RESEARCH SUMMARY

During 2009, research focused on i) interpretation of plasma edge experiments in DIII-D (DoE Grant DE-FG02-99-ER54538), ii) further analysis of the SABR fusion-fission hybrid actinide burner reactor concept developed at Georgia Tech (DoE Grant DE-SC0002202), and iii) related topics in theoretical/computation plasma physics. This work and publication/presentations based on it are summarized in this report.

FRC FACULTY & STUDENTS

Faculty---W. M. Stacey (NRE), B. Petrovic (NRE), S. M. Ghiaasiaan (ME), J. Schwartz (MSE)
Graduate Students---C. Bae, Z. W. Friis, J.-P. Floyd, C. M. Sommer, C. Stuart, T. S. Sumner (NRE)
Undergraduate Students---R. Lober (NRE), L. Zhou (Phys)

COLLABORATION

Faculty and students in the Georgia Tech Fusion Research Center collaborate in the analysis and interpretation of plasma physics experiments with researchers from General Atomics, Lawrence Livermore National Laboratory, Oak Ridge National Laboratory and other research institutions as a member of the DIII-D National Tokamak Facility. This includes participation in the DIII-D H-Mode Edge Pedestal (HEP) activity which is comparing various codes and methods for interpreting transport in the tokamak edge pedestal.

The Georgia Tech FRC joined an ongoing study at the KIT—Karlsruhe Institute of Technology and CEA—Cadarache of various scenarios for the deployment of actinide burner reactors to support an expansion of nuclear power in Europe. The present study, including critical and accelerator-driven (ADS) fast burner reactors is being expanded to include the SABR fusion-fission hybrid burner reactor concept developed at Georgia Tech.

TECHNICAL PRESENTATIONS

PEER-REVIEWED PAPERS

8. “DIII-D research in support of ITER”, Nucl. Fusion 49, 104008 (2009); E. J. Strait and the DIII-D Team.

TECHNICAL SUMMARIES
Analysis of neutral particle recycling and pedestal fueling in an H-mode DIII-D discharge

Z. W. Friis & W. M. Stacey, Georgia Tech
A. W. Leonard, General Atomics,
M. E. Rensink, Lawrence Livermore National Laboratory
November, 2009
INTRODUCTION

• Experimental ion fluxes to the wall and UEDGE calculations of background edge plasma parameters and recombination rates exist for several DIII-D shots from the work of Leonard, et al.

• The 2D GTNEUT neutral transport code has been modified to accept geometric and background plasma input from UEDGE by Friis, Rognlien, et al.

• This allows a detailed analysis of neutral particle recycling and fueling in a DIII-D H-mode discharge.
Ion Recycling & Recombination Sources (#/s)

From exp.
UPPER BAFFLE  8.13E+20
LOWER BAFFLE  8.13E+20
INNER DIVERTOR  5.55E+21
OUTER DIVERTOR  6.04E+21

From UEDGE
RECOMBINATION (attached)  0.147E+21
RECOMBINATION (detached)  6.243E+21
Neutral Particle Mean Free Path
Poloidal Distribution—Neutral Fluxes Recycling from Wall

Comparison of Neutral Fluxes Recycling off the Wall

- FROM INCIDENT NEUTRAL FLUX: ATTACHED
- FROM INCIDENT NEUTRAL FLUX: DETACHED
- FROM INCIDENT ION FLUX

Neutral Particle Flux (\(\text{# s}^{-1} \text{m}^{-2}\)) Recycling from Wall

Poloidal Length Along Chamber wall (m) (Starting at the outer most cell of the inboard divertor Going in a Clock Wise Fashion Back to the inner most cell of the Private flux region.)
GTNEUT Inward Neutral Fluxes Across Separatrix

Comparison of Partial Inward Flux and Net Inward Flux

- Net Inward Flux: ATTACHED
- Partial Inward Flux: ATTACHED
- Net Inward Flux: DETACHED
- Partial Inward Flux: DETACHED

radial inward neutral flux ($\# \text{s}^{-1} \text{m}^{-2}$) vs. poloidal length along separatrix (m)
# Global Neutral Particle Balance on Entire Computation Domain

