Memo: ARIES-22

Date: 14 April 2005
Subject: ARIES Project Meeting Minutes, 24-25 February 2005, General Atomics
To: ARIES Team
From: L. Waganer

Organization	**ARIES Compact Stellarator**
ANL	
Boeing	Waganer
DOE	
FZK	Ihli
General Atomics	Lao, Politzer, Turnbull, Wong
Georgia Tech	Abdel-Khalik
INL	Merrill
LLNL	Jayakumar
MIT	
NYU	
ORNL	Lyon
PPPL	Ku, Solomon
RPI	McGuinness
UCSD	Grossman, Mau, Malang, Najmabadi, Raffray, Rudakov, Tillack, Wang
U of Wisc	El-Guebaly, Martin (Carl)

Ref: Agenda and Links to Presentations: http://aries.ucsd.edu/ARIES/MEETINGS/0502/

Administrative

Welcome – Alan Turnbull welcomed the team to General Atomics at LaJolla and provided logistical information around the meeting location. Our thanks to Alan for hosting the meeting.

Status of ARIES Program – Farrokh Najmabadi understands the FY06 Presidential budget is roughly the same as the present year, with no cuts to the Systems Studies budget. He has been working with FESAC to enable the ARIES team to provide DOE program scenarios and R&D planning recommendations and emphasis. It is widely recognized that ARIES provides a valuable service to the national and international fusion community.

The next ARIES meeting is scheduled to be held at the University of Wisconsin – Madison on Tuesday 14 June (all day) and a half day, Wednesday, 15 June 2005. At that meeting, we need to have near closure on our design approach and technical analyses. Farrokh emphasized that the ARIES team must be technically ready (final design approach and supporting analyses) for the September Town Meeting to be held at PPPL. It will be an in-depth, 3-day assessment of the ARIES Compact Stellarator approach. A wide spectrum of stellarator advocates and knowledgeable experts will be invited to attend in person and electronically via a web-based meeting. We need to have a consistent engineering and physics design approach to be evaluated at the Town Meeting. In addition to the baseline approach, the design space should be determined sufficiently to enable the selection of the optimal design point selection. This implies that the project must have near final plasma/coil configurations, design approaches, and system study parameter assessments to justify the chosen design parameter selection.

The ISFNT conference will be held in May. Abstracts were due in July 04 and papers in January 05. Please send papers to Rene Raffray for approval and project files.

The ARIES Web site has been updated with many archived files and presentations being added.

Compact Stellarator Reactor Integrated Systems Assessment

Status of Systems Code Studies – Jim Lyon summarized the Systems Code changes since the last meeting, namely with added transport scaling laws, coil structure engineering details, two new coil configurations, LiPb blanket configuration with SiC inserts, power core geometry constraints, and increased output data listings. He discussed the reactor optimization code changes and the planned updates.

Six coil configurations are now incorporated, including four NCSX type 3-FP and two MHH2 2-FP configurations. He highlighted the two newest NCSX configuration results. He described how the B_{max}/B_{axis} depends on the coil cross-section. He more closely examined the B_{max}/B_{axis} variation on the NCSX coils with square coil packs. He also showed the results for LiPb/FS blanket approaches, with SiC inserts.

The new output data listing were shown and the group helped critique the credibility of the economic results. The COE data were compared to the results from prior tokamak and stellarator studies. All the reactor cost accounts were examined. Several results did not appear to be correct, such as reactor plant equipment, special materials, blanket/shield, coils, and vacuum systems. The cost and weight of the coils seemed to have the largest errors; therefore Jim will confer again with Leslie Bromberg to resolve those discrepancies. Laila will also evaluate the cost of the replacement blanket components. The coil bucking structure seems to be missing.

Jim showed a variation of the reactor parameters with respect to the B_{max}; however the results seemed to be counter-intuitive. Jim will look into these data results. A similar backwards relationship was observed for beta variations. Jim thought the incorrect coil weights might incorrectly influencing the relationship.

