. With this 2-cm thick layer, tritium self-sufficiency (TBR ~1.4) can be achieved with LiPb enriched to ~35% Li-6 or higher. Thus, tritium self-sufficiency is not a feasibility issue for FLiNaBe if used as a FFLL and breeder (with Be multiplier) or as a FFLL only with enriched LiPb blanket. This was shown to be the case when account is made to all sources of uncertainties in the achievable and required TBR.[10]

2.1.5 Activation and waste disposal

A more detailed study was performed to understand the impact of switching to liquid walls on the volumes and classification of waste. Structural waste volume and hazard were compared in a thick liquid-wall (LW) Li/V-4Cr-4Ti system with maximum neutron wall load of 10 MW/m^2 and in a conventional solid-wall (SW) Li/V blanket with maximum neutron wall load of 5 MW/m^2. The comparison was made for two configurations, namely: (1) Fixed Radii where the LW and SW first wall/blanket (FW/B) have the same plasma and FW radii, and (2) Fixed Fusion Power with the LW FW/B having half the plasma and FW radii and hence twice the neutron wall load of the conventional SW FW/B. Shield optimization was performed to satisfy the same acceptable damage parameters in the magnet. The objectives were to quantify the advantage of using the thick LW blanket option over a conventional SW system in reducing disposed waste volume and its hazard.

The analysis indicated that the total waste volume from the machine (including magnets and VV) were dominated by waste from the shield (~50-63%). The FW/B contributed ~22% of the total waste volume in the conventional SW blanket and ~6% in the two LW options. The structure waste volume per GW_{th} was almost the same for both LW options. The waste volume per GW_{th} of the conventional SW FW/B was larger than that in both LW options by a factor of seven. However, the total waste volume per GW_{th} in the conventional SW option was ~2.14 larger than the value in both LW options. Regarding WDR, the results for the two LW options were identical. The WDR values for the LW and conventional SW concepts were comparable. All components were classified as Class C low level waste. Values for activity and decay heat per MW_{th} were comparable for the LW and conventional SW blanket options for the permanent components. However, values in the FW/B zone of the SW were much higher due to the much larger structure content.[11, 12, 13]

In addition to waste disposal issues, another major concern with using FLiBe and FLiNaBe in fusion blankets is that transmutation of the constituent elements leads to the production of the highly corrosive free F and the relatively less corrosive TF. Controlling the activities of free F and TF is essential for consideration as viable breeder/coolant candidates. Neutronics calculations were performed for blanket designs using either FLiBe or the low melting point FLiNaBe to determine the transmutation rates of constituent elements and the rates of production of other elements. The results showed that at least from mass balance considerations no free F will be left provided that the recombination reactions with freed Be, Li, Na, and produced tritium are fast enough. However, more than 95% of the tritium bred will be in the form of TF. The results indicate also that O and N are produced at about 7% and 14% of the rate of tritium breeding. Enrichment of Li was found to have minor impact on mass balance with the conclusions remaining the same. The results imply that we only have to worry about chemistry control of the less corrosive TF. Be can be used to reduce both TF and F_2 to BeF_2 and T_2.

The transmutation rates and the thermodynamics define the chemistry conditions of the FLiBe and FLiNaBe under neutron irradiation. These conditions need to be reproduced in the experimental setup at INEEL to study the thermodynamics and kinetics of the REDOX reaction. After the REDOX condition is established, the material corrosion experiment can be carried out to assess the compatibility between the molten salt, with proper chemistry control, and different structural materials.[14]

2.2 ITER Test Blanket Module (TBM)

A wide variety of activities related to the neutronics performance of a potential ITER TBM were pursued, ranging from high level consideration of R&D strategies and needs to detailed analysis of specific design options.

2.2.1 Neutronics Testing in ITER

One such high level assessment was a discussion of strategies and requirements placed on the machine operation (fluence, wall load, etc.) and on the TBM geometrical arrangement (size, dimension, spatial locations) in order to have a meaningful neutronics testing in ITER. Neutronics tests planned to be performed in ITER aim at resolving key neutronics issues such as the demonstration of tritium self-sufficiency and verification of the adequacy of calculation tools and nuclear data bases in accurately assessing the nuclear environment. Although each ITER partner will perform these tests inside their designated TBM that may utilize different breeders, coolants, and structure, neutronics issues to be resolved are generic in nature and are important to each TBM type. Dedicated neutronics tests specifically address the accuracy involved in predicting key neutronics parameters such as tritium production rate (TPR), volumetric heating rate, induced activation and decay heat, and radiation damage to the reactor components. In addition, neutronics analyses are required to provide input support for other tests (e.g. heating rates for thermo-mechanics tests). While tritium self-sufficiency for a given blanket concept can only be demonstrated in a full sector, as envisioned in a DEMO, including a closed tritium fuel cycle, testing in ITER TBM can provide valuable information regarding the main parameters needed to assess the feasibility of achieving tritium self-sufficiency.

From instrumental consideration, all neutronics parameters, except induced activation, can be performed in either low fluence (~1 MW·s/m^2) or very low fluence (~1 W·s/m^2) operation mode. The low fluence level could be achieved, for example, with a wall load of 0.0025 MW/ m^2 and 400 s pulse. In ITER, the wall load at the TBM is ~0.78 MW/ m^2 , which is much larger than the 0.0025 MW/ m^2 value. One pulse or two is adequate for these tests.

Unlike the case of using engineering scaling to reproduce DEMO-relevant parameters in an "act-alike" test module, dedicated neutronics tests require a "look-alike" test module for a given blanket concept. Furthermore, measured neutronics data requires a high spatial resolution. This necessitates the measured quantity to be as flat as possible in the innermost locations inside the test module. In the present work, we confirmed this requirement based on results from two-dimensional calculations in an $R-\theta$ model that accounts for the presence of the ITER shielding blanket and the surrounding frame of the port where the US and Japan TBMs are placed. It is shown that the profiles of the TPR and heating rates have flat values over a range of 10-20 cm in the toroidal direction and a range of ~1 cm in the radial direction inside the breeding layers of the helium-cooled pebble bed (HCPB) TBM. [15, 16]

2.2.2 Tritium self-sufficiency

In considering the design of an ITER TBM, we identified the need for a comprehensive review of the physics and technology conditions for attaining tritium self-sufficiency for the DT fuel cycle. There is no practical external source of tritium for fusion energy development beyond ITER, and all subsequent fusion systems have to breed their own tritium. Tritium self-sufficiency in DT fusion systems cannot be assured unless specific plasma and technology conditions are met. We addressed these conditions and shed light on a possible phase-space of plasma, nuclear, material, and technological conditions in which tritium self-sufficiency can be attained.

It is crucial that the tritium fractional burnup in the plasma be kept high, at least above a minimum of 2% and most preferably above 5%. Thus, plasma edge physics modes that lead to higher tritium recycling need to be explored. A reserve tritium inventory to keep fueling the plasma and continue reactor operation during periods of malfunction of the tritium processing system is necessary. To keep the reserve inventory, and hence the required TBR, sufficiently low requires high reliability/availability of the tritium processing system, and redundancy in some of the tritium processing system, especially the plasma exhaust processing line. Design options that minimize tritium inventories in reactor components such as the blanket, FW, and divertor are needed.

