CONF-791204--26

STARFIRE - INITIAL CONCEPTUAL DESIGN OF A COMMERCIAL TOKAMAK POWER PLANT

by

C.C. Baker, M.A. Abdou, D. DeFreece, C. Trachsel,
D. Graumann and J. Kokszenski

Prepared for

Second Miami International Conference
on
Alternative Energy Sources

Miami Beach, Florida

December 10-13, 1979

UofC-AUA-USDOE

ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

Operated under Contract W-31-109-Eng-38 for the
U. S. DEPARTMENT OF ENERGY

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C. C. Baker and M. A. Abdou
Argonne National Laboratory
Argonne, Illinois 60439, U.S.A.

D. DeFreece and C. Trachsel
McDonnell Douglas Astronautics Company - East
St. Louis, Missouri 63166, U.S.A.

D. Graumann
General Atomic Company
San Diego, California 92138, U.S.A.

J. Kokoszenski
Ralph M. Parsons Company
Pasadena, California 91124, U.S.A.

EXTENDED ABSTRACT

Argonne National Laboratory, in conjunction with an industrial team led by McDonnell Douglas Astronautics Company, and including General Atomic Company and the Ralph M. Parsons Company, is carrying out a comprehensive conceptual design study called STARFIRE of a commercial fusion tokamak power reactor and balance plant. The purpose of the study is to provide a mechanism for the U.S. Department of Energy to further assess the commercial potential of tokamak magnetic confinement for power reactors. The initial reference parameters for a helium cooled option are summarized in Table 1. This study is placing particular emphasis on utility requirements, safety and maintenance considerations.

TABLE 1. STARFIRE - INITIAL CONCEPTUAL DESIGN PARAMETERS FOR HELIUM COOLED SYSTEM

Fusion Thermal Power	3020 MW(th)
Total Thermal Power	3500 MW(th)
Net Electric Power	1000 MW(e)
Average Neutron Wall Loading	2.5 MW/m^2
Reactor Major Radius	7.2 m
Torus Aspect Ratio	3
Plasma Elongation	1.6
Plasma Average Toroidal Beta	8%
Toroidal Magnetic Field	3.8 T
Maximum Toroidal Magnetic Field Including Extra Field Margin of 1.7 T	9.0 T
Plasma Current	17 MA
Plant Availability Goal	75%
Plant Lifetime	30 years

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General Atomic Company
San Diego, California 92138, U.S.A.

J. Kokoszenski
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at

2nd Miami International Conference on Alternative Energy Sources

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* Work supported by the U.S. Department of Energy.

The safety ramifications are being considered simultaneously with the evolution of the STARFIRE design. Efforts are being made to minimize the tritium inventory and the radioactivity induced in the candidate first wall and structural materials in the blanket and shield. Included in the calculations are the total radioactivity both during operation and after shutdown or removal, decay heat, total biological hazard potential in air and atmospheric dispersion following a hypothetical accident. Radiation levels in various portions of the plant during normal operation are also being calculated.

The design of the first wall and tritium breeding blanket is based heavily on a previous comprehensive review of possible blanket/shield designs [1]. Table 2 presents a summary of the combinations of coolant, tritium breeding and structural materials under study at this time.

TABLE 2. BLANKET OPTIONS

Coolant	He	H_2O	Li
Tritium Breeder	Li_2SiO_3*	Li_2SiO_3*	Li
Structural Material	FS/Ti	Ti/FS	V/FS
Neutron Multiplier	PbO	PbO	--

FS - ferritic steel

* Backup choices $LiAlO_2$, Li_2O

The first wall and blanket are designed as a separate system with only the necessary minimum of structural and mechanical interfaces to other reactor subsystems, to permit removal of complete first wall blanket circumferential segments for maintenance operation. Both inner and outer first wall and blanket regions are separate from the surrounding shield system, which has a design life equal to plant design life.

Major maintenance features of the design include large removal blanket sectors, location of the welded vacuum boundary in the benign environment of a fixed shield to assure reliable operation, use of redundant components in auxiliary reactor subsystems to minimize unscheduled shutdowns, EF coils that are raised and lowered for blanket removal, removable shield doors for blanket access, and open access to the top and sides of the reactor.

The reactor hall and remote/hands-on maintenance system is designed for frequent replacement of components that use simple push-pull operations. Items designed for the life of the plant include the overhead crane, TF coils, EF coils, coolant piping and radiation shielding. The blanket modules impurity control components, rf launchers, pumps, valves, fueling mechanisms, etc., are replaced on a scheduled basis. The spares for the superconducting EF coils trapped below the TF coils are stored in place so reactor disassembly is unnecessary in event of EF coil failure.

Availability goals have been established as 85% for the reactor and 75% for the complete plant including the reactor. The reliability of major subsystems has been assessed and allocations of time-to-repair and time-between-failures have been made. These criteria provide a basis for design of maintenance equipment. The maintenance scenario incorporates the current utility practice of shutting down annually for one month of maintenance and a 16-20 week shutdown every 10 years for turbine repair.

A particularly innovative feature that has been adopted for the initial conceptual design is the consideration of various ideas for sustaining the plasma current in a tokamak device, thus producing a steady-state reactor operating mode. Such an operating mode is expected to result in increased first wall lifetime and overall increased system reliability. Particular attention has been given to use lower hybrid rf waves for driving the plasma current.

Plasma current density profiles have been computed due to electron Landau dumping of lower hybrid waves launched into a variety of model density and temperature profiles. The total current and current profile shape are chosen consistent with the requirements of MHD equilibrium and stability against ballooning, ideal kink, interchange, and tearing modes. Toroidal magnetic fields of $\gtrsim 9$ T at the magnet, and very broad current profiles appear to result in the minimum rf wave power required to sustain steady state operation. In addition, plasma temperatures of $T_e = 16$ keV and the hot ion mode of operation ($T_i > T_e$) appear beneficial. A low aspect ratio ($A \approx 3$) is readily achieved for this design since there are no ohmic transformer windings at the center of the machine. It is possible to sustain steady state with 3% of the fusion power recycled as rf power to the plasma, provided narrow spectra lower hybrid waves can be launched with low refractive indices ($n_{\parallel} \approx 1.4$).

An important design consideration is the choice of plasma impurity/alpha particle removal concept. Several candidate concepts have been considered including torus limiter/vacuum pumping schemes, bundle divertors, gas puffing and plasma boundary flows, and additional margin in the toroidal magnetic field. Initial investigations indicate that modest pumping of helium with a limiter pumping system ($\sim$ 15% of the alpha particle flux) coupled with about a 1.7 T margin in the toroidal field may eliminate the need for a divertor, providing that a significant portion of the alpha particle energy can be radiated to the first wall rather than be deposited on the limiter.

REFERENCE

1. D. L. Smith, et al., "Fusion Reactor Blanket/Shield Design Study," Argonne National Laboratory and McDonnell Douglas Astronautics Company, ANL/FPP-79-1 (to be published).