<table>
<thead>
<tr>
<th>Source/Region</th>
<th>‘Attached’</th>
<th>‘Detached’</th>
<th>Determined</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Core</strong></td>
<td>0.129E+21 (1%)</td>
<td>0.193E+21 (1%)</td>
<td>GTNEUT</td>
</tr>
<tr>
<td><strong>Ped</strong></td>
<td>0.798E+21 (6%)</td>
<td>1.120E+21 (6%)</td>
<td>‘‘</td>
</tr>
<tr>
<td><strong>Sol</strong></td>
<td>1.910E+21 (15%)</td>
<td>2.670E+21 (14%)</td>
<td>‘‘</td>
</tr>
<tr>
<td><strong>Halo</strong></td>
<td>1.120E+21 (9%)</td>
<td>2.020E+21 (11%)</td>
<td>‘‘</td>
</tr>
<tr>
<td><strong>In Div</strong></td>
<td>4.380E+21 (33%)</td>
<td>7.390E+21 (39%)</td>
<td>‘‘</td>
</tr>
<tr>
<td><strong>Out Div</strong></td>
<td>4.770E+21 (36%)</td>
<td>5.430E+21 (29%)</td>
<td>‘‘</td>
</tr>
<tr>
<td><strong>PFR</strong></td>
<td>0.107E+19 (&lt;&lt;1%)</td>
<td>0.365E+19 (&lt;&lt;1%)</td>
<td>‘‘</td>
</tr>
<tr>
<td><strong>Total Ionization</strong></td>
<td><strong>13.108E+21</strong></td>
<td><strong>18.827E+21</strong></td>
<td></td>
</tr>
<tr>
<td><strong>Sources (#/s)</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Ion recycling</td>
<td>13.217E+21 (99%)</td>
<td>13.217E+21 (66%)</td>
<td>experiment</td>
</tr>
<tr>
<td>Recombination</td>
<td>0.147E+21 (1%)</td>
<td>6.243E+21 (34%)</td>
<td>UEDGE</td>
</tr>
<tr>
<td><strong>Total Sources</strong></td>
<td><strong>13.364E+21</strong></td>
<td><strong>19.460E+21</strong></td>
<td></td>
</tr>
</tbody>
</table>
Poloidal Distribution of Ionization Rates
Poloidal Distribution of Charge-Exchange Rates
Poloidal Distribution of Neutral Density on Pedestal Flux Surfaces

![Graph showing the distribution of neutral particle density in the pedestal region. The graph plots neutral particle density against poloidal length along the separatrix. Different curves represent different values of ρ (0.999, 0.997, 0.994, 0.991, 0.983, 0.984, 0.979, 0.973, 0.965, 0.958, 0.947, 0.933). Each curve is labeled with a specific value of θ, indicating the angle at which the density data was measured. The graph highlights the variation of density with respect to poloidal length and gives insights into the behavior of neutral particles in the pedestal region.]
Poloidal Distribution of Ionization Rate on Pedestal Flux Surfaces

Distribution of Neutral Particle Density in Pedestal: DETACHED CASE

- Rho values: 0.999, 0.997, 0.994, 0.991, 0.983, 0.984, 0.979, 0.973, 0.965, 0.958, 0.947, 0.933

Neutral Particle Density (#/m^3)

Poloidal length along separatrix (m)
IMPLICATIONS

Order of magnitude variation of neutral density, ionization and charge-exchange rates on pedestal flux surface implies:

• A 2D, rather than a 1D, ion transport calculation is needed in the pedestal?