Jim showed geometry relationships for peaked plasma temperatures and plasma impurity levels. He also inferred larger H-ISS95 values are required to offset the higher alpha particle losses.

The COE reduction effect afforded by larger power plant sizes was shown by Jim. A 2-GWe power plant has a COE of 45.3 mills/kWh as compared to a COE of 58.2 mills/kWh for the 1-GWe plant, a 23% reduction in the COE. This is a typical COE sizing relationship, but for comparative purposes, fusion plants are usually reported at 1 GWe, net.

Jim concluded his presentation with a list of questions to be answered before the September meeting, especially a more thorough treatment of the port maintenance system, including the removable shield elements.

Compact Stellarator Reactor Physics Basis

Attractive 2-FP and 3-FP Plasma and Coil Configurations – Recent Configuration Development Results – Long Poe Ku first discussed two newly analyzed A~2.5, 2-field period configurations of the MHH2 class being defined by Paul Garabedian. These 2-field period configurations are geometrically simpler than the 3-field configuration with more coils. For the MHH2-1104 configuration at a 5% beta conditions, the main magnetic modes are well suppressed and the principal mirror component does not harm the alpha confinement. Correspondingly, the effective ripple for this case with finite beta is significantly lower for regions of r/a > 0.5. The rotational transform produces high quality flux surfaces. The prescribed rotational transform does need externally driven currents. Reasonable coil designs have been defined with smooth contours and winding surfaces. Eight coils are required per period with four types of coils. There area a few interior areas that have tight spacing between coils. Both full and half period boundaries are compatible with sector maintenance approaches. Long Poe Ku feels further optimization of the coils is needed to regain the good confinement of the alpha particles.

The MHH2-K14 case of the same ultra-low aspect ratio configuration family has a rising rotational transform profile with bootstrap currents, but the main modes are higher. The alpha loss is less than 10%. The ripple is roughly twice as high as the other case. There are a few areas of instability (internal modes at beta of 4% and external modes for beta >5%). The flux surfaces are degraded by numerous field islands. Preliminary coil configurations resulted in coils with significant “kinkiness” that must be smoothed.

The embedded figure summarizes the geometric properties of the two MHH2 designs. The coil designs will be improved before the next meeting.

MHH2-1104

No. of Coils: 8/period

Different Types of Coils: 4

R/Δ_{min} (coil-plasma)=5.60

R/Δ_{min} (coil-coil)=17.9

I /R-B (max)=0.312 MA/m-T

coil lengths/R = 5.91, 5.63, 5.35, 5.08

B(max)/B(0) = 3.56 for 0.4 m by 0.4 m square conductors.

MHH2-K14_V

No. of Coils: 8/period

Different Types of Coils: 4

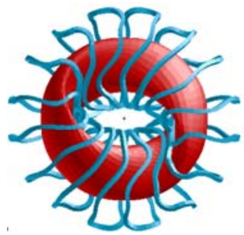
R/Δ_{min} (coil-plasma)=5.01

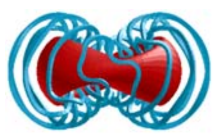
R/Δ_{min} (coil-coil)=17.8

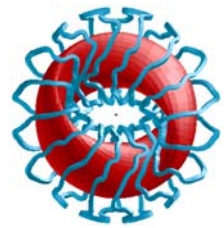
I /R-B (max)=0.314 MA/m-T

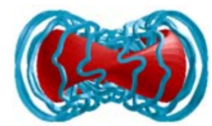
coil lengths/R=5.75, 5.29, 5.18, 5.20

B(max)/B(0) = 2.94 for 0.4 m by 0.4 m square conductors.

