DOE/ER/54462-1

Fusion Advanced Design Studies
Project #DE-FG02-98ER54462

Final Report for the Period
January 1, 2013 to December 31, 2015

L.A. El-Guebaly, D.L. Henderson, P.P. Wilson, and J. Blanchard

Fusion Technology Institute
University of Wisconsin – Madison

March 24, 2017

1. Introduction

During the January 1, 2013 – December 31, 2015 contract period, the UW Fusion Technology Institute personnel have actively participated in the ARIES-ACT and FESS-FNSF projects, led the nuclear and thermostructural tasks, attended several project meetings, and participated in all conference calls. The main areas of effort and technical achievements include:

- Updating and documenting the nuclear analysis for ARIES-ACT-1. This includes:
    - Adjusting the blanket and structuring ring (SR) to catch 99% of the nuclear heating, updating the overall TBR, redefining the radial/vertical builds for 45 cm thick inboard blanket, 30 cm thick SR, 10 cm thick vacuum vessel (VV), and WC-free shield
    - Updating the nuclear heat loads to all components and overall energy multiplication
    - Estimating the lifetimes of SR and VV based on radiation damage, taking into account the peaking due to neutrons streaming through assembly gaps
    - Evaluating the temperature response of all components during LOCA/LOFA events.
- Performing nuclear analysis for ARIES-ACT-2 with R=9.75 m. This includes the poloidal distribution of neutron wall loading, radial/vertical build definition, 3-D TBR assessment, 3-D nuclear heat loads to all components, 3-D radiation damage, and estimate of component lifetimes based on radiation damage
- Performing thermostructural analysis of the ARIES divertor concepts, including transients (ELMs). Analyses were carried out parametrically, due to the uncertain nature of the ELM loads for an advanced power plant. Both “plate” and “finger” designs were considered.
- Disruption analyses were carried out to assess the impact of a current quench on the structural integrity of the ARIES vacuum vessel. A 30 ms linear quench was applied to the plasma current and the resulting currents and forces in the vessel were calculated. An assessment of the effect of these forces on this structure is imminent
- Defining nuclear constraints on power plant studies, their origin, and issues foreseen for FNSF
- Developing blanket testing strategy for FNSF with specific goals and requirements
- Defining FNSF testing environment and timeline to qualify DCLL blanket for DEMO via a series of testing in Test Blanket Modules (TBM)
- Proposing Materials Test Module (MTM) to test existing and more advanced materials in FNSF and develop comprehensive multi-materials database for up to 90 dpa with possibility of extending tests to 125 dpa
- Performing 3-D nuclear analysis for ARIES-ACT-2 to provide additional nuclear parameters needed for the FNSF timeline table. These include the peak radiation damage at the structural ring and VV, and divertor, and peak fast neutron fluence and nuclear heating at the magnet
- Working with Ghoniem (UCLA) to build a multiscale model for failure prediction in plasma-facing components. The model will predict global deformation using a finite element code and then pass the results, locally, to an atomistic model for failure prediction.
- Documenting a White Paper on “Challenges of Tritium Self-Sufficiency Facing FESS-FNSF”
- Optimizing the compositions of the FNSF shield, vacuum vessel, and structural ring for the inboard, top/bottom, and outboard regions
- Defining the initial radial and vertical builds, and identifying foreseen shielding, breeding, activation, and streaming issues for FNSF
- Performing activation analysis for all components to classify the radwaste and examine the options of recycling and clearance
- Providing a list of systems to monitor the parameters as the FNSF is brought up from non-nuclear (He/H), to lightly nuclear (D-D), and then to fully nuclear (D-T)
- Performing transient (ELM and disruption) analyses of first wall behavior, as companion to similar analyses carried out for the divertor
- Refining divertor models to further explore the limits for heat loads expected on the divertor of an FNSF