Up to 30% reduction in TBR could result from using 20% structure in the blanket. Hence, it is necessary to accurately determine the amount and configuration of structure required to ensure structural integrity of the blanket under normal and abnormal conditions. Practical FW thickness and blanket structure content based on detailed structural-mechanical and thermal-hydraulics analyses need to be well defined. Accurate definition of other blanket design considerations that introduce uncertainties in the TBR (e.g., using separate coolant and/or neutron multiplier, and the need for electric insulator) is necessary. Using stabilizing shells and conducting coils for plasma control and attaining advanced plasma physics modes should be examined carefully to minimize the impact on tritium breeding. The size and materials used in plasma heating and current drive components and fueling and exhaust penetrations impact the TBR. Use of strong neutron absorbers in these systems should be eliminated or minimized and design options that minimize streaming path should be considered. Calculation of the TBR should be based on detailed 3-D models that account for all design details. Neglecting heterogeneity effects results in errors up to ~10% in predicting the TBR. Integral experiments are needed to validate and improve nuclear data.

This analysis led to the conclusion that it is necessary to urgentlyestablish an extensive parallel and highly interactive R&D program in plasma physics, plasma control technologies, plasma chamber systems, materials science, safety, and systems analysis to determine the phase-space of plasma, nuclear, material, and technological conditions in which tritium self-sufficiency can be attained.[17]

2.2.3 Molten Salt Design Variations

Early design concepts for a US ITER TBM considered molten salts as breeding coolants. The design and analysis of the above specific concept led to an interest in a number of different molten salt based design concepts. Neutronics assessment was performed for a number of molten salt breeding blanket concepts that could be utilized in fusion power plants. Special attention was given to concepts that could be developed, qualified and tested in the time frame of ITER. The conventional ferritic steel alloy F82H with a temperature limit of 550°C is considered as the structural material. The concepts evaluated were a self-cooled FLiNaBe blanket with Be multiplier and dual-coolant blankets with He-cooled FW and structure. Several options were considered for the dual-coolant concept, including using Be or Pb multiplier. In addition, three different molten salts were considered including the high melting point FLiBe, a low melting point FLiBe, and FLiNaBe. Several iterations were made to determine the blanket radial build that achieves adequate TBR. Larger margins were considered to account for uncertainties resulting from approximations in modeling. The same TBR can be achieved with a thinner self-cooled blanket compared to the dual-coolant blanket. A thicker Be zone is required in designs with FLiNaBe. The overall TBR will be ~1.07 based on 3-D calculations and excluding breeding in the divertor region. Minor design modifications can be made to enhance the TBR if needed to ensure tritium self-sufficiency. We concluded that the molten salt design concepts have the potential for achieving tritium self-sufficiency. Using Be yields higher blanket energy multiplication. A modest amount of tritium is produced in the Be (~3 kg) over the blanket lifetime of ~3 FPY. Using He gas in the dual-coolant blanket resulted in about a factor of 2 lower blanket shielding effectiveness. With a total blanket/shield/VV radial build of 105 cm in the IB and 120 cm in the OB it was possible to ensure that the shield would be a lifetime component, the VV would be reweldable, and the magnets would be adequately shielded. Based on this analysis we concluded that molten salt blankets can be designed for fusion power plants with neutronics requirements such as adequate tritium breeding and shielding being satisfied.[18]

2.2.4 Expanded Liquid Breeder Assessment

As candidate blankets for a U.S. Advanced Reactor Power Plant design, and in consideration of the time frame for the ITER development, we assessed first wall and blanket design concepts based on the use of reduced-activation ferritic steel (RAFS) as structural material and liquid breeder as the coolant and tritium breeder. The liquid breeder choice includes the conventional molten salt Li_2BeF_4 and the low melting point molten salts such as $LiBeF_3$ and FLiNaBe. Both self-cooled and dual coolant molten salt options were evaluated. We have also included the dual coolant lead-eutectic Pb-17Li design in our assessment. We based our first wall and blanket assessment on an Advanced Reactor Power Plant design, which has a maximum neutron wall loading of 5.4 MW/m^2 and a maximum surface heat flux of 1 MW/m^2 at the outboard mid-plane of the tokamak reactor. Molten salt blankets will require an additional neutron multiplier like Be or Pb to provide adequate tritium breeding. For the dual coolant design, helium is used to remove the first wall surface heat flux and to cool the entire steel structure. The liquid breeder is circulated to external heat exchangers to extract the heat from the breeding zone (a self-cooled breeding zone). We take advantage of the molten salt low electrical and thermal conductivity to minimize impacts from the magneto-hydrodynamic (MHD) effect and the heat losses from the breeder to the actively cooled steel structure. For the Pb-17Li breeder we employ flow channel inserts (FCIs) with a low electrical and thermal conductivity to perform respective insulation functions. For the dual-cooled molten salt blanket options, to avoid the formation of a thin solid layer of high melting point molten salt in the blanket, the utilization of a lower melting point molten salt is recommended. For the lower melting point molten salt FLiNaBe, physical properties like melting point will need to be further established. For the molten salt designs, REDOX control will need to be demonstrated to mitigate the compatibility issue between the generated TF and structural material. Due to the low tritium solubility of molten salt and Pb-17Li, tritium permeation control will be needed to minimize contamination of the environment. For the duel-coolant Pb-17Li design, successful development of the FCIs, such as SiC/SiC composite, is required to arrive at an acceptable MHD pressure drop and to thermally insulate the high temperature self-cooled breeder from the RAFS structure. Results of the above R&D items forms the technical basis for the selection of the reference blanket concept for the U.S. and provide input for the formulation of the U.S. ITER test module program and its corresponding test plan.[19, 20]

As part of this broader analysis, detailed 3-D neutronics calculations were performed for the dual coolant molten salt blanket designs with the low melting point FLiBe or FLiNaBe in a tokamak power plant configuration. The model included the detailed heterogeneous geometrical arrangement of the blanket sectors. The Be multiplier zone thickness was 5 cm with FLiBe and 8 cm with FLiNaBe. The 3-D model included water-cooled steel with tungsten armor in the divertor region. The total TBR was determined to be ~1.07. This was a conservative estimate since it assumed no breeding in the double null divertor zones on which 12% of the source neutrons impinge. We demonstrated that minor design modifications such as increasing the Be zone thickness can be made to enhance the TBR if needed to ensure tritium self-sufficiency. We concluded that the dual coolant molten salt design concept has the potential for achieving tritium self-sufficiency. The calculated TBR that accounts for heterogeneity and 3-D geometrical effects was ~6% lower than estimates based on 1-D calculations. The 1-D calculations tended to overestimate nuclear heating in the blanket by ~8% resulting in overestimating the plant thermal power. The peak dpa and helium production rates in the structure were 28 dpa/FPY and 356 appm/FPY, respectively, and occurred in the OB blanket at mid-plane. Comparing the 3-D results with the 1-D results indicated that the approximate 1-D calculations overestimated damage and nuclear heating in the FW and front zone of the blanket by factors of 1.3-1.7. However, the 1-D calculations resulted in a steeper radial drop in nuclear parameters leading in significant underestimation (by up to a factor of 3) of radiation effects at the back of the blanket. When combined with peaking factors of up to ~3 obtained due to the 3-D geometrical heterogeneity effects, it was concluded that 1-D calculations significantly underestimated radiation damage in the shield and vacuum vessel behind the blanket and large design margins should be allowed when 1-D calculations are used in shielding assessment.[21]

