Over the past decade the group at Georgia Tech has examined (Refs. 1-16) the application of a tokamak D-T fusion neutron source, based (insofar as possible) on the physics and technology of ITER, that could drive a sub-critical fast transmutation (burner) reactor fueled with the transuranics (TRU) from spent nuclear fuel (SNF). The purpose of such a reactor would be to stabilize the accumulation of spent fuel being discharged from LWRs by fissioning the TRU in SNF to significantly reduce the number of high-level-waste storage repositories required. Both gas-cooled reactors operating on a “burn and bury” non-reprocessing fuel cycle with TRISO TRU fuel (GCFTR) and liquid metal cooled reactors (SABR, FTWR) operating on a “reprocessing” fuel cycle with TRU metallic fuel have been considered, all driven by essentially the same tokamak neutron source. Determination of the required neutron source parameters for the SABR design is discussed here.

I. The SABR design concept

The SABR reactor is a loop type, sodium-cooled fast reactor fueled with the 40Zr-TRU metal fuel being developed at ANL. A fission power level of 3000 MWth was chosen to provide a significant transmutation rate and because there is experience with nuclear reactors of that power, so existing power handling and balance of plant equipment can be used. The SABR design concept configuration is shown in Fig. 1.

Figure 1. Configuration of SABR

An annular fission core is located outboard of the tokamak plasma chamber, within the toroidal field coils. The reactor core was adapted from an ANL fast reactor design and consists of 918 vertical hexagonal assemblies 15.5 cm across flats, each containing 271 wire-wrapped 3.63 mm radius fuel pins with an active height of 2m. This “leaky” annular core has a small, but positive, sodium void worth and a small negative Doppler coefficient. The fuel assemblies are arranged vertically in four concentric annular rings just outboard of the plasma chamber.

The power density is a modest 72.5 MW/m³ and the heat can readily be removed with Na flowing at 8700 kg/s, which requires 454 kW of pumping power (a lithium-niobate insulating coating on all fuel pins and structure eliminates MHD pressure drop). Heat is removed from the core in two separate sodium loops, each with 4 EM pumps and 2 intermediate heat exchangers coupling to an intermediate sodium loop and finally to a secondary water systems and turbines.

Tritium self-sufficiency (TBR = 1.16) is achieved with a 15 cm Li4SiO4 breeding blanket surrounding the plasma chamber and the reactor core, confirmed with dynamic burnup and production-decay calculations.

II. Fuel Cycle—Determination of Required Neutron Source Strength

A four batch reprocessing fuel cycle in which all the TRU from SNF is fissioned to >>90% was examined, most recently using the ERANOS fast reactor physics code, in order to determine an upper limit on neutron source requirements. The total fuel residence time in the reactor between reprocessing steps is limited by 200 dpa radiation damage in the ODS steel clad and structure. Core refueling would be accomplished remotely.
With 32 MT of “fresh” TRU fuel at beginning of life (BOL), \( k_{eff} = 0.972 \). An out-to-in fuel shuffling scheme was used, in which at the end of a 700 day burn cycle the fuel in the innermost fuel ring (next to the plasma) is removed for reprocessing, the fuel in the other three rings is shifted inward by one ring, and new fuel is loaded into the outermost ring. The fuel removed from the innermost ring is partitioned pyrometallurgically into fission products, which are sent to a geological repository, and transuranics, which are mixed with “fresh” TRU from SNF, refabricated into fuel by arc casting, and recycled into the outermost ring of a SABR (To be conservative, 1% of the fission products are assumed to remain with the transmuranics, and vice-versa.) Each batch of fuel is in the reactor for a residence time of 7.7 years (four burn cycles). After several burn cycles, an equilibrium is established in the compositions at the beginning (BOC) and end (EOC) of the burn cycle. This equilibrium fuel cycle has 29.0 MT of TRU and \( k_{eff} = 0.894 \) at BOC, and 27.1 MT TRU and \( k_{eff} = 0.868 \) at EOC. The corresponding neutron source strengths required to maintain 3000 MW fission power are \( P_{fus} = 75 \) MW (BOL), 240 MW (BOC) and 370 MW (EOC). Four batch reprocessing fuel cycles with fuel residence time limited by 100 and 300 dpa structural material radiation damage limits required maximum (EOC) \( P_{fus} = 220 \) and 460 MW, respectively. Thus, it seems appropriate to design the fusion neutron source to produce up to \( P_{fus} = 400-500 \) MW. If the radiation damage limits could be overcome, TRU burnup greater than 90% could be achieved without reprocessing by leaving the fuel in until \( k_{eff} < 0.6 \), which would require a neutron source strength corresponding to \( P_{fus} >> 500 \) MW to maintain \( P_{fission} = 3000 \) MW.