• If neutral penetration determines pedestal structure, then pedestal structure should display similar poloidal variation—exp?
Effectiveness of Individual Neutral Sources in Fueling Pedestal

Fraction of Ionizations from Sources: Detached Case

- INNER DIVERTOR
- OUTER DIVERTOR
- UPPER BAFFLE
- LOWER BAFFLE
- RECOMBINATION

Ionization Fractions from Source

Rho
Ionization Rates in Pedestal from Individual Neutral Sources

Ionization Rates from Sources: Detached Case

- INNER DIVERTOR
- OUTER DIVERTOR
- UPPER BAFFLE
- LOWER BAFFLE
- RECOMBINATION

[Graph showing ionization rates for different sources as a function of Rho]
## Fraction of neutrals from each source ionized in PED+CORE

<table>
<thead>
<tr>
<th>SOURCE</th>
<th>ATTACHED</th>
<th>DETACHED</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>PED</td>
<td>CORE</td>
</tr>
<tr>
<td>Upper baffle recycling</td>
<td>17.5%</td>
<td>4.0%</td>
</tr>
<tr>
<td>Lower baffle recycling</td>
<td>15.2%</td>
<td>3.7%</td>
</tr>
<tr>
<td>Inner divertor recycling</td>
<td>4.0%</td>
<td>0.7%</td>
</tr>
<tr>
<td>Outer divertor recycling</td>
<td>4.7%</td>
<td>0.4%</td>
</tr>
<tr>
<td>Recombination</td>
<td>2.4%</td>
<td>1.3%</td>
</tr>
</tbody>
</table>
CONCLUSIONS

• Majority of recycling neutrals are ionized in the divertor; only about 7% cross the separatrix.

• Neutrals penetration into pedestal is highly non-uniform poloidally—i) need for 2D transport calc. in PED?; ii) poloidally non-uniform pedestal structure if neutrals determine structure?

• Neutrals recycling from baffle are more effective in penetrating pedestal than neutrals recycling from divertor target, but

• Majority of pedestal fueling is due to neutrals recycling or recombining in divertor.
Access to GTNEUT & Documentation

• CODES
  GTNEUT  u4/friis/GTNEUT/
  Geometry Input  /u4/friis/uedge/GTNEUT_DEMO

• DOCUMENTATION
  Methodology—Nucl. Fusion,34,1385(1994);
  Automated Geometry & Background Plasma Input from
  UEDGE---User’s Manual at www.frc.gatech.edu on
  “Neutrals” page.
## Global Ion Particle Balance on CORE+PED

<table>
<thead>
<tr>
<th>Loss--Ion Outflow Across Separatrix (#/s)</th>
<th>‘Attached’</th>
<th>‘Detached’</th>
<th>Determined</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>2.74E+21</td>
<td>2.74E+21</td>
<td>Experiment</td>
</tr>
</tbody>
</table>

**Sources**

<table>
<thead>
<tr>
<th>Neutral Beam Ion Source (#/s)</th>
<th>0.386E+21</th>
<th>0.386E+21</th>
<th>Known</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ionization of Recycling Neutrals (#/s)</td>
<td>0.927E+21</td>
<td>1.313E+21</td>
<td>GTNEUT</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Total Ion Source (#/s)</th>
<th>1.316E+21</th>
<th>1.699E+21</th>
<th></th>
</tr>
</thead>
</table>

| Ion (Outflow – Sources) (#/s)         | 1.424E+21 (108%) | 1.041E+21 (61%) |            |
Comparison of GTNEUT & DEGAS2 Calculation of Pedestal Ionization Rates—’Attached’

[Graph showing comparison of GTNEUT and DEGAS2 calculations of pedestal ionization rate per unit separatrix area for the attached case.]

- X-POINT
- UPPER BAFFLE
- INNER MIDPLANE
- OUTER MIDPLANE
Comparison of GTNEUT & DEGAS2 Calculation of Pedestal Ionization Rates—’Detached’
Comparison of ionization rates in the PED+CORE calculated by GTNEUT and DEGAS2

<table>
<thead>
<tr>
<th></th>
<th>GTNEUT</th>
<th>DEGAS2</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>‘Attached’</td>
<td>‘Detached’</td>
</tr>
<tr>
<td>Source Strength (#/s)</td>
<td>1.34E+22</td>
<td>1.96E+22</td>
</tr>
<tr>
<td>Ionization Rate (#/s)</td>
<td>0.927E+21</td>
<td>1.313E+21</td>
</tr>
<tr>
<td>PED + CORE</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Percent of Source Neutrals Ionized in PED + CORE</td>
<td>6.9%</td>
<td>6.7%</td>
</tr>
<tr>
<td>Calculation Uncertainty</td>
<td>1.8%</td>
<td>3.2%</td>
</tr>
</tbody>
</table>
Particle Pinch and Diffusion Coefficients in the Edge Pedestal