- Developing electromagnetic models to study the effect of disruptions on all components inside the FNSF vacuum vessel
- Developing the initial CAD model for FNSF with inputs from various team members
- Preparing 21 presentations by El-Guebaly, Blanchard, and Marriott given at the ARIES-ACT and FNSF project meetings
- Eight ARIES-related papers published in Fusion Science and Technology journal, Progress in Nuclear Science and Technology journal, and ANS Transactions of ANS 2013 winter meeting.
- Responding to reviewers' comments, and submitting two FNSF-related papers to the Fusion Science and Technology journal based on talks presented at the 21st Topical Meeting on the Technology of Fusion Energy
- Four ARIES and FNSF-related abstracts submitted to the 21th Topical Meeting on the Technology of Fusion Energy and will be published in Fusion Science and Technology journal.
- Documenting six FNSF-related UW-FDM reports on neutronics, shielding, and activation analyses
- Participating in all conference calls and exchanging numerous E-mails with members of the ARIES and FNSF teams and fusion community.

2. Summary of Tasks Accomplished

2.1 ARIES-ACT-2 Nuclear Assessments

El-Guebaly and Mynsberge applied the newly developed UW DAGMC code to the DCLL blanket to accurately estimate the TBR for the final design with R=9.75 m. All blanket features were incorporated in the 3-D model to account for the details of the blanket. To distinguish the impact of the individual design elements on TBR, a stepwise approach was applied as well. It involves building the CAD model from scratch, and, in multiple steps, adds the internals/externals of the blanket: first, side, and back walls, cooling channels, SiC flow channel inserts, stabilizing shells, penetrations, and assembly gaps, along with varying the Li enrichment. At each step, the impact of the design element on TBR was recorded. With 0.65 m inboard blanket and 1 m outboard blanket, the final ARIES-ACT-2 design satisfies the breeding requirement (overall TBR of 1.05) with 40% Li-6 enrichment.

The 3-D model was used to evaluate the nuclear heating and overall energy multiplication using LiPb with 40% ^{6}Li enrichment. This heating is needed to obtain details on the thermal hydraulic analysis and eventually the thermo-mechanical stresses the device is subjected to during operation. The total recoverable nuclear heating amounts to 2458 MW, meaning an overall energy multiplication of 1.165. The split of the nuclear heating between the He and LiPb high temperature coolants is 49:51. About 50 MW (1.6% of the total thermal power) is deposited in the VV and LT shield (mainly in the IB side), which will be dumped as low-grade heat.

The FW and divertor 3-D geometries were used to create the 3-D model for the neutron wall loading (NWL) profile. The NWL peaks at the OB midplane of ARIES-ACT-2 at 2.2 MW/m^2. The corresponding peak radiation damage reaches 23 dpa/FPY. Based on the 200 dpa limit for advanced ferritic steel (FS), the blanket and divertor should be replaced every 8 FPY.

2.2 ARIES-ACT-2 Radial/Vertical Builds

El-Guebaly takes prime responsibility in defining, issuing, and updating the radial and vertical builds that meet the breeding requirement, capture most of the nuclear heating, and protect the superconducting magnets. The inboard, outboard, and divertor builds include the size and optimal composition of all components. These radial builds are a precursor to the physics and systems codes and CAD drawings. She also updates the scaling laws that govern the variation of shield thickness with neutron wall loading. This is an essential element for the parametric study conducted by the systems code.

2.3 ARIES-ACT-1 3-D Streaming Analysis

El-Guebaly and Mynsberge have been concerned about neutron streaming through the assembly gaps. They are nominally 2 cm wide at room temperature on a new set of sectors. As the power core heats up, these gaps partially close. They examined the radiation damage for a range of gap widths from 0 to 2 cm. The analysis determined the radiation damage peaking behind the gap, the dpa and He production in the Structural Ring and Vacuum Vessel materials, as well as their estimated lifetime (related to a 200 dpa limit). The IB Structural Ring should be replaced once during the plant's lifetime and would not be reweldable (given the present material properties of 1 He appm). On the other hand, the entire VV should survive as a lifetime component (dpa << 200) although it is not reweldable at the inboard even in the absence of a gap. The He/dpa ratio for IB and OB FS structure were also evaluated. This ratio is of great importance to the materials community to access the relevance of irradiating FS samples in fission facilities. Our results indicate a range for the He/dpa ratio of 0.3-2 for the SR and 0.1-8 for the VV. For comparison, this ratio could reach 10 for a FS FW attached to a LiPb blanket.