The TRU fission rate is 1.05 MT per full power year (FPY) of operation. Allowing for 60 days downtime for fuel shuffling after every 700 full power day burn cycle at 76% availability during the burn cycle, SABR could fission all the transuranics in the SNF discharged annually from three 1000 MWe LWRs. For a given batch of fuel that is resident in the reactor for the four burn cycles between reprocessing steps, 23.8% of the TRU is fissioned.

III. Neutron Source Physics Parameters

Standard systems analysis of the tokamak physics and engineering constraints was used to establish the operating parameter range for tokamaks over a range of major radii and plasma currents. For a \( R = 3.75 \) m tokamak, the wide operating range over which the required fusion power levels can be achieved is shown in Fig. 2, and the detailed parameters for \( I = 8.3 \) and 10.0 MA are given in Table 1. For comparison, the corresponding design parameters for ITER also are given in the table. Clearly, the plasma parameters for the neutron source are within the range which is anticipated in ITER, and, with the exception of \( Q_p \), \( \eta_{cd} \) and availability, have already been achieved in existing experiments. Note that the required values of the important \( \beta \) and confinement parameters are routinely achieved in current experiments. The bootstrap current fractions are based on the ITER formula, and the values of current drive efficiency that result therefrom are only somewhat larger than has been achieved (0.45 in JET).

Figure 2: Operating space of SABR(GCFTR) at 10 MA.
(Horizontal lines indicate \( P_{fus} \), slanted lines \( P_{aux} \))
Table 1  Tokamak Neutron Source Parameters

<table>
<thead>
<tr>
<th>Parameter</th>
<th>SABR nominal</th>
<th>SABR extended</th>
<th>ITER</th>
<th>Pure Fusion Electric</th>
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<tr>
<td>Major Radius R, m</td>
<td>3.75</td>
<td>3.75</td>
<td>6.2</td>
<td>5.2</td>
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<td>Current I, MA</td>
<td>8.3</td>
<td>10.0</td>
<td>15.0</td>
<td>13.0</td>
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<td>P$_{fus}$, MW</td>
<td>180</td>
<td>500</td>
<td>400</td>
<td>3000</td>
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<td>Neutron Source, 10$^{19}$/s</td>
<td>7.1</td>
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<td>106</td>
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<td>Aspect Ratio A</td>
<td>3.4</td>
<td>3.4</td>
<td>3.1</td>
<td>4.0</td>
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<td>Elongation κ</td>
<td>1.7</td>
<td>1.7</td>
<td>1.8</td>
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<td>Magnetic Field B, T</td>
<td>5.7</td>
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<td>5.3</td>
<td>5.8</td>
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<td>B$<em>{TF}$/B$</em>{GJ}$ at coils, T</td>
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<td>11.8/13.5</td>
<td>11.8/13.5</td>
<td>11.8/13.5</td>
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<td>Safety Factor q$_{BS}$</td>
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<td>3.0</td>
<td>2.0</td>
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<td>Confinement H$_{IPB98}$</td>
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<td>Normalized Beta β$_{N}$</td>
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<td>Plasma Energy Mult. Q$_{p}$</td>
<td>3</td>
<td>5</td>
<td>5-10</td>
<td>&gt;30</td>
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<td>CD eff., γ$_{cd}$, 10$^{-20}$/A/Wm$^2$</td>
<td>0.61</td>
<td>0.58</td>
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<td>Bootstrap cur. frac f$_{bs}$</td>
<td>0.31</td>
<td>0.26</td>
<td>0.9</td>
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<td>Neutron Γ$_{in}$, MW/m$^2$</td>
<td>0.6</td>
<td>1.8</td>
<td>0.5</td>
<td>4.9</td>
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<tr>
<td>FW Heat q$_{fus}$, MW/m$^2$</td>
<td>0.23</td>
<td>0.65</td>
<td>0.5</td>
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<tr>
<td>Availability, %</td>
<td>76</td>
<td>76</td>
<td>&gt;90</td>
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</tbody>
</table>

IV. Neutron Source Technology

Six 20 MW LHR launchers, based on the ITER design, are used for plasma heating and current drive. The austenitic steel in the ITER first wall and divertor was replaced with an advanced feritic steel (ODS) and the designs were adapted to helium (GCFTR) and sodium (SABR) coolants. The helium coolant required a redesign of the divertor design for GCFTR to get more surface heat transfer area, while the same ITER channel dimensions could be used with sodium for SABR. However, the coolant channel walls must be coated with an electrical insulator (LiNbO$_3$) to prevent excessive MHD pressure drops with sodium. Heat removal capability was confirmed by detailed FLUENT calculations using the adapted ITER designs.