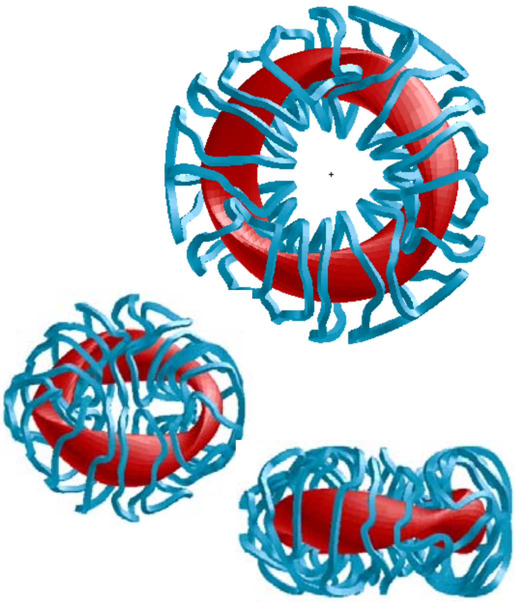








Long Po Ku presented his newest results on the three-field period, 4.5 aspect ratio configuration (KQ26A) of the SNS/LPS family in which the iota profile is specified that the low-order resonance is minimized. The operating beta is estimated to be in the range of 4%. Good QAS was predicted with minimal nonaxisymmetric residues and effective ripples (0.7% at a beta of 4%.) Alpha losses are expected to be around 7%. This configuration also has good equilibrium flux surface quality, but there are some remnants of the m=4 islands. It is slightly unstable to both low and high-n internal modes. Flux shaping may help stabilize the free-boundary modes and external kinks and ballooning modes. The level of effective ripples may result in enhanced loss of alpha particles. The physical properties of this configuration are shown below. Another configuration from the 6 SNS family will be evaluated and developed during the next few months.



KQ26Q_13 Coil Characteristics:

No. of Coils: 6/period

Different Types of Coils: 3

R/Δ_{min} (coil-plasma)=5.8

R/Δ_{min} (coil-coil)=10.2

I /R-B (max)=0.278 MA/m-T

coil lengths/R=4.87, 4.49, 4.52

B(max)/B(0) = 2.11 for 0.4 m by 0.4 m square conductors

Progress on Divertor Heat Load Assessment – TK Mau showed the divertor assessment strategy flow chart with the configuration optimization codes coupled with the divertor design codes. His assessment strategy is to develop acceptable stellarator equilibrium conditions and a suitable free-boundary VMEC configuration. After obtaining field line tracing in the SOL, the divertor plate locations will be adjusted to satisfy the heat load limitations. The divertor surface heat load is partially due to energetic alpha particle loss from the plasma. Calculations for alpha particle orbits including gyro-motion were completed for alphas with energies ranging from 0.035 to 3.5 MeV in an example tokamak-like configuration. Future effort will include analysis for both the NCSX-like and 3-FP configurations, more accurate modeling of the divertor locations, and field line tracing.

GOURDON/GEOM Parallelization Progress and Outlook – Hayden McGuinness described the flow charting of the codes that interact with the GOURDON code to establish coil currents, mass profiles alpha losses and magnetic field lines. The output from the GOURDON code is the heat loads for given geometry surfaces. The GEOM code describes the geometry of the stellarator with a 2-D matrix determining the plasma surface, the scrapeoff layer and the wall boundary. The intent is to trace the guiding centers and calculate the intersection points for the divertor plates, first wall, and last closed magnetic surface (LCMS). The present effort is to enable the code to be run in parallel on a many PC network. The code has been benchmarked the LCMS of VMEC for the W7-X machine to a reasonable degree of accuracy except for the plasma tip regions. The next effort will be to test a full case for the entire field period. Xueren Wang will provide the first wall surface definition to Hayden. Hayden intends to probe the island structure for divertor applications.

Beta Limits for Compact Stellarators: Are They Real? - Alan Turnbull stressed that both W7-AS and LHD have exceeded the ideal MHD Beta limits. W7-AS has achieved an average beta of 3.4%, whereas the predicted stability limit was ~2%. LHD has achieved an average beta of ~ 4%, clearly violating the predicted interchange limit at low beta. Moreover, W7-AS has achieved average beta values of greater than 3.2 for times in excess of 100 τ_E. Some MHD activity has been observed in W7-AS at intermediate beta values and high-n instabilities are observed, but the plasma will progress to operate at higher beta values. Much of this behavior is seen in LHD.