2.4 ARIES-ACT-1 3-D Activation Analysis

Past ARIES activation analyses relied on a 1-D method to address the waste management issues and evaluate the decay heat for safety analysis. The effect of heterogeneity on the system can now be closely examined, for the first time in ARIES history, using the more accurate 3-D predictions by the newly modified ALARA activation code. Mynsberge and El-Guebaly performed 3-D activation analysis for ARIES-ACT-1 FW and VV and compared the 1-D and 3-D results for specific activity, decay heat, waste disposal rating, recycling dose, and clearance index. Generally speaking, the 1-D analysis seems conservative as it overestimates the activation responses by 20-30%.

2.5 ARIES-ACT-1 2-D LOCA/LOFA Analysis

Martin investigated several power core design variations to determine the inboard hot spot temperatures during a loss-of-coolant accident. He assumed a complete loss of both water and helium coolants in the power core as well as loss of flow of the LiPb power core coolant. Additionally, the plasma remains on for a full three seconds after the onset of the LOCA/LOFA event. To help radiate the heat from the inboard components, he has assumed an emissivity of 0.9 on all facing surfaces. The 2-D analysis assumed the time history of the 1-D heat decay for the power core components as provided by El-Guebaly. Changing the inboard shield filler from WC to borated-FS reduced the maximum inboard temperature by 46ºC. The inboard maximum temperatures for the ODS-FS Structural Ring, VV, and LT shield approach 900ºC, but the outboard FS-based components remain below 700ºC. Also, the ODS-FS structure of the divertor closely approaches 900ºC. The only feasible solution for the inboard and divertor temperature excursions is to use advanced nano-structured FS that can survive such accidents if the temperature during LOCA/LOFA remains below 1050ºC.

2.6 Structural and Disruption Analyses

Blanchard and Martin carried out a series of studies for the temperatures in the finger and plate divertor concepts under steady state and transient (ELM) conditions. Maximum temperatures were calculated for a variety of combinations of steady and transient heat fluxes in order to permit future design work as the physics of ELMs becomes more certain. Three-dimensional finite element models were used for the analysis. The finger concepts were found to have slightly better performance than the plate when only thermal effects were considered. It is assumed that the difference would be greater if stresses were considered as well.

Three-dimensional finite element models were also used to study the effects of a current quench on the structural integrity of the ARIES vacuum vessel. The UCSD vacuum vessel model was used as a starting point, but Blanchard and Martin added additional elements to model the plasma current, kink shells, structural ring, etc. The plasma current was assumed to ramp down linearly in 30 ms. Currents induced in all surrounding structures were calculated along with the resulting forces (due to the poloidal and toroidal magnetic fields). Finally, the equivalent pressure distribution on the vessel wall was determined and delivered to UCSD for incorporation into their structural analyses. Preliminary results indicate that the loads on the vessel can be handled with the current design.

2.7 FNSF Nuclear Task

El-Guebaly defined the nuclear environment that drives the purpose of the FNSF study, the nuclear constraints imposed on the ultimate power plant designs, and the differences that impact the FNSF design. She also defined the pre-FNSF tritium breeding-related R&D program and reviewed the radiation damage limits for structural ferritic steel, tungsten alloys, and magnets. At Kessel's request, El-Guebaly reviewed previous neutronics analyses performed by UW for FNSF-like designs such as the ST-FNSF by PPPL, the FDF by GA and the Pilot Plant by PPPL. In addition, she actively participated in the group discussion that defined the mission, goal and timeline of the FNSF study.