An activation analysis was also performed for these two coolants. Two dual-coolant (DC) blanket concepts were studied by the US for Demo reactor in which helium is used to cool the FW and structure whereas molten salt is used as both coolant and breeder. The F82H conventional steel was used as the structural material. The low melting point FLiBe (~380 °C) was used in the DC/FLiBe option while the FLiNaBe (~305 °C), was used in the DC/FLiNaBe option. The blanket in both concepts had a thickness of 65 cm in the OB (40 cm in IB), including a 5 cm-thick front Be zone (8 cm in case of FLiNaBe). The FW was estimated to last ~5 full power years before replacing the blanket. The radioactivity and decay heat, after shutdown, were assessed separately for the structural material, the Be multiplier and the breeder (FLiBe/FLiNaBe). The total activity and decay heat in the F82H structure was very similar in both concepts. The FLiBe activity in the DC/FLiBe blanket was larger than the FLiNaBe activity in the DC/FLiNaBe blanket by a factor of 1.5-2 during few days to few years after shutdown. However, the decay heat was much larger in the FLiNaBe by up to two orders of magnitude in the time frame of 1 hour-10 years after shutdown. The Class C WDR was estimated for each material. For FLiBe, FLiNaBe and Be the WDR was much lower than unity. However, the WDR for F82H was ~0.6-1.3 due to reactions with Mo and Nb present in F82H with levels of 70 wppm and 4 wppm, respectively. To ensure that F82H qualifies for shallow land burial, it was suggested to reduce these two impurities to ~50 wppm and ~3 wppm, respectively. The results of this work was needed to assess safety concerns such as thermal response during accident conditions and the mobilization of the radiological inventories and site boundary dose following such accidents.[22, 20]

2.2.5 Dual Coolant Lithium Lead (DCLL) Test Blanket Module (TBM) Design and Analysis

Various strategic decisions combined with early scoping analysis results in the selection of a Dual Coolant Lithium Lead (DCLL) TBM concept, requiring more detailed neutronics design and analysis.[23, 24, 25, 26, 27, 28]

Neutronics calculations were performed to determine the relevant nuclear performance parameters for the reference US DCLL ITER TBM. These include tritium breeding, nuclear heating, radiation damage and transmutations, and shielding requirements. The neutron wall loading at the TBM is 0.78 MW/m^2. The front surface area of the module is 0.8 m^2 and the radial depth is 35 cm. The detailed Computer-Aided Design (CAD) model was utilized to divide the TBM into 7 vertical layers and perform calculations for each with the appropriate radial build. Results for the 7 layers were combined using their heights to determine the overall integrated parameters for the TBM.

The calculated TBR in the DCLL TBM is only 0.561 because of the relatively small thickness used. For the planned 3000 pulses per year the annual tritium production in the TBM is 0.97 g. The total nuclear heating in the TBM is 0.574 MW. For the ITER fluence goal of 0.3 MWa/m^2, the peak cumulative radiation damage and He production in the FW are 5.1 dpa and 56 appm, respectively. The corresponding end-of-life values for the SiC FCI are 4.8 dpa and 409 He appm. The dominant metallic transmutation product in the FCI is Mg that builds up to ~100 appm at end-of-life of ITER and its impact on electrical and thermal conductivities of the FCI need to be assessed particularly at elevated fluences in a fusion power plant. We estimated that ~1.2 m thick shield is required behind the DCLL TBM to allow personnel access for maintenance.[29, 30]

Detailed 3-D neutronics calculations were performed for the US DCLL TBM to accurately account for the complex geometrical heterogeneity and impact of source profile and other in-vessel components. The neutronics calculations were performed directly in the CAD model using the DAG-MCNP code that allows preserving the geometrical details. The 20 cm thick frame results in neutronics decoupling between the TBM and adjacent shield modules. The TBM CAD model was inserted in the CAD model for the frame and the integrated CAD model was used in the 3-D analysis. Detailed high-resolution, high-fidelity profiles of the nuclear parameters were generated using fine mesh tallies. The TBM heterogeneity, exact source profile, and inclusion of the surrounding frame and other in-vessel components result in lower TBM nuclear parameters compared to the 1-D predictions. The detailed 3-D analysis with the surrounding massive water-cooled frame and representation of the exact source and other in-vessel components yields total tritium production in the TBM that is 45% lower than the previous 1-D estimate (0.53 g/year). The detailed 3-D analysis of the TBM yields total nuclear heating in the TBM that is 35% lower than the 1-D estimate of 0.574 MW. The detailed 3-D analysis of TBM with the surrounding massive water-cooled frame and representation of exact source and other in-vessel components yields 28% lower peak dpa rate and 10% lower peak He production rate in the FW compared to the 1-D estimates. This is due to the more perpendicular angular distribution of incident source neutrons in the realistic 3-D configuration and reduced neutron multiplication and reflection from surrounding frame and other in-vessel components compared to the 1-D configuration with full coverage with DCLL TBM. This work clearly demonstrates the importance of preserving geometrical details in nuclear analyses of geometrically complex components in fusion systems.[31]

Additional analysis was carried out to consider the consequences of activation/transmutation of such a design. The total radioactive inventory in the DCLL TBM at shutdown is relatively small (2.44 MCi) and drops rapidly within a minute to reach a level of ~0.7 MCi due to the decay of the Pb-207m isotope. It stays at that level for ~1 hr and drops slowly thereafter. The level is ~0.1 MCi after 1 year and is ~0.01 MCi after 10 years. The inventory is almost entirely due to the activation of the F82H structure in the TBM, and in particular, to the structure in the back breeder channel, the back plate, and the shield. A few minutes after shutdown, the activation level in the Pb-17Li breeder is ~2 orders of magnitude lower than the level in the structure, even with the inclusion of the activation of the tritium bred while the activation in the SiC insert is ~2-6 orders of magnitude lower.

At shutdown, the total decay heat is as low as ~0.022 MW. After the decay of the Pb-207m isotope, the total decay heat is attributed mainly to the structure. The total decay heat after 1 hour, 1 day, 1 year, 10 years, and 100 years is 3.5×10^{-3} MW, 1×10^{-3} MW, 1×10^{-4} MW, 2×10^{-6} , and 7×10^{-10} MW, respectively. These are extremely low values and impose no safety concerns. As is the case for the radioactive inventory, the decay heat generated in the FW is not the major contributor to the total decay heat. The decay heat generated in the Pb-17Li breeder is ~2-3 orders of magnitude lower for all times after few minutes following shutdown while the attainable level in the SiC insert is 2-6 orders of magnitude lower than the level in the structure.