Several theories exist as to why the ideal MHD beta limits are being exceeded. The maximum beta at low iota is close to a classical equilibrium limit if the axis is shifted inward a value of ~a/2. A degradation of the equilibrium may establish the W7-AS beta limit. Confinement models of container-like or sponge-like representations are being considered. Recent progress in equilibrium reconstructions has allowed a reconstructed self-consistent W7-AS equilibrium for a 3.4% beta plasma. The consensus of the community believes that:

- Maximum beta is not limited by MHD activity
- Maximum beta reached is much higher than predicted by linear stability thresholds
- Maximum beta appears to be controlled by loss of flux surface quality

<u>Modeling of Particle and Power Control for Compact Stellarators – An Update</u> - Arthur Grossman reported that he achieved agreement between the MBFE code and the VMEC LCMS results. Arthur presented the VMEC parameters used and the resultant VMEC output data for iota values form S=0 to S=1. He also presented the iota results from field line tracing inside the VMEC LCMS. His modeling of the iota from field tracing line indicated a 3/5 island structure outside the LCMS that is consistent with an iota of 0.6. The maximum width of these islands was obtained at the bullet shaped plasma cross-sectional area.

Compact Stellarator Reactor Engineering Assessment

<u>Power Core Engineering: Status and Next Steps</u> – Rene Raffray explained the major engineering focus during Phase II: divertor design and analysis, design analysis of the dual cooled blanket, analysis of the coils, and integration of the power core elements with maintenance considerations. He summarized the action items from the prior project meeting.

Rene showed three possible divertor plasma facing material concepts. The helium-cooled divertor will require a heat transfer enhancement technique. The plasma group is expected to provide a credible heat flux magnitude and area location. He suggested a separate town meeting specifically devoted to the divertor design.

The dual-cooled blanket module design is being analyzed. It is envisaged to cut/re-weld the coolant access pipes to the modules from the outside at a location between the shield and the poloidal manifolds. K. Ioki, of ITER, has been contacted about the remote handling aspects of this method. (This point was discussed with Ioki after the meeting. In his opinion, the space provided for cutting/re-welding should be sufficient for commercially available tools, but the space available inside the plasma chamber is very small for the insertion of an articulated boom. It may be mandatory to start any blanket exchange at the port location and to remove all blanket modules at this toroidal level first before the entire boom can be inserted.)

A corrosion workshop was held by Russ Jones at UC Berkley on February 17-18, 2005. The scope was to identify issues associated with corrosion of materials in reactor environments. Most of the material experts were unaware of the specific fusion reactor environments. The zeroing of the materials budget was an influencing factor in the discussions. Of particular emphasis was the understanding of the older, existing compatibility criteria and material properties, especially the margins designed into the specified temperature limits. Areas of fusion interest are the compatibility of ferritic steels and SiC-composites with Pb-17Li, and the compatibility between helium (impurities!) with tungsten and other refractory metals (Nb, Ta), and their alloys.

The design approaches for ancillary systems are important in these areas: tritium extraction from the exhaust stream and recovery and heat exchanger design and material choices for a dual cooled design. There is expected to be some knowledge to be gained from the ITER blanket test program.

The structural analysis of the coil structure is needed. It has been suggested to determine the mechanical forces between coil windings and supporting tube based on the current in each coil, and to use this as input for a detailed FE-analysis of the supporting tubes and the bucking cylinder.

<u>Comparison of 2-FP and 3-FP Compact Stellarator Coil Geometry Configurations</u> – Xueren Wang reviewed the 3-FP configuration for the R=8.25 m design with and without the plasma. Xueren then illustrated the modular maintenance approach for this configuration. Three ports are available that would accommodate module sizes of 2 m x 2 m.

Xueren then explained the new 2-FP coil configuration (a = 1 m, A = 2.75). This configuration would accommodate a range of large module sizes, depending on port location. After assessing the port sizes, Xueren concluded modular maintenance would be best accomplished with ports between adjacent coils.

