Electromagnetic Disruption Effects in the ARIES-RS Tokamak Design

J.A. Crowell and J.P. Blanchard

October 2000

UWFDM-1148

Presented at the 14th Topical Meeting on the Technology of Fusion Energy, October 15–19, 2000, Park City UT

FUSION TECHNOLOGY INSTITUTE

UNIVERSITY OF WISCONSIN

MADISON WISCONSIN

ELECTROMAGNETIC DISRUPTION EFFECTS IN THE ARIES-RS TOKAMAK DESIGN

Jeffrey A. Crowell
Sandia National Laboratories
P.O. Box 969, MS9042
Livermore, CA 94551–0969
650-566-9224

James P. Blanchard
University of Wisconsin—Madison
1500 Engineering Dr., ERB 143
Madison, WI 53706–1687
608-263-0391

ABSTRACT

A model of the induced currents and resultant electromagnetic forces and stresses during a disruption event in the ARIES-RS tokamak design is presented. Like many other power reactor concepts, the ARIES-RS has a modular design consisting of toroidally segmented internal structures to limit disruption induced currents and facilitate maintenance. During a disruption, currents driven in these structures cross the large toroidal magnetic field producing substantial electromagnetic forces.

To consider these effects, a transient three-dimensional electromagnetic finite element model of the ARIES-RS device was created, implemented by the commercial code ANSYS. The same code was used for dynamic structural analysis.

The model includes all major components (the first wall, blankets, divertor plates, stabilizing shells, vacuum vessel, shields) electromagnetically coupled together. The magnets are assumed to remain energized during the disruption. The plasma current decay is prescribed, not coupled with the current in the structure. Halo currents and vertical displacement events, which may produce larger forces than found in this study, are not considered.

The results illuminate important design tradeoffs. For a centered fast plasma current quench, the outboard first wall/blanket modules were found to experience the most severe loading. Current in the sidewalls generates forces that produce large torques on these structures. Supports are needed to react these loads; however, thermal stress considerations drive designs toward a first wall with a compliant support system. Thus, supports needed to reinforce against disruption loads can lower the maximum permissible heat flux on the first wall. Further, electromagnetic pressure on the first wall requires a factor of two reduction in the coolant channel width in some regions, resulting in higher pumping power.

Extension of the results to other modular tokamak designs is discussed.

Color versions of the figures are available on the world wide web at http://silver.neep.wisc.edu/disrupt or by contacting the authors.

I. INTRODUCTION

This study investigates the electromagnetic and resulting dynamic structural behavior of the ARIES-RS tokamak design during a disruption event to identify the components that are at risk and illuminate design options which mitigate these loads.

A disruption event in a tokamak occurs when magnetohydrodynamic instabilities in the plasma cause a rapid loss of plasma energy confinement. The plasma current decays away on a time scale of milliseconds in the resulting cold and highly resistive plasma. This change in plasma current induces currents in the surrounding conducting structures. The induced currents, flowing in the presence of a magnetic field, can apply substantial electromagnetic forces on the near-plasma components.

Chazalon, et al.[1] reviewed the frequency of disruptions in large experimental tokamaks such as JET and TFTR and found that between 5 and 50% of the discharges ended in a disruption. A statistical analysis

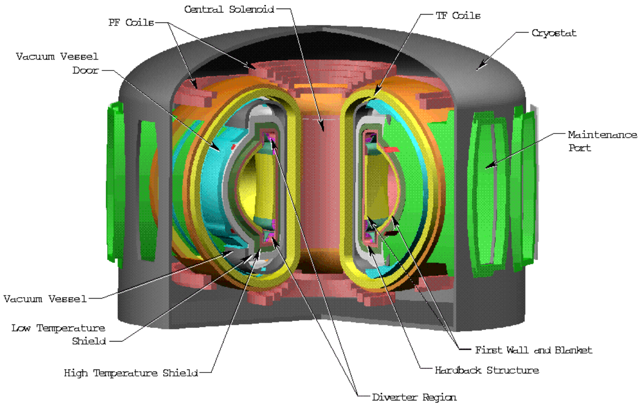

Figure 1: The ARIES-RS tokamak design.[3]

of disruptions in the ASDEX device was performed by Zohm, et al.[2] They found that, after eliminating the most dangerous operation regimes, 17% of all shots end with a disruption. These experiments often explored operational limits; lower disruption frequencies occur when these devices are operated more conservatively. Damage from disruption-induced currents has occurred in JET as well as other experimental tokamaks.