W. M. Stacey, Georgia Tech
R. J. Groebner, General Atomics

November, 2009

We develop a formalism to infer particle pinches and diffusion coefficients from experimental data and apply it to two DIII-D shots.
Derivation

The toroidal and radial components of the momentum balance for ion species “j” can be written as

\[
n_j m_j \left[ (v_{jk} + v_{dj}) V_{\phi j} - v_{jk} V_{\phi k} \right] = n_j e_j E_\phi^A + n_j e_j B_\theta V_{r j} + M_{\phi j}
\]  

(1)

and

\[
V_{\phi j} = \frac{1}{B_\theta} \left[ E_r + V_{\theta j} B_\phi - \frac{1}{n_j e_j} \frac{\partial p_j}{\partial r} \right]
\]  

(2)

Using Eq. (2) to eliminate the toroidal velocities for both species, Eq. (1) may be rewritten

\[
V_{r j} = \frac{m_j (v_{jk} + v_{dj}) T_j}{(e_j B_\theta)^2} \left[ \frac{1}{p_j} \frac{\partial p_j}{\partial r} \right] - \frac{e_j v_{jk}}{e_k} \left[ \frac{1}{p_k} \frac{\partial p_k}{\partial r} \right] + V_{r j}^{\text{pinch}}
\]  

(3)

where

\[
V_{r j}^{\text{pinch}} = \frac{1}{e_j B_\theta} \left[ - \left( e_j E_\phi^A + \frac{M_{\phi j}}{n_j} \right) + \frac{m_j v_{dj}}{B_\theta} \left( E_r + V_{\theta j} B_\phi \right) + \frac{m_j v_{jk} B_\phi}{B_\theta} \left( V_{\theta j} - V_{\theta k} \right) \right]
\]  

(4)
Derivation (continued 2)

This equation can be simplified to the form

\[ V_{rj} \approx - D_j \left( \frac{1}{p_j} \frac{\partial p_j}{\partial r} \right) + V_{rj}^{pinch} \quad (5) \]

Eq. (5) can be used to infer the experimental particle diffusion coefficient

\[ D_{j}^{exp} = \left( V_{rj} - V_{rj}^{pinch} \right) \frac{L_{nj}^{exp} L_{Tj}^{exp}}{\left( L_{nj}^{exp} + L_{Tj}^{exp} \right)} \quad (6) \]

once \( V_{rj}^{pinch} \) and \( V_{dj} \) are determined
Determination of Momentum Transfer Frequency

Solving Eqs. (1) for the momentum transfer frequencies for the two plasma species, $j$ and $k$, using a perturbation analysis in the assumed small parameter $\left(1 - V_{\phi_j}/V_{\phi_k}\right)$, yields to leading order

\[ \nu_{dj} = \frac{(n_j e_j E_\phi^A + n_j e_j B_\theta V_{\phi_j} + M_{\phi_j}) + (n_k e_k E_\phi^A + n_k e_k B_\theta V_{\phi_k} + M_{\phi_k})}{(n_j m_j + n_k m_k)V_{\phi_k}^\text{exp}} \]  (A.1)

and

\[ (V_{\phi_j} - V_{\phi_k})_0 = \frac{\left( n_j e_j E_\phi^A + n_j e_j B_\theta V_{\phi_j} + M_{\phi_j} \right) - n_j m_j \nu_{dj} V_{\phi_k}^\text{exp}}{n_j m_j \left( \nu_{jk} + \nu_{dj} \right)} \]  (A.2)

and to first order

\[ \nu_{dk} = \frac{(n_k e_k E_\phi^A + n_k e_k B_\theta V_{\phi_k} + M_{\phi_k}) + n_k m_k \nu_{kj} \left( V_{\phi_j} - V_{\phi_k} \right)_0}{n_k m_k V_{\phi_k}^\text{exp}} \]  (A.3)
Experimental Data for Shot 98889