The WDR depends on the level of the long-term activation. For the F82H structure, Nb-94, Mn-53, Ni-59, and Nb-91 are the main contributors. In Pb- 17Li, the main contributor is the Pb-205 isotope. The C-14 isotope and the Be-10 isotope are the main contributors in the SiC insert. The WDR values for F82H structure, the Pb-17Li breeder, and the SiC insert according to the conservative Fetter limits are 1.3×10^{-2} , 8.7×10^{-3} , and 2.1×10^{-4} , respectively. They are thus much lower than unity and therefore these materials are qualified for shallow land burial according to the Class C limits.[32]

2.3 Conceptual Design of Fusion Nuclear Development Facilities

The Fusion Nuclear Science Facility (Advanced Tokamak) (FNSF-AT) (originally named the Fusion Development Facility (FDF)) was a small tokamak concept (R = 2.49 m, r = 0.71 m) with copper magnets with consideration for both He and water cooling, and for DCLL and helium-cooled ceramic breeder (HCCB) FW/B systems. Table 2.1 shows the impact of different IB design choices on the IB nuclear parameters. Since the OB responses are not sensitive to the IB design choice, Table 2.2 shows the OB responses as a function of different OB blanket design options.[33, 34]

Table 2.1: Impact of IB design on FDF nuclear parameters

IB Design Option	He-cooled shield		Water-cooled shield	
	DCLL	HCCB	DCLL	HCCB
IB TBR	0.25	0.23	0.23	0.22
OB TBR	0.93	0.83	0.92	0.83
Total TBR	1.18	1.06	1.15	1.05
Peak He appm in SS VV	1.68	0.80	0.31	0.29
Peak He appm in FS VV	0.10	0.15	0.07	0.11
Peak fast neutron fluence in OH coil (10^{19} n/cm^2)	19	12	4.7	4.4
Peak organic insulator dose in OH coil (10^{10} Rads)	21	11	4.6	4.4
Peak fast neutron fluence in TF coil (10^{19} n/cm^2)	7.3	6.2	2.9	3.3
Peak organic insulator dose in TF coil (10^{10} Rads)	2.2	1.2	4.8	4.8
Shield e-fold for organic insulator dose (cm)	7.6	7.6	5.5	6.3
Added shield for organic insulator in OH coil (cm)	~23	~19	~8	~10
Added shield for organic insulator in TF coil (cm)	~23	~19	~8	~10

Three-dimensional analysis led to some design modifications to improve the nuclear performance of this system, with an ultimate TBR of 1.09 for the HCCB option and 1.0 for the DCLL option. These low TBR values are, in part, due to an assumption that no tritium production occurs in the 16 irradiation ports. These analyses also provided updated estimates of peak responses in sensitive components including both TF and poloidal field (PF) coils.[35]

Table 2.2: Impact of blanket design on OB nuclear parameters

	DCLL	HCCB
Peak He appm in SS VV	1.7	0.4
Peak He appm in FS VV	0.10	0.1
Peak fast neutron fluence (10^{20} n/cm^2)	18	1.3
Peak organic insulator dose (10^{11} Rads)	13	1.9
Added He-cooled shield for organic insulator (cm)	~37	~23
Added H_2O-cooled shield for organic insulator (cm)	~27	~19

Chapter 3

Development of Improved Predictive Capability

Over the life of this project, there was a growing recognition of the impact of 3-D features on the nuclear performance of fusion energy systems, whether local material heterogeneity or large scale penetrations that both consume valuable first wall and breeding space and provide streaming paths for radiation to the back of the FW/B system. This provided motivation for the development of new software tools to easily incorporate such 3-D features, ideally by directly being able to use CAD models of the systems in question

3.1 Direct Accelerated Geoemtry Monte Carlo

The fundamental innovation was the Direct Accelerated Geoemtry Monte Carlo (DAGMC) toolkit that can be integrated with any Monte Carlo radiation transport tool to allow the direct use of CAD-based geometry descriptions. While Monte Carlo radiation transport has long been the preferred way to incorporate complex geometry into an accurate physics model, faithful representation of many features was a tedious, time-consuming, and error-prone process. Manual translation of the dimensions, surfaces and volumes that might exist in a CAD drawing introduced analysis bottlenecks that often led to approximations.

The DAGMC[36] toolkit was introduced as part of the open source Mesh-Oriented datABase (MOAB)[65] library for representing surfaces in a mesh-like data structure, and relied on capability in the Common Geometry Module (CGM)[66] library for extracting geometric features. The principle scientific contributions developed as part of this toolkit were a series of accelerations that allowed the Monte Carlo process to proceed on these complex geometric representations at a speed that was comparable to the native analytic repesentation. In particular, the use of oriented bounding box (OBB) trees based on the triangular facets that the solid modeling engine uses to represent each surface, allows for a logarithmic search for the facet that will experience the next intersection. The imprinting and merging capability provided by CGM provides additional benefits, both in reducing the computational cost of crossing a surface, and in allowing for the implicit definition of the very complex non-solid space. The latter feature results in large benefits in the preparation of a model.[37] It was demonstrated and validated against a number of standard experimental benchmarks[38, 39], as well as during a joint exercise based on a simplified ITER geometry[67], and has since been used in the majority of 3-D analysis in this and other projects.[21, 40, 31][68, 69]

Other improvements have focused on robustness, most notably the introduction of the capability to guarantee so-called watertightness of any analysis geometry. This algorithm takes advantage of the topological knowledge embedded in the MOAB representation, and checks every curve in the geometry for logical consistency of the triangles from different surfaces that meet on each curve. If necessary, new triangles are introduced to ensure that each surface has identical discretization along their common curves.[41] Other more subtly improvements were made to the algorithms to improve the robustness for complex CAD-based geometries.[42]

3.1.1 Unstructured Mesh

With complex geometries available for the radiation transport phase of nuclear analysis, it became increasingly important to support the tallying of results on similarly complex geometries. Prior capability required the use of a rectlinear Cartesian mesh that overlays the geometry, with different cells/materials in many of the hexahedral voxels. While various approximations were available for interpreting such results, the most accurate respresentation requires a mesh that conforms to the shape of each shell, composed of tetrahera. Such a capability was added to the DAGMC toolkit and was demonstrated in a number of fusion relevant problems.[70]

3.2 Hybrid Variance Reduction

Many fusion neutronics analyses focus on predicting the effectiveness of shields that protect sensitive components or personnel. Such analyses are inherently challenging for Monte Carlo radiation transport, even though that approach is important to properly capture the 3-D features of a model. By contrast, deterministic radiation transport methods are much better suited to shielding problems, but cannot capture effects that arise with detailed geometric and energy treatments. One solution is to use these methods together, with the more approximate deterministic approach providing acceleration parameters for the Monte Carlo solution. The most widely used implementation of such a concept is known as the Consistent Adjoint-Driven Importance Sampling (CADIS) method.[71]

Using the CADIS method requires the generation of a rectilinear grid representation of the complex CAD-based geometry. We implemented a ray-tracing approach using the same acceleration schemes developed for radiation transport and demonstrated that such an approach is more efficient than prior point sampling approaches, particularly when the geometry includes smaller features.[43, 44]

As the geometries grow larger and more complex, the grid representation also grows, whether covering a larger domain at the same resolution or refining the resolution to capture smaller features. Regardless of the motivation, the size of this grid is limited by two distinct parts of the solution methodology:

1. even with the domain decomposed over many parallel processors, there is a maximum number of mesh voxels that can fit on each process of the deterministic calculation, and

2. there is a maximum size of the acceleration parameters for the Monte Carlo process. The refinement of the grid for the deterministic calculation should not be uniform, but rather should focus on resolving important features and heterogenities. In addition, the second of these limits is often more restrictive than the first, requiring a coarsening of the acceleration parameters from the finest possible mesh achievable in the deterministic calculation. This coarsening should also not be uniform, in order to maximize the utility of the acceleration parameters. This project supported an effort automate both the refinement of the mesh for the deterministic calculation and the subsequent coarsening of the acceleration parameter mesh. This capability was demonstrated on a number of different problems.[45, 46, 47][72]