The proposed ARIES-RS reactor exceeds these experimental devices in major radius (5.52 m), plasma current (11.3 MA) and on-axis magnetic field (8.0 T),[3] making disruption damage a greater concern. Figure 1 shows a cut-away view of the device. The first wall and blanket material is vanadium alloy. Tenelon (a high manganese steel) is used both in the vacuum vessel and with vanadium alloy in the shields. The passive stabilizing shells are tungsten. More on the ARIES-RS concept can be found in a series of articles in reference 4.

This investigation used the finite element method with a differential formulation as implemented by the commercial finite element code ANSYS version 5.3 (University high-option)[5]. A transient three-dimensional electromagnetic model of the ARIES-RS design was created. ANSYS was also used for dynamic structural analysis of some components.

II. FINITE ELEMENT MODEL

To facilitate component replacement and maintenance and reduce thermal stresses, the power core of the ARIES-RS design is toroidally segmented into 16 sectors. As shown in Figure 2, an individual module includes the first wall, divertor plates, blanket, and some shield structures. The lack of toroidal electrical continuity in these near-plasma components greatly reduces the disruption induced currents. However, the currents that are induced must cross the strong magnetic field, generating substantial electromagnetic forces.

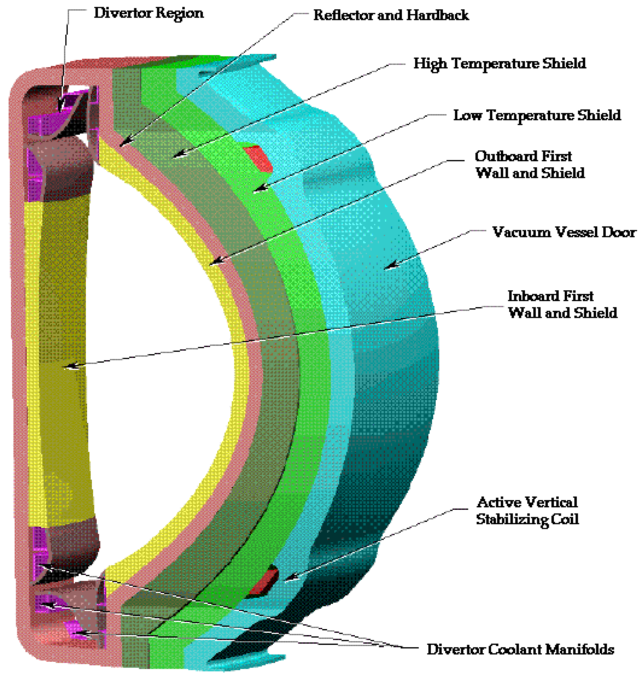

Figure 2: A sector module of the ARIES-RS tokamak.[6]

Modeling the current induced in these box-like structures requires a three-dimensional formulation. There are several such formulations available. The most successful of these solve for electric and magnetic fields in terms of potential fields. These approaches select potentials which automatically satisfy one or more of Maxwell's equations, reducing the number of equations and unknowns which must be simultaneously satisfied. The formulation used in this study employed a magnetic vector potential with an electric scalar potential in conducting regions.

Note that each ARIES-RS module is identical, and each is symmetric about a vertical midplane. If the decaying plasma current does not become azimuthally asymmetric, then a substantial reduction in problem size can be achieved. As discussed in reference 7, modeling a half-module with appropriate boundary conditions produces results applicable to the entire system—only 1/32 of the full domain needs to be solved. (Two solutions using different boundary conditions need to be found and appropriately combined.) For a "centered" disruption, where the plasma current does not move as it decays, symmetry about the horizontal midplane permits a further reduction in problem size by a factor of

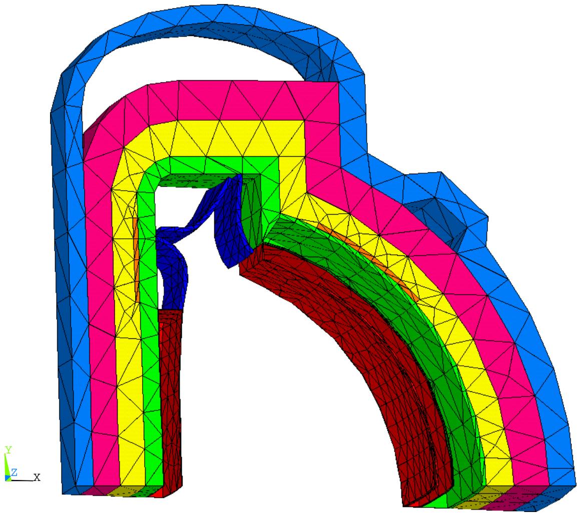

Figure 3: Finite element mesh of the conducting structures.

two.