- Electron density ($10^{19}/m^3$)
- Neutral D density ($10^{15}/m^3$)
- Electron temp (keV)
- Ion temp (keV)

Rotation Velocities (m/s), Electric Field (V/m)

Normalized Radius, $\rho$

- $V_{\phi}^C$, exp
- $V_{\phi}^{CO}$, exp
- $E_{rad}$, exp
Inferred Momentum Transfer Frequencies and Pinch Velocity for 98889
Deuterium Diffusion Coefficient for 98889

\[
D_{\text{exp}} = (V_r - V_{\text{pinch}}) L_n \frac{L_T}{(L_n + L_T)}
\]

\[
D_{\text{exp}} = (V_r - V_{\text{pinch}}) L_n
\]

\[
D_{\text{exp}} = V_L L_n \quad \text{(REF)}
\]
Experimental Data for 119436

- Electron density ($10^{19}/m^3$)
- Neutral D density ($10^{17}/m^3$)
- Electron temp (keV)
- Ion temp (keV)

Rotation Velocities (m/s), Electric Field (V/m)

Normalized Radius, rho
Pinch Velocity for 119436

Alternative form

\[ V_{rj}^{\text{pinch}} = \frac{1}{e_j B_\theta} \left[ -\left( e_j E_\phi^A + \frac{M_{\phi j}}{n_j} \right) + \frac{m_j (v_{jk} + v_{dj})}{B_\theta} (E_r + V_{\theta j} B_\phi) - m_j v_{jk} V_{\phi k} \right] \]  \tag{7}
When Eq. (2) was used in Eq. (1) to obtain Eqs. (4) and (5), an explicit expression was also obtained for the diffusion coefficient in Eq. (5)

\[
D_{\text{exp}} = \frac{m_j \left( \nu_{jk} + \nu_{dj} \right) T_j}{\left( e_j B_{\theta} \right)^2}
\]  \hspace{1cm} (8)
CONCLUSIONS

• There is a large inward particle pinch in the edge pedestal which can be evaluated from experimental data.

• When this particle pinch is taken into account in interpreting the experimental particle diffusion coefficient, $D_{\text{exp}}$ increases rather than decreases in the edge pedestal.

• Thus, it seems that the ‘particle transport barrier’ inferred from previous interpretations of diffusion coefficients using purely diffusive models were an artifact of neglecting the large inward pinch.
GENERALIZED DIFFUSION THEORY CALCULATIONS OF THE EDGE PEDESTAL DENSITY PROFILE

J-P. Floyd and W. M. Stacey
Georgia Tech
November, 2009
Overview

- A generalized diffusion theory has been derived based on a pinch-diffusion relation for the ion particle flux\(^1\)
- A methodology for interpreting the pinch velocity and the diffusion coefficient from experimental data has been developed\(^2\)
- This work examines the numerical solution of the generalized diffusion equation using the D and \(V_{\text{pinch}}\) inferred from experiment
The toroidal and radial momentum balances

\[ n_j m_j \left[ (v_{jk} + v_{dj}) V_{\phi j} - v_{jk} V_{\phi k} \right] = n_j e_j E_{\phi}^A + n_j e_j B_\theta V_{rj} + M_{\phi j} \]  

(1)

\[ V_{\phi j} = \frac{1}{B_\theta} \left[ E_r + V_{\theta j} B_\phi - \frac{1}{n_j e_j} \frac{\partial p_j}{\partial r} \right] \]  

(2)

Lead\(^1\) to a pinch-diffusion relation

\[ V_{rj} = -\frac{m_j (v_{jk} + v_{dj}) T_j}{(e_j B_\theta)^2} \left[ \left( \frac{1}{p_j} \frac{\partial p_j}{\partial r} \right) - \frac{e_j}{e_k} \frac{v_{jk}}{v_{jk} + v_{dj}} \left( \frac{1}{p_k} \frac{\partial p_k}{\partial r} \right) \right] + V_{rj}^{\text{pinch}} \]  