3.3 Shutdown Dose Rates

Although activation has always been an important consideration in nuclear analysis of fusion energy systems, radiation doses have typically be estimated based on 1-D or cruder approximations. With the ability to represent complex 3-D geometries for radiation transport combined with the licensing of a real project (ITER), there is a strong motivation for predictive modeling of the shutdown does rates in a way that takes account of those complex geometries. One of the primary methodologies for this is known as the Rigorous 2-Step (R2S) methodology because it performs separate radiation transport steps for the neutrons and the activation photons. Over time there have been various implementations with slight differences, but most implementations rely on a high-fidelity mesh over the entire geometry, with an activation calculation at each mesh location. Leveraging the above mentioned capabilities, we have developed two alternatives for performing an R2S analysis: a Cartesian rectilinear mesh overlaying the whole geometry or an unstructured mesh (see Section 3.1.1) conforming to each component. In the former case, the ray-firing approach described in 3.2 is necessary to determine the intial material composition of the activation problem in each mesh voxel. In the latter, a novel sampling technique is necessary to randomly select a position within an arbitrarily shaped tetrahedron.[48, 49] Both of these methodologies were demonstrated and benchmarked using the ITER shutdown dose rate benchmark experiment at the Frascati Neutron Generator.[50, 51]

3.3.1 JET DT Campaign Benchmarks

A collaboration has been established with EURATOM-JET for the neutronics analysis of JET. In addition to validating the EU neutronics transport codes and data, the JET Project, through JET3 - DT Technology EUROfusion Consortium, would like to widen the scope of JET experiments by also including the validation of other newly developed transport codes/data used by the US. Specifically, the collaboration would seek to validate ADVANG, ATTILA, and DAG-MCNP, including the R2S methodology that relies upon it, by using the experimental data to be provided by JET.

Two different benchmark exercises were devised, one focusing on streaming of neutrons and another on the shutdown dose rate. The former is particularly valuable for the validation of advanced hybrid variance reduction schemes, such as ADVANTG. The other is valuable for validation of Shutdown Dose Rate (SDR) calculation methodologies, and particularly the

R2S methodology developed under this project.

Because the geometric description of the JET facility is already available in the native MCNP text files, participating in this effort requires first generating a CAD-based description. This was performed using the MCNP2CAD conversion tool developed in conjunction with the DAGMC toolkit. These CAD-based models are currently being used for preliminary analysis of these benchmarks.

3.4 Groupwise Transmutation CADIS (GT-CADIS)

The final capability that was supported, in part, by this project achieves an integration of the above-mentioned hybrid acceleration with the R2S SDR predictions. Although various implementations of the CADIS methodology allow for the acceleration of Monte Carlo-based simulations of the prompt photon dose, it is more challenging to accelerate the simulation of delayed photon dose. Whereas cross-sections exist and can be measured for the production of prompt neutrons, delayed neutrons arise from a combination of transmutation and decay that requires an activation calculation to predict the exact source. Under certain assumptions, however, it is possible generate effective cross-sections for the production of delayed photons. This allows for a multi-step approach to the CADIS methodology in which acceleration parameters for the neutron transport step can be determined such that they ultimately result in an optimum estimate of the delayed photon dose. The single neutron interaction, low burnup (SNILB) approximation was shown to be valid for nearly all magnetic fusion energy (MFE) environments, thus allowing the Groupwise Transmutation CADIS (GT-CADIS) methodology to be applied to such systems. Using this methodology, problem specific pseudo-cross-sections are evaluated for each material and each shutdown time of interest in a problem, and used to develop neutron transport acceleration parameters. This methodology was demonstrated on the ITER shutdown dose rate benchmark experiments as well as a characteristic fusion energy system.[52, 53]

3.5 Ongoing Developments

Three different improvements began in the final months of this project, all extensions of the GT-CADIS capability:

1. rigorous propagation of statistical error from the neutron transport to the delayed photon dose,
2. extension of the pseudo-cross-sections for time-integration to account for moving activated components and/or moving dose recipients, and
3. application of the generic multi-step CADIS capability to heat generation instead of activation.

Chapter 4

Nuclear Data for Fusion Applications

The nuclear data community is constantly seeking to improve the data that is available for nuclear analysis, particulary the neutron transport and transmutation cross-sections. This results in regular changes to the standard data sets for all nuclear applications, some of which are adopted by the standard data set that is targeted at fusion neutronics applications, the Fusion Evaluated Nuclear Data Library (FENDL). Under this project, we monitor the developments of these nuclear data sets and test any updates to understand the impact of those updates on the predictions of nuclear responses in fusion energy systems.

During this project, there was a new release of the fundamental US-based data set, ENDF/B-VII.0, some of which was incorporated in an update of the the international fusion nuclear data set, FENDL-2.1. Version 2.1 of FENDL includes 71 elements or isotopes. Data for most of the isotopes/elements (40) were taken from ENDF/B-VI.8. Following the release of ENDF/B-VII.0 we performed MCNP calculations for a 1-D calculational benchmark representative of an early ITER design that was utilized during the FENDL development process. Calculations were carried out using FENDL-2.1 with the data for the 40 isotopes/elements replaced by the recent data from ENDF/B- VII.0 and the results for flux, heating, dpa, and gas production were compared to those obtained using the FENDL-2.1 library.

The results of the ITER calculational benchmark clearly show that the previously observed differences in nuclear heating and radiation damage are removed when we use the recent correctly processed ENDF/B-VII.0 data. Differences in all nuclear parameters are very small implying that minimal impact is expected on ITER analysis and updating the FENDL-2.1 library is not urgently needed for ITER analysis.

However, a larger effect is expected when used in analysis of other fusion systems with breeding blankets (Demo and power plants). Calculations for an inertial fusion energy (IFE) power plant showed relatively large changes in nuclear parameters due to the large changes in the H-3 and Au-197 data that affect the energy spectrum of neutrons emitted from the IFE target. The results confirm the need for updating FENDL-2.1 for use in analysis of fusion systems beyond ITER.[54, 55, 56]

Similar benchmarking occurred with the release of FENDL-3, showing a modest increase of the neutron flux levels in the deep penetration regions and a substantial increase in the gas production in steel components. The comparison to experimental results showed good agreement with no substantial differences between FENDL-3.0 and FENDL-2.1 for most responses. There is a slight trend, however, for an increase of the fast neutron flux in the shielding experiment and a decrease in the breeder mock-up experiments. The photon flux spectra measured in the bulk shield and the tungsten experiments are significantly better reproduced with FENDL-3.0 data. In general, FENDL-3, as compared to FENDL-2.1, shows an improved performance for fusion neutronics applications. It is thus recommended to ITER to replace FENDL-2.1 as reference data library for neutronics calculation by FENDL-3.0.[57, 58]

Chapter 5

Neutronics Support for Fusion Materials

One of the ongoing tasks of this project was to provide *ad hoc* neutronics support to the fusion materials community. Fusion energy systems are an extremely challenging environment for materials, and their selection and design must consider many factors, some of which are related to nuclear analysis. In many cases, this support played a role in guiding other materials research without any specific outcomes or publications. In a number of cases, however, this project did produce standalone outcomes.