One quarter of a module was modeled as the finite element mesh shown in Figure 3. For clarity, only conducting elements are shown—the model includes additional elements which make up the toroidal and poloidal field coils and fill the plasma chamber and external space. "Infinite" elements modeled the far-field boundary.

Some of the ARIES-RS blankets are internally divided into an array of poloidal coolant channels. The walls forming these channels have electrically insulated surfaces to limit magnetohydrodynamic pressure drops in the liquid metal coolant. These walls were not included in the model. It is expected that neglecting these additional conduction paths will cause the computational results to underestimate the induced currents.

Since the superconducting toroidal and poloidal field coils will have a quench time on the order of seconds, it was assumed that they remained energized during the comparatively brief disruption.

A fast plasma current quench of 10 ms where the plasma remains stationary during the disruption was chosen as the reference case for this study, for consistency with previous design studies[8] and because this scenario is considered representative of one of the most severe seen experimentally. Vertical displacement events, which may produce more severe forces, were not considered in this study. The current quench is modeled as a linear rampdown of the plasma current density at all points in the plasma chamber. Halo currents, which are currents following through both plasma and structure, were not included in the model.

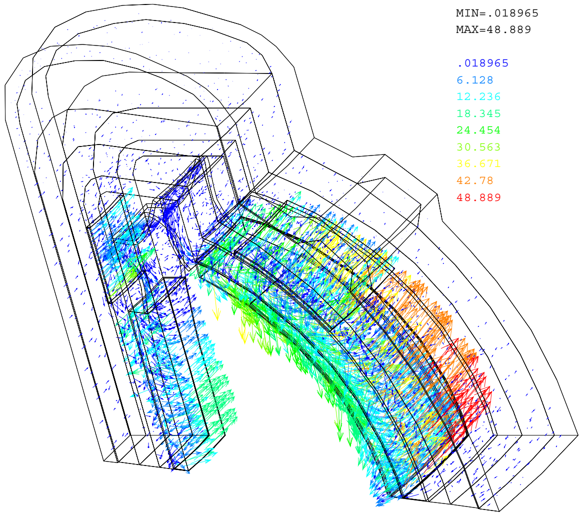

Figure 4: Current density (in $\mathrm{MA/m^2}$) in the conducting structures at the end of a 10 ms disruption.

III. RESULTS

The current density and magnetic force density in the conducting structures of the ARIES-RS device at the end of a 10 ms centered disruption are shown in Figures 4 and 5. At this moment, the currents and forces both peak. The current density is at a maximum in the near-plasma wall of the outboard blanket/reflector (a blanket behind and independent of the first wall/blanket structure). However, the greatest magnetic force density is applied to the outboard first wall/blanket. This component experiences the greatest electromagnetic disruption induced stresses.

Figure 6 plots current density and Figure 7 plots the magnetic force density in the outboard first wall/blanket.

The internal walls which form coolant channels inside the outboard first wall/blanket join the first wall, kink stabilizing shell and back wall so that they behave mechanically as a composite plate. To determine the mechanical behavior of this component, a structural model of plate elements was made and the magnetic forces computed over time in the electromagnetic model above were mapped onto this model.

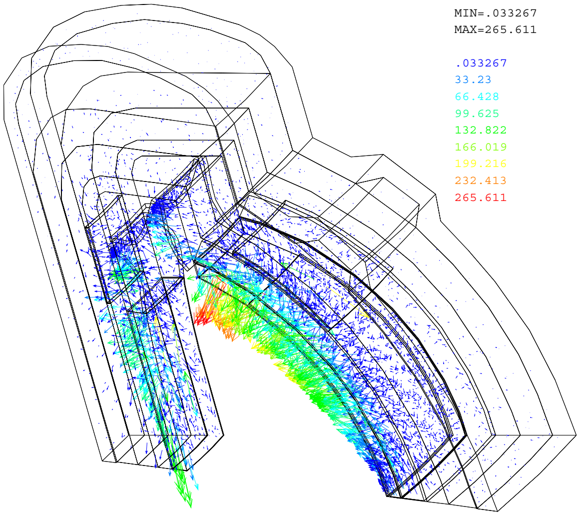

Figure 5: Magnetic force density (in MN/m^3) in the conducting structures at the end of a 10 ms disruption.