(3)

\[ V_{rj}^{\text{pinch}} = \frac{1}{e_j B_\theta} \left[ -\left( e_j E_{\phi}^A + \frac{M_{\phi j}}{n_j} \right) + \frac{m_j v_{dj}}{B_\theta} (E_r + v_{\theta j} B_\phi) + \frac{m_j v_{jk} B_\phi}{B_\theta} (v_{\theta j} - v_{\theta k}) \right] \]  

(4)
Diffusion Equation

- Using \((3)\) in the continuity equation

\[ \nabla \cdot \Gamma_{rj} \equiv \nabla \cdot n_j v_j = S_j \quad (5) \]

- yields a diffusion equation\(^1\)

\[ - \frac{\partial}{\partial r} \left( D_{ji} \frac{\partial n_j}{\partial r} \right) - \frac{\partial}{\partial r} \left( D_{ji} \left( \frac{n_j}{T_j} \right) \frac{\partial T_j}{\partial r} \right) + \frac{\partial}{\partial r} \left( n_j v_{pj} \right) = S_j \quad (6) \]
Experimental Determination of Parameters

- The pinch velocity \( \phi \) can be determined from experiment.
- Fitting measured density and temperature profiles produce\(^2\)

\[
D_{j}^{\text{exp}} = \left( V_{rj} - V_{rj}^{\text{pinch}} \right) \frac{L_{nj}^{\text{exp}} L_{Tj}^{\text{exp}}}{L_{nj}^{\text{exp}} + L_{Tj}^{\text{exp}}}
\]  (7)
Numerical Solution

- Equation (6) is numerically integrated

\[ a_{i,i-1} n_{i-1} + a_{i,i} n_i + a_{i,i+1} n_{i+1} = S_i \]  \hspace{1cm} (8)

\[ a_{i,i+1} = \frac{\alpha_{i+1} + \alpha_i}{2\Delta^2} \]  \hspace{1cm} (9)

\[ a_{i,i} = \frac{2\Delta \beta_i - \alpha_{i+1} - 2\alpha_i - \alpha_{i-1}}{2\Delta^2} \]  \hspace{1cm} (10)

\[ a_{i,i-1} = \frac{\alpha_i + \alpha_{i-1} - 2\Delta \beta_{i-1}}{2\Delta^2} \]  \hspace{1cm} (11)

\[ \alpha_i = -D_i \quad \beta_i = \frac{D_i}{L_T} + V_{ri}^{\text{pinch}} \]  \hspace{1cm} (12)

- and solved via Gaussian elimination for the density
Experimental Data - shot 98889

- Normalized Radius, \( \rho \)
- Electron density \( n_e (10^{19}/m^3) \)
- Neutral D density \( n_o (10^{15}/m^3) \)
- Electron temp \( T_e (keV) \)
- Ion temp \( T_i (keV) \)
Calculated Density and Ion Diff. Coeff.
For diffusion theory to be accurate, the interval width must be smaller than the mean free path.

The expression shown here is:

$$\lambda_i = \sqrt{\frac{D_i}{v_{iz} + v_{di}^*}}$$ (13)

The mean free path is plotted here.
Interval Size Sensitivity

Below is a plot showing the change in the calculated density as interval width is decreased from near the minimum mean free path.
Boundary Conditions

- A global particle balance including a neutral recycling calculation was performed to obtain the ion particle flux crossing the separatrix, which was used as a boundary condition.

- The neutral recycling calculation is not exactly certain, and the results are shown for the increase of the neutral recycling flux by 50% and 100% below, decreasing the discrepancy with experiment.
Modified Neutral Recycling Flux
Conclusions

- This work is currently ongoing

- Continued investigations of the numerical solution of the generalized diffusion equations will be carried out

- Continued investigations of effects of uncertainties on solution will be carried out
References


SABR
SUBCRITICAL ADVANCED BURNER REACTOR

W. M. STACEY
Georgia Tech
APS-DPP Atlanta
November, 2009
MOTIVATION FOR TOKAMAK FUSION-FISSION HYBRID TRANSMUTATION REACTOR

• The growing stockpile of discharged LWR fuel is an impediment to the world-wide expansion of carbon free nuclear power.