In collaboration with S. Malang and L. Waganer, El-Guebaly developed a staged blanket testing strategy tailored to meet the FNSF timeline in order to test, develop, and qualify the DCLL blanket for US DEMO and power plants. This strategy reaches beyond the traditional testing mission of ITER where four generations of DCLL blanket concept are tested first in TBMs (with limited dimensions), and then converted (assuming +ve results) into full sector for qualification before use in DEMO and power plants. During operation, the TBM serves as "forerunner" and develops more advanced blanket technologies for GEN-II, III, and IV DCLL blanket systems. The combined results from TBMs and blanket systems are essential to build high confidence and lower risk for successful operation of advanced blankets in DEMO and power plants.

At the June 2014 project meeting, El-Guebaly proposed including a materials testing module (MTM) in FNSF that can test new generations of materials for more advanced DEMO and power plants (90 – 125 dpa). This materials testing is equally important as testing blankets in TBMs and may turn out to be best option for 14 MeV neutron source! The main goal is to develop comprehensive multi-materials database for whole list of new materials for blankets, divertors, magnets, insulators, etc. The most important attribute would be the much larger specimen volumes compared to the 10-500 ml range available in the SNS/IFMIF/FAFNIR series of neutron sources.

Based on her extensive experience in power plant designs and at Kessel's request, El-Guebaly provided the following comprehensive list of materials-related areas where the materials community could help perform work on the FNSF project:

Ferritic steels:

- dpa limit for advanced ferritic steel such as ODS (NS)
- Need compositions (including ALL impurities and alloy density) for: GEN-I RAFM, GEN-II RAFM, ODS (NS)
- Develop Bainitic ferritic steel (for VV) that does not require PWHT. Need composition (including density and list of ALL impurities)
- Reusability temperature limit for advanced steels (such as Nano Structured Ferritic Alloys) after severe LOCA/LOFA accidents. 1000 oC or more?
- Reweldability limit for ferritic steel. It is 1 helium appm for austenitic steel. Could ferritic steel be rewardable at higher He content?
- Continue developing low-activation materials that decay rapidly to allow recycling all materials after short cooling period
- Cost of controlling impurities in advanced steels (to very low levels, e.g., Nb < 0.5 wppm and Mo < 5 wppm) to avoid generating high-level waste.

W alloys:

- W alloy for divertor and stabilizing shells: W-1.1TiC, W-La2O3, VMW, or W/W composites?
- Composition and list of impurities for preferred W alloy.
- Lifetime limiting criteria for W structure? dpa limit?
- Need to develop design rules and codes for brittle materials (such as W)?

Copper:

- Lifetime limiting criteria for Cu of normal magnets?
- Does Cu become brittle at 0.1 dpa?
- Should Cu magnet operate at high temperature (200-300 oC) to anneal out damage?
- Need to develop design rules and codes for brittle materials (such as Cu)?

SiC FCI and SiC/SiC composites:

- Life limiting criteria for SiC/SiC composites?
- Life limiting criteria for SiC FCI?
- Change of FCI electric conductivity and thermal conductivity under irradiation
- List of impurities for SiC.

LiPb breeder:

- List of impurities for LiPb
- Process and cost of adjusting the Li-6 enrichment online from natural to 90%
- Process and cost of recycling LiPb and filtering out Bi and Po byproducts online.