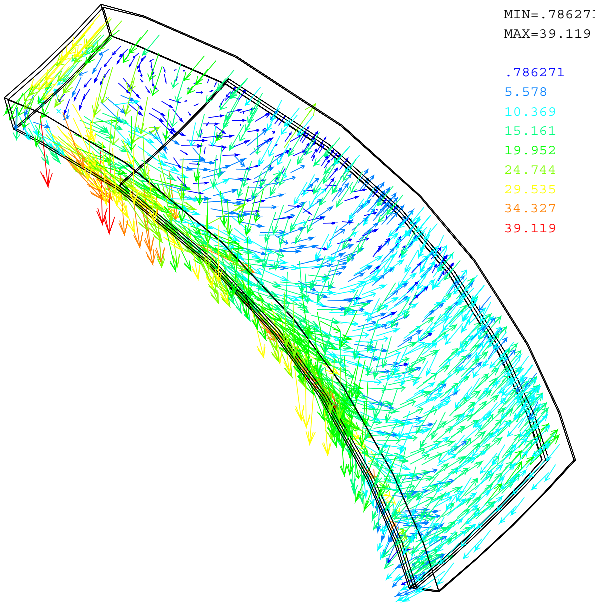

Figure 6: Current density (in MA/m^2) in the outboard first wall/blanket at the end of a 10 ms disruption.

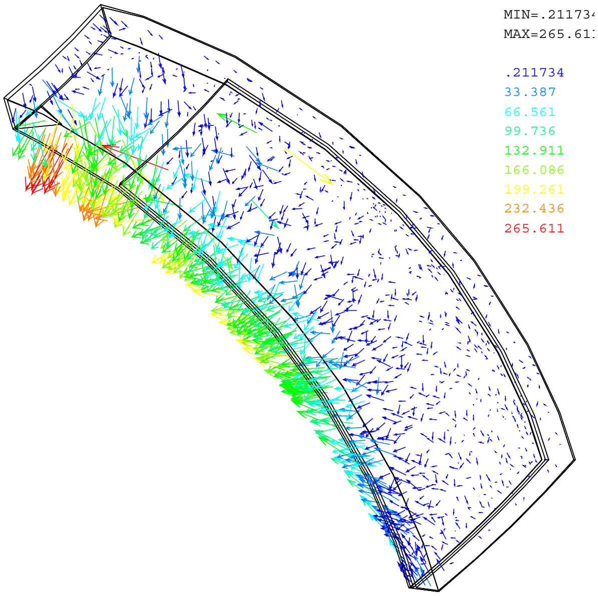

Figure 7: Magnetic force density (in MN/m^3) in the outboard first wall/blanket at the end of a 10 ms disruption.

The structural supports for this component have not been specified in the ARIES-RS design. To minimize thermal stresses compliant supports are desirable. However, note from Figure 7 that much of the load is applied within the side wall of the first wall/blanket where the current crosses the strong toroidal field. Supports are needed along the sides of the component to react this load and prevent disruption-induced stresses from becoming excessively large.

If the edges of the first wall/blanket structure are supported against displacement (but allowed to rotate), results show a peak von Mises stress of about 110 MPa occurs on the back surface of the component at 1.25 ms after the end of the disruption (see Figure 8). This stress is not in excess of the design allowable stress of 156 MPa in bending.

However, in addition to stresses computed by this structural model, there is localized stress from electromagnetic pressure across the the span of each coolant channel. This is a concern in the 3 mm first wall where the coolant pressure applies 0.5 MPa of pressure causing 156 MPa of stress—a marginal design even without disruptions. The peak electromagnetic pressure of 0.76 MPa adds 240 MPa of stress which would fail the first wall. A reduction in the width of the coolant channels by about a factor of two is required.