5.1 Nuclear Reponses in SiC Composites

One such effort focused on understanding the nuclear responses in SiC/SiC composite materials, in which the responses in the fiber, matrix, and interface may be important. Neutronics calculations have been performed to determine the radiation damage parameters in the fiber, matrix, and interface components of the SiC/SiC composite structural material employed in candidate breeding blankets. The radiation damage parameters were calculated for both the carbon and silicon sublattices. The breeder blanket concepts considered included LiPb/SiC, FLiBe/Be/SiC, and Li_2O/Be/He/SiC.

Nearly similar atomic displacement damage rates take place in Si and C. Helium production in C is about a factor of 4 larger than that in Si. On the other hand, significant hydrogen production occurs in Si with negligible amount in C. As a result, the burnup of Si is about a factor of 2 more than that of C. Property degradation depends on the kind of impurities introduced by transmutations. The transmutation products include Al, Mg, Li, and Be. The nonstoichiometric burnup of Si and C is expected to be worse than stoichiometric burnups and could be an important issue for SiC. Damage parameters were compared in candidate blanket concepts. We conclude that if the dpa is the lifetime driver for SiC/SiC structure, the lifetime will be significantly longer in FLiBe/Be/SiC or Li_2O/Be/He/SiC FW/B concepts than in a LiPb/SiC blanket operating at the same neutron wall loading. On the other hand, if gas production or burnup determine lifetime, the lifetime will be slightly longer in a LiPb/SiC blanket. In IFE systems, the geometrical, spectral, and temporal features influence the structure damage parameters. The pulsed nature of IFE systems results in very large instantaneous damage parameters with the dpa and gas production being affected differently. It is therefore essential to account for these unique features for accurate prediction of the structure lifetime in IFE systems.

The results given here provide an essential input for SiC/SiC composite lifetime assessment. The impact of damage parameters on properties and lifetime needs to be assessed. However, a determination of the effect of fusion neutron transmutations on the thermomechanical properties of SiC will be required to set a lifetime for SiC components. It is speculated that the lifetime will be set by swelling produced by transmuted helium.[59]

5.2 Assessing Applicability of Alternative Irradiation Environments for Fusion Materials Development

Given the lack of appropriate irradiation facilities that specifically represent a fusion energy environment, other irradiation facilities have been considered for supporting fusion materials development.

Irradiation tests for candidate fusion structural and plasma facing materials are usually performed in fission reactors such as the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL). To understand how to correlate and extrapolate results from such tests to the actual environment in fusion systems, we performed calculations to quantify the damage parameters in the leading structural and plasma-facing armor material candidates when used in MFE and IFE systems and when irradiated in HFIR. The structural materials considered are the ferritic steel alloy F82H, austenitic steel SS316, the vanadium alloy V4Cr4Ti and the SiC/SiC composite. Plasma-facing armor material candidates included Be, W, and carbon fiber composites (CFC). Atomic displacement damage and gas production rates are greatly influenced by the neutron energy spectrum. The results indicate that for the same neutron wall loading, atomic displacement damage is slightly lower in IFE systems than in MFE systems but gas production is about a factor of 2 lower due to the softened neutron spectrum. In addition, much lower gas production is obtained in samples irradiated in fission reactors. For the same atomic displacement damage level, gas production is significantly lower in fission irradiation compared to that in the first wall of a fusion blanket with the effect strongly dependent on the material. For SS316, the high helium production in B and Ni by low energy neutrons yields higher helium production following irradiation in fission reactors. The results of this work will help guide irradiation experiments in fission reactors to properly simulate the damage environment in fusion systems by possible gas implantation and will facilitate extrapolating to the expected material performance in fusion systems. In addition, the results represent a necessary input for modeling activities aimed at understanding the expected effects on mechanical and physical properties.[60, 61]

While similar analysis was desired for the irradiation environment of the Spallation Neutron Source (SNS) at ORNL, there is a lack of high quality data for the important nuclear responses, namely DPA, gas production and transmutation rates, to produce reliable estimates.

Chapter 6

Summary

With its origins in the detailed analysis of nuclear performance in fusion energy systems, this project evolved to incorporate the development of new tools to improve the predictive nature of that analysis. As the research priorities of the Office of Fusion Energy Sciences (OFES) shifted away from the design and analysis of nuclear systems, the ongoing benefit of this project has become increasingly based on those software developments, preparing the predictive capability for a return to fusion nuclear facility design in the future.

The APEX and ITER TBM programs included the analysis of a wide array of FW/B options for future fusion experimental facilities and power plants. This analysis resulted in valuable insight into the viable options for achieving tritium self-sufficiency and managing radioactive wastes, among other criteria.

New software tools were developed to model complex CAD-based geoemtries that are important in fusion energy systems. Since many of the components have a primary or secondary role as radiation shields, their geometries intentionally incorporate features to improve their shielding ability but also increasing the modeling complexity. The DAGMC toolkit allows the direct simulation of such geometries and subsequent improvements and extensions also allow hybrid variance reduction, SDR analysis, and the combination of both. This capability is garnering increased interest in other application areas in which complex geometries and/or radiation shielding dominate the application, including space travel, medical physics and accelerator systems.

As a consumer of nuclear data, we have been monitoring updates and improvements in the evaluated nuclear cross-section sets, and devising computational benchmarks to help understand the impact of those changes on design and analysis of fusion energy systems. Changes to FENDL-2.1 and FENDL3 resulted in modest but non-negligible changes in some nuclear parameters.

Support for the fusion materials community often results in contributions that are not easily tracked as a deliverable for this project. One notable exception was a comprehensive study of the radiation damage parameters, including DPA, gas production and transmutation, in SiC composite materials. Since this radiation-induced damage produces different consequences in different components of the composite (matrix vs fibers vs interface), it warranted a more careful study. We have also completed an assessment of the suitability of a fission irradiation environment, notably HFIR, as a substitute for fusion irradiation, but were unable to complete the same for SNS due to a lack of suitable nuclear data at higher energies.

Products of this Project

[1] M. E. SAWAN, H. Y. KHATER, and D. K. SZE, "Nuclear analysis for a flowing Li2O particulate blanket without structural first wall," *Fusion Engineering and Design*, **49**, 667 (2000).

[2] M. E. SAWAN and S. MALANG, "Neutronics features of a high power density first wall/blanket with lithium evaporation cooling," *Fusion Engineering and Design*, **55**, 45 (2001).

[3] R. F. MATTAS et al., "EVOLVE – an advanced first wall/blanket system," *Fusion Engineering and Design*, **49**, 613 (2000).

[4] M. A. ABDOU et al., "On the exploration of innovative concepts for fusion chamber technology," *Fusion Engineering and Design*, **54**, 181 (2001).

[5] I. N. SVIATOSLAVSKY, M. E. SAWAN, and S. MAJUMDAR, "Spiral Blanket: An Innovative Idea for a Solid Wall Blanket with Potential for High Neutron Wall Loading," *Fusion Science and Technology*, **44**, 59 (2003).

[6] I. N. SVIATOSLAVSKY et al., "Engineering description of a solid first wall blanket based on advanced ferritic steel and re-circulating Flibe," *Proc. 20th IEEE/NPSS Symposium onFusion Engineering, 2003.*, pages 118–121, 2003.

[7] I. N. SVIATOSLAVSKY et al., "Engineering and geometric aspects of the solid wall re-circulating fluid blanket based on advanced ferritic steel," *Fusion Engineering and Design*, **72**, 307 (2004).