• Separating and burning the actinides would dramatically reduce the required number and lifetime of geological HLW repositories for spent nuclear fuel.

• Pure fission burner reactors and separations facilities could be deployed for this purpose over about 2030-2080, but studies indicate some reactors should be subcritical to achieve deep actinide burnup.

• A large worldwide R&D effort is supporting the leading tokamak magnetic fusion concept, and ITER will demonstrate the fusion physics and technology needed for a hybrid over 2018-2030, although high reliability, steady-state operation must be achieved.

• A series of studies has been performed at Georgia Tech to evaluate combining ITER fusion physics and technology with the leading Na-cooled, metal-fuel fast burner reactor technology to design a fusion-fission hybrid burner (transmutation) reactor.
SUB-CRITICAL ADVANCED BURNER REACTOR (SABR)

- **ANNULAR FAST REACTOR (3000 MWth)**
  1. Fuel—TRU from spent nuclear fuel. TRU-Zr metal being developed by ANL.
  2. Sodium cooled, loop-type fast reactor.
  3. Based on fast reactor designs being developed by ANL in DoE Nuclear Program.

- **TOKAMAK D-T FUSION NEUTRON SOURCE (200-500 MWth)**
  1. Based on ITER plasma physics and fusion technology.
  2. Tritium self-sufficient (Li$_4$SiO$_4$).
  3. Sodium cooled.
**FUEL**

**Axial View of Fuel Pin**

Composition 40Zr-10Am-10Np-40Pu (w/o)
(Under development at ANL)

**Design Parameters of Fuel Pin and Assembly**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Length rods (m)</td>
<td>3.2</td>
</tr>
<tr>
<td>Length of fuel material (m)</td>
<td>2</td>
</tr>
<tr>
<td>Length of plenum (m)</td>
<td>1</td>
</tr>
<tr>
<td>Length of reflector (m)</td>
<td>0.2</td>
</tr>
<tr>
<td>Radius of fuel material (mm)</td>
<td>2</td>
</tr>
<tr>
<td>Thickness of clad (mm)</td>
<td>0.5</td>
</tr>
<tr>
<td>Thickness of Na gap (mm)</td>
<td>0.83</td>
</tr>
<tr>
<td>Thickness of LiNbO$_3$ (mm)</td>
<td>0.3</td>
</tr>
<tr>
<td>Radius Rod w/clad (mm)</td>
<td>3.63</td>
</tr>
<tr>
<td>Mass of fuel material per rod (g)</td>
<td>241</td>
</tr>
<tr>
<td>Volume Plenum / Volume$_{fm}$</td>
<td>1</td>
</tr>
</tbody>
</table>

**Cross-Sectional View Fuel Assembly**
4-BATCH FUEL CYCLE

• Fuel cycle constrained by 200 dpa clad radiation damage lifetime. 4 (700 fdp) burn cycles per residence
• OUT-to-IN fuel shuffling
• BOL $k_{\text{eff}} = 0.972$, $P_{\text{ fus }} = 75\text{MW}$, 32 MT TRU
• BOC $k_{\text{eff}} = 0.894$, $P_{\text{ fus }} = 240\text{MW}$, 29 MT TRU
• EOC $k_{\text{eff}} = 0.868$, $P_{\text{ fus }} = 370\text{MW}$, 27 MT TRU
• 24% TRU burnup per 4-batch residence, >90% with repeated recycling
• 1.05 MT TRU/FPY fissioned
• Supports 3.0 1000 MWe LWRs (0.25 MT TRU/yr) at 76% availability during operation (2 mo refueling).