<u>Dual-Cooled Blanket Modular Replacement Design Approach</u> - Xueren Wang showed the dual-cooled blanket module that can be replaced through a few large ports (3-FP approach). He reviewed the radial build provided by Laila El-Guebaly. This radial build is used to create the exploded view of the blanket that shows the flow passages through the module. Approximately sixty percent of the thermal power is extracted with the Pb-17Li coolant. The inlet coolant temperature is 460ºC and the outlet coolant temperature is 700ºC. The inlet and outlet coolants flow through coaxial tubes. A similar layout shows the helium coolant flow paths through the blanket module with coaxial inlets and outlets. Helium flows cool the front wall, side walls, and separation plates of the module. A three-stage compression Brayton cycle is being used to achieve gross thermal conversion efficiency between 40% and 42%. Xueren provided the thermal hydraulic system input data and results for the cycle efficiency a function of the FS/LiPb interface temperature and neutron wall loading.

Xueren illustrated an approach for mechanical attachment of the modular blanket to the coolant manifold behind the blanket. He also explained his approach for cutting and rewelding the coaxial tubes. His approach for the coolant manifolds was shown for several locations around the power core. The supply and return helium and LiPb coolant manifolds are arranged inside the VV to reduce the number of VV penetrations. The basic idea of this design is to operate blanket modules, shields, poloidal manifolds, and the outer tubes of the toroidal coolant access pipes at nearly uniform temperature, and to provide sliding bearings between these zones and the cooler vacuum vessel.

Changes to LiPb/FS/He System and Implications for Radial Build – Laila El-Guebaly outlined the selected blanket concepts for the internal vacuum vessel concept, namely LiPb-cooled blankets with either SiC or FS internal structures and shield coolant of LiPb or He, respectively. Laila explained the latest changes to the blanket and shield design for less blanket coverage, LiPb and He manifolds, segmented WC-shield, and no gap between back wall and shield. These changes infer lower TBR, increased radial build thickness, and higher Waste Disposal Rating. Laila then discussed both the prior and current radial builds with and without the manifolds. Details of the manifolds were shown. She also compared the plasma to mid-coil distances for the shield only, nominal blanket and shield w/o manifold, blanket and shield with manifold, and the SPPS configurations.

Specific coverage fraction and compositions were presented. A coverage fraction was shown as a map of the field period surface. This determined the areas where the transition from shield-only to blanket-only occurred. Laila schematically illustrated this transition zone, which then determined the thicknesses and compositions of different materials. These data then established the local tritium breeding ratio for the transition zone (TBR=0.84) as compared to the full blanket (TBR=1.25), which yielded an overall TBR of ~ 1.1. Laila showed the decay heat as a function of time. The afterheat from the WC components dominate for an hour, and then they decay slowly over time. It has been mentioned in the discussion that self-shielding of W could reduce activation and afterheat considerably, and should therefore be assessed. The WDR for the FW/WC-shield-I/back wall is 0.9 and the WC-Shield-II is 0.3. The segmented WC shield generates Class C low-level waste.

Laila has posted her results and definition data on her web site http://fti.neep.wisc.edu/aries-cs/builds/build.html. Her future work is to generate LiPb decay heat, impact of divertor on TBR, perform a 3-D TBR analysis, heating loads, and radial build for 2-FP configuration.

Helium-Cooled Divertor Development in the EU: Helium Jet Cooled Divertor HEMJ – Thomas Ihli said the EU divertor requirement has a heat load of 10 MW/m^2 with strike point movement over a 10-year lifetime. This was accomplished with high-pressure helium (100 bars) coolant. Tungsten was adopted as the plasma facing armor material. The first approach was to use a coaxial helium delivery with the inner tube helium flow directed to a tungsten cap with the armor attached. Several flow/cap schemes were analyzed with the hemispherical cap with holes being the best baseline approach. The holes provided many jet impingements that have been successfully used on other applications. Several fabrication approaches were analyzed to be assembled into a larger assembly and have a reliable bond between the component materials. CFD was used to optimize the hole size, number, location, and pattern as well as the flow rate. With the baseline parameters, thermal analyses were accomplished on the armor and cap. These configurations were tested to confirm the expected performance parameters. The cap hole pattern was tailored to different locations on the outboard divertor. The intent is to develop an approach applicable to several applications such as DEMO, commercial power plants, etc., and divertor tests in ITER.