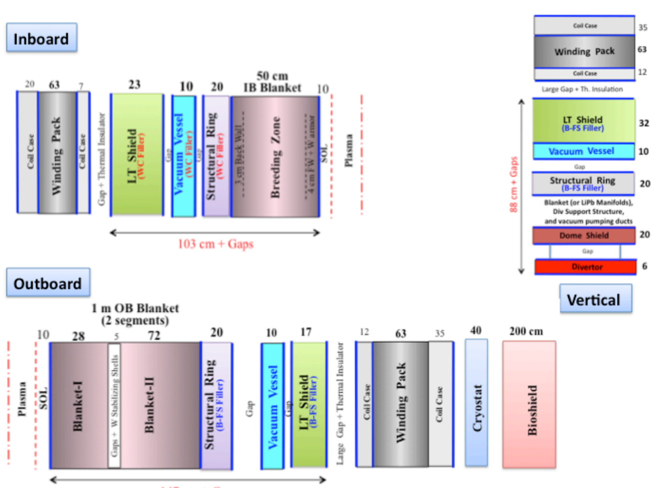
Materials Testing Module (1 m x 1 m) in FNSF (He/dpa=10):

- Testing of existing and new materials in MTM (e.g., Advanced ferritic steels, W alloys, ceramic breeders, beryllium, LT superconductors, HT superconductors, insulators, etc.). Preferred list of materials?
- Large specimen volume available. Minimum dimensions (length, width, depth) of each specimen?
- Range of specimen geometry. Preferred geometry for each material?

To provide the nuclear parameters listed in the "Missions and Metrics" table that Kessel put together for FNSF, El-Guebaly, Davis, and D'Angelo performed 3-D neutronics analysis for the ARIES-ACT-2 design using the neutron source distribution generated recently by Kessel on R-Z grid. These parameters include the 3-D radiation damage data and colored mapping of dpa, He, and heating for the structural elements of ARIES-ACT-2 inboard and outboard components. Such database presents important resources for the materials community in order to examine the helium to dpa ratio for specific components (such as the FW, structural ring, vacuum vessel, and shield) and determine the suitability of irradiating the various structural materials in fission reactors.

El-Guebaly takes prime responsibility in defining, issuing, and updating the radial and vertical builds that include the size and optimal composition of all components. These builds are a precursor to the physics code, systems code, and CAD drawings. It involves preliminary 1-D estimates for the TBR and radiation damage to structural components and magnets in absence of penetrations. Such estimates should be confirmed in FY-16 with detailed 3-D analyses that take into consideration the effect of neutron streaming through several large penetrations needed for plasma heating and current drive. The radial/vertical builds are shown in the figure below.

To support the UCLA multiscale model of the inboard DCLL blanket, El-Guebaly provided the radial distribution of nuclear heating for three individual elements of the DCLL blanket: LiPb breeder, RAFM steel structure, and SiC flow channel insert.



The activation assessments included two main tasks:

1. Radioactive inventory, decay heat, and radwaste classification and management options for IB, OB, and divertor regions. The high decay heat of the WC filler in IB components raises safety concerns and requires a specific analysis to check the temperature response during LOCA accidents. All FNSF components are recyclable shortly after shutdown. Most components qualify as Class A low-level waste – the least hazardous type of waste based on the NRC classification. The final results are documented in two UW reports:
    - B. Madani and L. El-Guebaly, "Shielding and Activation Analyses for Inboard Region of FESS-FNSF Design," University of Wisconsin Fusion Technology Institute Report, UWFDM-1423 (November 2015). Available at: http://fti.neep.wisc.edu/pdf/fdm1423.pdf.
    - M.T. Elias and L. El-Guebaly, "Shielding and Activation Analyses for Outboard Region of FESS-FNSF Design," University of Wisconsin Fusion Technology Institute Report, UWFDM-1424 (November 2015). Available at: http://fti.neep.wisc.edu/pdf/fdm1424.pdf
2. Activation of several advanced alloys that offer potential solutions for the corrosion problem facing the DCLL blanket. We assessed the activation implications of adding 5 wt% Al (and other Zr and Hf additives) to the ODS structure and compared their activation characteristics to that of F82H – the first generation of reduced-activation ferritic/martensitic (RAFM) steel that limits the blanket operating temperature to less than 550°C. Our results showed that all candidate corrosion-resistant ODS alloys qualify as low-level waste at the end of FNSF operation. These results were presented at the US/JA workshop held in October 2015 in Denver, CO:

    L. El-Guebaly, S. Malang, A. Rowcliffe, and L. Waganer, "FNSF Nuclear Analysis and Blanket and Materials Testing Strategy," presented at US/JA Workshop, Oct. 28-30, 2015, Denver, CO. Available at: https://sites.google.com/a/pppl.gov/fusion-energy-system-studies-fess/u-s-ja-workshop-october-2015.