The central solenoid (CS) and toroidal field (TF) coils were scaled down from the ITER designs using the same cable-in-conduit Nb$_3$Sn superconductor surrounded by an Incoloy 908 jacket and cooled by a central channel carrying super-cooled helium, with maximum fields of 11.8 and 13.5 T, respectively. The dimensions of the CS coil were constrained by the requirement to provide inductive startup and to not exceed a maximum stress of 430 MPa, set by matching ITER standards and Incoloy properties. The dimensions of the 16 TF coils were set by conserving tensile stress calculated as in the ITER design, taking advantage of an Incoloy 908 jacket for support.

V. Shielding and Plant Lifetime

A multilayered shield, surrounding the plasma chamber and reactor, protects the superconducting magnets. MCNP and EVENT neutronic analyses predict that during the design lifetime of 30 FPY (40 years at 75% availability), the neutron fluence to the superconductor would be 7x10$^{18}$ n/cm$^2$ and the radiation dose to the insulators would be 7x10$^{7}$ rads, safely below the limiting values of 10$^{19}$ n/cm$^2$ and 10$^{9}$-10$^{10}$ rads. The ODS steel first wall accumulates 200 dpa in 6.5 FPY and would need to be replaced during the 2-month fuel shuffling shutdown after every third fuel cycle.

VI. Dynamic Safety Analysis

Loss-of-power (LOPA), loss-of-heat-sink (LOHSA), loss-of-flow (LOFA), control rod ejection, and neutron source excursion accidents were simulated for the Na-cooled SABR design using the RELAP5 code to model the reactor and heat removal systems dynamics, together with a plasma power and particle balance model for the neutron source dynamics. It was found that: i) the core power can be reduced to decay heat levels in a couple of seconds by turning off the neutron source heating power when any accident condition is detected; ii) a LOPA thus reduces the core to the decay heat level in a couple of seconds and natural circulation prevents core damage; iii) undetected (neutron source remains on) LOFAs in which 50% of the primary coolant pumps fail can be survived without core damage, and only when 75% of the pumps fail does fuel melting occur (at 8.4 s); iv) an undetected LOHSA with 50% loss of sodium flow in the intermediate loop can be survived without core damage, and only with 75% loss of sodium flow in the intermediate loop does fuel melting occur (at 150 s); and v) neutron source excursions due to inadvertent increases in plasma heating or fueling could be limited by operation near inherent density and beta limits. An ECCS was not analyzed for SABR, but an ECCS was designed for the He-cooled GCFTR design which prevented core damage even with total loss of primary flow.

VII. Comparison with Pure Fusion Electric Power
The technical requirements—neutron wall load, confinement, beta, availability—are much less demanding for SABR than for pure fusion electric power, as shown by the last column in Table 1 for the ARIES-AT design.

VII. R&D Requirements

R&D FOR ANY FUSION-FISSION HYBRID

FUSION R&D
1. Fusion neutron source physics and plasma support technology.
2. Tritium breeding & extraction technology.

NUCLEAR R&D
3. Fission reactor physics and technology.
4. TRU processing and fuel fabrication.

MATERIALS R&D
5. Radiation damage resistant structural material (200dpa)

INTEGRATION OF FUSION AND FISSION TECHNOLOGIES

A TOKAMAK FFH NEUTRON SOURCE IS ON THE PATH TO PURE FUSION POWER

FUSION R&D FOR A TOKAMAK F-F HYBRID
1. Ongoing worldwide tokamak physics R&D program, including ITER-specific issues (e.g. ELM suppression, startup scenarios).
2. ITER construction and operation experience. ITER is the prototype for FFH neutron source.
3. Physics R&D on reliable steady-state, disruption-free operation, burn control, etc.
4. Plasma Support Technology (magnets, heating systems, etc.) R&D for component reliability.
5. Fusion Nuclear Technology (tritium breeding, etc.) R&D
6. Advanced Structural Materials (200 dpa) R&D
7. Integration of tokamak and nuclear reactor technologies

FURTHER FUSION R&D FOR TOKAMAK ELECTRIC POWER
7. Physics R&D on improved confinement and operational limits.
8. Advanced DEMO (prototype commercial reactor) construction and operation.

References

Neutron Source for Transmutation (Burner) Reactor

Transmutation (Burner) Reactor Design Studies

Transmutation Fuel Cycle Analyses

Dynamic Safety Analyses