[8] C. P. C. WONG et al., "Molten salt self-cooled solid first wall and blanket design based on advanced ferritic steel," *Fusion Engineering and Design*, **72**, 245 (2004).

[9] M. E. SAWAN, "Nuclear aspects of a solid wall blanket based on advanced ferritic steel and re-circulating Flibe," *Proc. 20th IEEE/NPSS Symposium onFusion Engineering, 2003.*, pages 122–125, 2003.

[10] M. Z. YOUSSEF, M. E. SAWAN, and D. K. SZE, "The breeding potential of flinabe and comparison to flibe in CLiFF high power density concept," *Fusion Engineering and Design*, **61**, 497 (2002).

[11] M. Z. YOUSSEF, M. E. SAWAN, and H. Y. KHATER, "Comparison of radwaste volume and hazard in liquid wall and conventional solid wall concepts," *Proc. Proceedings of the 19th IEEE/IPSS Symposium on Fusion Engineering. 19th SOFE (Cat. No.02CH37231)*, pages 168–171, 2002.

[12] M. Z. YOUSSEF and M. E. SAWAN, "Component lifetime comparison and waste volume in CLiFF Sn/Flibe and Sn/LiPb blankets," *Fusion Engineering and Design*, **63**, 277 (2002).

[13] M. Z. YOUSSEF and M. E. SAWAN, "Radwaste volume in lithium and Flibe thick liquid wall and comparison to conventional SW concepts," *Fusion Engineering and Design*, **63**, 263 (2002).

[14] M. E. SAWAN and D.-K. SZE, "Transmutation and Production Rates of Elements in Flibe and Flinabe with Impact on Chemistry Control," *Fusion Science and Technology*, **44**, 64 (2003).

[15] M. Z. YOUSSEF and M. E. SAWAN, "On the Strategy and Requirements for Neutronics Testing in ITER," *Fusion Science and Technology*, **47**, 1038 (2005).

[16] M. Z. YOUSSEF, M. E. SAWAN, and A. YING, "Nuclear analyses for two look-alike helium-cooled pebble bed test blanket sub-modules proposed by the US for testing in ITER," *Fusion Engineering and Design*, **81**, 1567 (2006).

[17] M. E. SAWAN and M. A. ABDOU, "Physics and technology conditions for attaining tritium self-sufficiency for the DT fuel cycle," *Fusion Engineering and Design*, **81**, 1131 (2006).

[18] M. E. SAWAN, S. MALANG, C. P. C. WONG, and M. Z. YOUSSEF, "Neutronics Assessment of Molten Salt Breeding Blanket Design Options," *Fusion Science and Technology*, **47**, 510 (2005).

[19] C. P. C. WONG et al., "Assessment of First Wall and Blanket Options with the Use of Liquid Breeder," *Fusion Science and Technology*, **47**, 502 (2005).

[20] M. ABDOU et al., "U.S. Plans and Strategy for ITER Blanket Testing," *Fusion Science and Technology*, **47**, 475 (2005).

[21] M. E. SAWAN and M. Z. YOUSSEF, "Three-dimensional neutronics assessment of dual coolant molten salt blankets with comparison to one-dimensional results," *Fusion Engineering and Design*, **81**, 505 (2006).

[22] M. Z. YOUSSEF and M. E. SAWAN, "Activation Analysis for Two Molten Salt Dual-Coolant Blanket Concepts for the U.S. Demo Reactor," *Fusion Science and Technology*, **47**, 518 (2005).

[23] C. P. C. WONG et al., "Overview of the US ITER Dual Coolant Lead Lithium (DCLL) Test Blanket Module Program," *Proc. 21st IEEE/NPS Symposium on Fusion Engineering SOFE 05*, pages 1–6, 2005.

[24] S. SMOLENTSEV et al., "Numerical analysis of MHD flow and heat transfer in a poloidal channel of the DCLL blanket with a SiCf/SiC flow channel insert," *Fusion Engineering and Design*, **81**, 549 (2006).

[25] C. P. C. WONG et al., "An overview of dual coolant Pb17Li breeder first wall and blanket concept development for the US ITER-TBM design," *Fusion Engineering and Design*, **81**, 461 (2006).

[26] A. YING et al., "An overview of US ITER test blanket module program," *Fusion Engineering and Design*, **81**, 433 (2006).

[27] C. P. C. WONG et al., "An overview of the US DCLL ITER-TBM program," *Fusion Engineering and Design*, **85**, 1129 (2010).

[28] C. P. C. WONG et al., "Progress on DCLL Blanket Concept," *Fusion Science and Technology*, **64**, 623 (2013).

[29] M. E. SAWAN and M. Z. YOUSSEF, "Neutronics Features of a Dual Coolant Lithium Lead Test Blanket Module for ITER," *Proc. 21st IEEE/NPS Symposium on Fusion Engineering SOFE 05*, pages 1–4, 2005.

[30] M. E. SAWAN, E. P. MARRIOTT, and M. DAGHER, "Neutronics performance parameters for the US dual coolant lead lithium ITER test blanket module," *Proc. 2009 23rd IEEE/NPSS Symposium on Fusion Engineering*, pages 1–4, 2009.

[31] M. E. SAWAN, B. SMITH, E. P. MARRIOTT, and P. P. H. WILSON, "Three-dimensional nuclear analysis for the US dual coolant lead lithium ITER test blanket module," *Fusion Engineering and Design*, **85**, 1027 (2010).

[32] M. Z. YOUSSEF and M. E. SAWAN, "Assessment of the Activation, Decay Heat, and Waste Disposal of a Dual Coolant Lithium Lead Test Blanket Module for ITER," *Proc. 21st IEEE/NPS Symposium on Fusion Engineering SOFE 05*, pages 1–5, 2005.

[33] C. P. C. WONG et al., "Fusion Nuclear Science Facility - Advanced Tokamak Option," *Fusion Science and Technology*, **60**, 449 (2011).

[34] C. P. C. WONG et al., "Fusion Nuclear Science Facility-AT: A Material and Component Testing Device," *Fusion Science and Technology*, **62**, 283 (2012).

[35] M. E. SAWAN et al., "Neutronics Analysis in Support of the Fusion Development Facility Design Evolution," *Fusion Science and Technology*, **60**, 671 (2011).

[36] P. P. H. WILSON et al., "Direct Accelerated Geometry Monte Carlo," http://svalinn.github.io/DAGMC/, 2017.

[37] P. P. H. WILSON, T. J. TAUTGES, J. A. KRAFTCHECK, B. M. SMITH, and D. L. HENDERSON, "Acceleration techniques for the direct use of CAD-based geometry in fusion neutronics analysis," *Fusion Engineering and Design*, **85**, 1759 (2010).

[38] P. SNOUFFER, *Validation and Verification of Direct Accelerated Geometry Monte Carlo*, MS Nuclear Engineering and Engineering Physics, University of Wisconsin-Madison, Madison, WI, United States, 2011.

[39] T. D. BOHM, S. T. JACKSON, M. E. SAWAN, and P. P. H. WILSON, "Benchmarking a CAD-Based Monte Carlo Code Using Fusion-Specific Experiments," *Nuclear Technology*, **175**, 264 (2011).

[40] M. E. SAWAN et al., "Application of CAD-neutronics coupling to geometrically complex fusion systems," *Proc. 2009 23rd IEEE/NPSS Symposium on Fusion Engineering*, pages 1–6, 2009.