He-Cooled Divertor Design Approach – Thomas Ihli said he would consider the tungsten caps, tubes and plate design approaches for the ARIES-CS divertor. He discussed the need to have flat surfaces rather than the surfaces of parallel tubes. Regardless of the surfaces, the tube configuration must be analyzed: series, U-tubes in parallel, L-tubes in parallel, or T-tubes in parallel. He then showed analyses for the T-tube configuration with coaxial feed tubes. Temperature gradients limit the size due to bending distortion.

He then discussed the transition region from Fe to W surfaces, end tube designs, and heat transfer enhancement approaches with some thermal analyses. Then he showed some design details of the divertor assembly. He feels this design has some technical promise, but will consider other options for assessment.

Pressurization Accidents in ARIES-CS – Brad Merrill presented his modeling assumptions and parameters for the baseline pressurization accidents. He captured the geometry and physical characteristics of the power core. The helium loop pressure is 8 MPa and the LiPb loop is 4 MPa. The loop configurations are assumed to be similar to ARIES-AT design. The MELCOR model is also based on the ARIES-AT plumbing schematics and building configuration. The LOFA for both the LiPb and He results in a prompt temperature spike to 630ºC followed by a cooling period that stabilizes the temperature of most components around 500ºC.

Brad listed several possible reference accidents including helium and water LOCA to demonstrate pressurization does not fail confinement and limited chemical reactions. A LOCA of in-blanket helium would be assessed for gas pressurization and leakage. These analyses are applied to the ex-vessel helium and LiPb LOCA to determine pressurization and effects on the FW/blanket. He feels these accidents are the most likely confinement bypass event initiators.

Brad showed how two pressurization accidents of the VV for modular maintenance were considered: a single FW channel rupture and a helium header inlet failure. Shutdown and loss-of-coolant occurs after 1 hour and VV cooling enters natural convection modes. Internal pressures reach 18 atmospheres within a few seconds for a large break and a few minutes for a small break. Both will require a pressure relief system.

For the field period maintenance approach, two accidents were also considered. Shutdown and loss-of-coolant occurs within 1 hour and VV cooling enters natural convection mode. Pressures reach two atmospheres within seconds for large breaks and 10 minutes for small breaks. This may not require a pressure relief system.

LOCA/LOFA Analyses for Blanket and Shield Only Regions – LiPb/FS/He System – Carl Martin said the gap between the blanket and the shield has been removed and replaced with a perfect contact condition. Adiabatic conditions at the back of the vacuum vessel are still assumed. An emissivity of 0.3 is assumed across the vacuum gaps. The analyses have been rerun. The maximum temperature for LOCA and LOFA as a function of the heat flux are presented to establish thermal operating limits.

The LiPb LOFA and LOCA for He results in a 715ºC temperature excursion for a 1-cm gap between the blanket and shield. Perfect contact conditions results in a 701ºC excursion. A

LOCA in the FW/blanket and LOFA in VV results in a temperature excursion of 729ºC (1-cm gap) and 706ºC (no gap). Scaling the input power by 1.5 exceeds the limit to 801ºC and 760ºC, respectively, which exceed the design limits for reusable components (740ºC). Carl also showed the maximum temperature conditions for parametric neutron wall loading values and recommended an average NWL ≤ 2.3 MW/m^2 to meet the design limit.

Carl also analyzed the thermal excursions for the WC-shield-only configuration. A maximum temperature of 1427ºC occurred after 24 hours. Natural convection of water in VV was assumed. At a time of one day, there is a large gradient across the vacuum gap. The maximum WC shield temperature is dependent on gap emissivity.