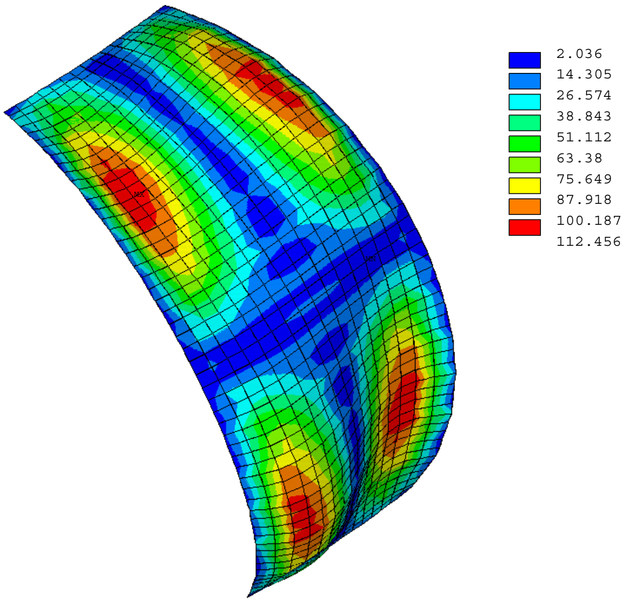

Figure 8: von Mises stress (in MPa) on the back surface of the outboard first wall/blanket 1.25 ms after the end of a 10 ms disruption. Unlike Figures 6 and 7 which show a quarter module, this model represents a full module.

The loads on other components are less severe. While the peak induced current in the vacuum vessel is 9.6 MA (nearly the original plasma current), this current is largely aligned with the magnetic field. As a result, the peak hoop stress is 56 MPa in the inboard region, well below the allowable stress for tenelon. Other components are discussed in reference 7.

It is difficult to extend these results to other tokamak geometries. However, all else being equal, the currents and magnetic forces and corresponding stresses in the first wall/blanket structures are approximately inversely proportional to resistivity for materials with higher resistivity than vanadium alloy and inversely proportional to disruption duration for disruptions longer than 10 ms. By contrast, for the toroidally continuous components such as the vacuum vessel, the maximum currents, magnetic forces and resultant stresses are, to a point, independent of material resistivity or disruption duration.

IV. CONCLUSIONS

The most significant design issues for the ARIES-RS tokamak arise in the outboard first wall/blanket structure. In the design of plasma facing components, there is a trade-off between supporting the structure rigidly enough to limit disruption induced stresses but compliantly enough to mitigate stresses from differential thermal expansion. The results show that for ARIES-RS, the blanket structures must be supported at their edges due to large loads on their side walls.

Disruption-induced stress in the thin first wall was found to be excessive. A reduction in coolant channel width by a factor of two is required.

All other components were found to be within acceptable stress limits during a severe centered disruption.

REFERENCES

1. Max Chazalon, Jean-Louis Boutard, Michael Ian Budd, Antonino Cardella, Wolfgang Dänner, Paul Dinner, Dain Evans, Markus Iseli, Bernard Libin, Frans Moons, Jos Nihoul, Marc A. Vassiliadis, Gottfried Vieider, Chung Hsiung Wu and Ezio Zolti, “Next European Torus In-Vessel Components,” *Fusion Technology*, vol. 14, pp. 82–144 (1988).
2. H. Zohm, K. Lackner and C. Ludescher, “Statistical Analysis of Disruptions in ASDEX,” *Nuclear Fusion*, vol. 33, no. 4, pp. 655–662 (1993).
3. F. Najmabadi and the ARIES team, “Overview of the ARIES-RS Reversed-Shear Tokamak Power Plant Study,” *Fusion Engineering and Design*, vol. 38, pp. 3–25 (1997).
4. M. A. Abdou, O. G. Filatov, et al., eds., *Fusion Engineering and Design*, vol. 38, (1997).
5. ANSYS, Inc., Canonsburg, PA.
6. S. Malang, F. Najmabadi, L. M. Waganer, and M. S. Tillack, “ARIES-RS Maintenance Approach for High Availability,” *Proc. 4th International Symposium on Fusion Nuclear Technology*, Tokyo, Japan, April 1997, *Fusion Engineering and Design*, vol. 41, pp. 377–383 (1998).
7. J. A. Crowell, *Designing Tokamaks to Withstand Electromagnetic Disruption Loads*, pp. 32–49, Ph.D. Thesis, University of Wisconsin—Madison (1999).
8. R. O. Sayer, “TSC Disruption Simulations for the ITER 24 MA Design,” ITER report number ITER/US/95/INV-VV-01, p. 3 (1995).