[41] B. SMITH, T. TAUTGES, and P. WILSON, "Sealing Faceted Surfaces to Achieve Watertight CAD Models," *Proc. Proceedings of the 19th International Meshing Roundtable*, pages 177–194, Chattanooga, TN, United States, 2010, Springer.

[42] B. M. SMITH, *Robust Tracking and Advanced Geometry for Monte Carlo Radiation Transport*, PhD Nuclear Engineering and Engineering Physics, University of Wisconsin-Madison, Madison, WI, United States, 2011.

[43] D. MOULE and P. WILSON, "Mesh Generation methods for Deterministic Radiation Transport Codes," *Proc. Transactions of the American Nuclear Society*, volume 104, pages 407–408, 2009, keep me.

[44] D. MOULE, *Sampling Material Composition of CAD Geometries*, MS Nuclear Engineering and Engineering Physics, University of Wisconsin-Madison, Madison, WI, United States, 2011.

[45] A. M. IBRAHIM et al., "Assessment of fusion facility dose rate map using mesh adaptivity enhancements of hybrid Monte Carlo/deterministic techniques," *Fusion Engineering and Design*, **89**, 1875 (2014).

[46] E. BIONDO, A. M. IBRAHIM, S. W. MOSHER, and R. E. GROVE, "Accelerating Fusion Reactor Neutronics Modeling by Automatic Coupling of Hybrid Monte Carlo/Deterministic Transport on CAD Geometry," *Proc. Mathematics & Computations (M&C+SNA+MC 2015)*, Nashville, Tennessee, 2015.

[47] A. IBRAHIM, *Automatic Mesh Adaptivity for Hybrid Monte Carlo/Deterministic Neutronics Modeling of Difficult Shielding Problems*, PhD Nuclear Engineering and Engineering Physics, University of Wisconsin-Madison, Madison, WI, United States, 2012.

[48] E. RELSON, P. WILSON, and E. BIONDO, "Improved Mesh Based Photon Sampling Techniques For Neutron Activation Analysis," *Proc. M&C 2013*, Sun Valley, ID, 2013, American Nuclear Society.

[49] E. RELSON, *Improved Methods For Sampling Mesh-Based Volumetric Sources In Monte Carlo Transport*, MS Nuclear Engineering and Engineering Physics, University of Wisconsin-Madison, Madison, WI, United States, 2013.

[50] E. BIONDO, A. DAVIS, A. SCOPATZ, and P. P. WILSON, "Rigorous Two-Step Activation for Fusion Systems with PyNE," *Transactions of the American Nuclear Society*, **112**, 617 (2015).

[51] E. D. BIONDO, A. DAVIS, and P. P. WILSON, "Shutdown Dose Rate Analysis with CAD Geometry, Cartesian/Tetrahedral Mesh, and Advanced Variance Reduction," *Fusion Engineering and Design*, **106**, 77 (2016).

[52] E. D. BIONDO and P. P. WILSON, "Transmutation Approximations for the Application of Hybrid Monte Carlo/Deterministic Neutron Transport to Shutdown Dose Rate Analysis," *Nuclear Science and Engineering*, **187**, 27 (2017).

[53] E. D. BIONDO, *Hybrid Monte Carlo/Deterministic Neutron Transport for Shutdown Dose Rate Analysis*, PhD Nuclear Engineering and Engineering Physics, University of Wisconsin, Madison, Madison, WI, United States, 2016.

[54] M. E. SAWAN, "Impact of ENDF/B-VII.0 Release on Fusion Evaluated Nuclear Data Library FENDL-2.1," *Fusion Science and Technology*, **56**, 766 (2009).

[55] M. E. SAWAN, "Benchmarking FENDL-2.1 with impact of ENDF/B-VII.0 release on nuclear heating and damage," *Proc. 2009 23rd IEEE/NPSS Symposium on Fusion Engineering*, pages 1–4, 2009.

[56] M. E. SAWAN and T. D. BOHM, "Impact of FENDL-2.1 Updates on Nuclear Analysis of ITER and Other Fusion Systems," *Journal of the Korean Physical Society*, **59**, 1158 (2011).

[57] U. FISCHER et al., "Benchmarking of the FENDL-3 Neutron Cross-section Data Starter Library for Fusion Applications," *Nuclear Data Sheets*, **120**, 230 (2014).

[58] T. D. BOHM and M. E. SAWAN, "The Impact of Updated Cross Section Libraries on ITER Neutronics Calculations," *Fusion Science and Technology*, **68**, 331 (2015).

[59] M. E. SAWAN, L. SNEAD, and S. ZINKLE, "Radiation Damage Parameters for SiC/SiC Composite Structure in Fusion Nuclear Environment," *Fusion Science and Technology*, **44**, 150 (2003).

[60] M. E. SAWAN, "Damage parameters of structural materials in fusion environment compared to fission reactor irradiation," *Fusion Engineering and Design*, **87**, 551 (2012).

[61] M. E. SAWAN, "Transmutation of Tungsten in Fusion and Fission Nuclear Environments," *Fusion Science and Technology*, **66**, 272 (2014).

[62] R. E. NYGREN et al., "A fusion chamber design with a liquid first wall and divertor," *Proc. 20th IEEE/NPSS Symposium onFusion Engineering, 2003.*, pages 288–295, 2003.

[63] B. J. MERRILL et al., "Safety assessment of two advanced ferritic steel molten salt blanket design concepts," *Fusion Engineering and Design*, **72**, 277 (2004).

[64] A. YING et al., "Progress on an integrated multi-physics simulation predictive capability for plasma chamber nuclear components," *Fusion Engineering and Design*, **85**, 1681 (2010).

References to other Work

[65] T. J. TAUTGES, R. MEYERS, K. MERKLEY, C. STIMPSON, and C. ERNST, "MOAB: A Mesh-Oriented Database," SAND2004-1592, Sandia National Laboratories (2004), Report.

[66] T. J. TAUTGES, "CGM: A geometry interface for mesh generation, analysis and other applications," *Engineering with Computers*, **17**, 299 (2001), Journal Article.

[67] P. WILSON et al., "State-of-the-art 3-D radiation transport methods for fusion energy systems," *Fusion Engineering and Design*, **83**, 824 (2008).

[68] T. D. BOHM et al., "Detailed 3-D nuclear analysis of ITER blanket modules," *Fusion Engineering and Design*, **89**, 1954 (2014).

[69] T. BOHM et al., "Detailed 3-D nuclear analysis of ITER outboard blanket modules," *Fusion Engineering and Design*, **9697**, 222 (2015).

[70] T. D. BOHM, M. E. SAWAN, S. T. JACKSON, and P. P. H. WILSON, "Detailed nuclear analysis of ITER ELM coils," *Fusion Engineering and Design*, **87**, 657 (2012).

[71] A. HAGHIGHAT and J. C. WAGNER, "Monte Carlo Variance Reduction with Deterministic Importance Functions," *Progress in Nuclear Energy*, **42**, 25 (2003).

[72] A. M. IBRAHIM, E. POLUNOVSKIY, M. J. LOUGHLIN, R. E. GROVE, and M. E. SAWAN, "Acceleration of calculation of nuclear heating distributions in ITER toroidal field coils using hybrid Monte Carlo/deterministic techniques," *Fusion Engineering and Design*, **109-111, Part A**, 255 (2016).