2.8 FNSF Structural Task

Blanchard proposed an approach for addressing the effects of ELMS and disruptions on the FNSF first wall. Similar calculations have been carried out for divertors, but none have been done for the first wall. Blanchard found the primary ELM loads on the first wall to be as follows: frequency ranges from 4-20 Hz, an event duration of approximately 1.3 ms, and peak heat fluxes of approximately 360 MW/m^2. These ELM fluxes are not spatially uniform. For these cases, the effect on the first wall has found to be relatively minor.

For disruptions, the loads are found to be associated with an energy deposition of 345-621 MJ in a single disruption, an effective wall area of 198 m^2, and a time constant of approximately 8 ms. This gives a heat flux of at least 218 MW/m^2. These effects are more severe than for ELMS and the quantification of the effect is still under way.

Blanchard is also building a system-level model of a shield module. This will permit comprehensive treatment of spatial variations in surface heating and coolant pressure (due to frictional pressure drop), and the effects of global constraints into our thermomechanical models.

Blanchard carried out a series of simulations for the effect of ELM and disruption loading on an FNSF first wall. It was found that the first wall could likely withstand the expected ELM loads in such a device, based on the heat load models provided by Kessel. Disruptions will certainly cause melting in a steel first wall, but a thin tungsten coating is sufficient to prevent melting. Surface cracking would be expected.

A series of parametric studies were carried out for the FNSF divertor. ELM loading is still considered a severe challenge for the divertor, assuming a detached, slot-type configuration. More advanced divertor configurations or control of the ELMS themselves will likely be needed to create a credible FNSF design.

Model development has begun for the electromagnetic analysis of the FNSF components. The approach is following previous work done for ARIES. That is, we will consider stationary plasma that loses current linearly over a prescribed time period. The finite element code ANSYS was chosen for the simulations and symmetry has been invoked to reduce the size of the models. The attachment of the stabilizing shells has been identified as a key issue, as they must support a substantial load resulting from strong toroidal currents.

In parallel, Blanchard and Ghoniem (UCLA) are developing a multi-scale model for thermomechanical behavior that will use finite element methods to assess global deformations and to couple microscopic and mesoscopic material models with macroscopic models for stress and strain. These latter outcomes are then used for failure analysis. This will permit us to address numerous issues not previously considered, including grain boundaries, microstructural evolution, initial damage state, etc. This will be a significant advance over existing methodologies and will be a substantial contribution to the fusion effort. The key challenge identified so far is the complexity introduced as we incorporate appropriate turbulence models in a gas-cooled design modeled with high fidelity. A second challenge is the coupling of the mesoscopic and macroscopic models in a self-consistent, computationally efficient manner. Progress is being made on both of these issues.

2.9 CAD Activity

Marriott initiated the CAD activity for FNSF. The initial work focused on the general layout of all components as defined by the radial/vertical builds. Of importance is the location of the OB legs of the TF coils that allows horizontal maintenance. Leaving sufficient lateral space (17 cm) for clearance, maintenance port walls, and thermal insulation, the minimum radius of the OB leg is at 10.43 m from the machine centerline. Besides the radial/vertical components, the initial CAD drawing, shown in the figure below, displays the three divertor plates, He manifolds for blankets and locations of Kink and vertical stabilizing shells. This CAD activity required active interactions with all team members. Three presentations were given at conference calls and at the October 2015 project meeting in Denver.