SABR TRU FUEL COMPOSITION (w/o)

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Fresh Fuel</th>
<th>To Re-Process</th>
<th>Core Av EOC/BOC</th>
</tr>
</thead>
<tbody>
<tr>
<td>Np-237</td>
<td>17.0</td>
<td>7.25</td>
<td>9.1/8.3</td>
</tr>
<tr>
<td>Pu-238</td>
<td>1.4</td>
<td>17.3</td>
<td>14.6/17.3</td>
</tr>
<tr>
<td>Pu-239</td>
<td>38.3</td>
<td>18.3</td>
<td>21.9/20.3</td>
</tr>
<tr>
<td>Pu-240</td>
<td>17.3</td>
<td>29.2</td>
<td>27.2/28.2</td>
</tr>
<tr>
<td>Pu-241</td>
<td>6.5</td>
<td>7.31</td>
<td>5.55/5.55</td>
</tr>
<tr>
<td>Pu-242</td>
<td>2.6</td>
<td>7.45</td>
<td>6.50/6.99</td>
</tr>
<tr>
<td>Am-241</td>
<td>13.63</td>
<td>7.45</td>
<td>8.87/8.35</td>
</tr>
<tr>
<td>Am-242m</td>
<td>0.00</td>
<td>0.84</td>
<td>0.71/0.74</td>
</tr>
<tr>
<td>Am-243</td>
<td>2.8</td>
<td>2.79</td>
<td>2.82/2.85</td>
</tr>
<tr>
<td>Cm-242</td>
<td>0.00</td>
<td>0.59</td>
<td>0.33/0.35</td>
</tr>
<tr>
<td>Cm-243</td>
<td>0.00</td>
<td>0.10</td>
<td>.075/.080</td>
</tr>
<tr>
<td>Cm-244</td>
<td>0.00</td>
<td>2.51</td>
<td>2.01/2.24</td>
</tr>
<tr>
<td>Cm-245</td>
<td>0.00</td>
<td>0.56</td>
<td>0.42/0.49</td>
</tr>
</tbody>
</table>

ANNULAR CORE CONFIGURATION
# Neutron Source Design Parameters

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Nominal SABR</th>
<th>Extended SABR</th>
<th>ITER</th>
<th>Pure Fusion Electric ARIES-AT</th>
</tr>
</thead>
<tbody>
<tr>
<td>Current, I (MA)</td>
<td>8.3</td>
<td>10.0</td>
<td>15.0</td>
<td>13.0</td>
</tr>
<tr>
<td>$P_{\text{fus}}$ (MW)</td>
<td>180</td>
<td>500</td>
<td>400</td>
<td>3000</td>
</tr>
<tr>
<td>Major radius, R (m)</td>
<td>3.75</td>
<td>3.75</td>
<td>6.2</td>
<td>5.2</td>
</tr>
<tr>
<td>Magnetic field, B (T)</td>
<td>5.7</td>
<td>5.7</td>
<td>5.3</td>
<td>5.8</td>
</tr>
<tr>
<td>Confinement $H_{\text{IPB99}(y,2)}$</td>
<td>1.0</td>
<td>1.06</td>
<td>1.0</td>
<td>1.7</td>
</tr>
<tr>
<td>Normalized beta, $\beta_N$</td>
<td>2.0</td>
<td>2.85</td>
<td>1.8</td>
<td>5.4</td>
</tr>
<tr>
<td>Plasma Mult., $Q_p$</td>
<td>3</td>
<td>5</td>
<td>5-10</td>
<td>&gt;30</td>
</tr>
<tr>
<td>H&amp;CD Power, MW</td>
<td>100</td>
<td>100</td>
<td>110</td>
<td></td>
</tr>
<tr>
<td>Neutron $\Gamma_n$ (MW/m²)</td>
<td>0.6</td>
<td>1.8</td>
<td>0.5</td>
<td>4.9</td>
</tr>
<tr>
<td>FW $q_{fw}$ (MW/m²)</td>
<td>0.23</td>
<td>0.65</td>
<td>0.5</td>
<td>1.2</td>
</tr>
<tr>
<td>Availability (%)</td>
<td>76</td>
<td>76</td>
<td>10 ?</td>
<td>&gt;90</td>
</tr>
</tbody>
</table>
ADAPTED ITER NEUTRON SOURCE TECHNOLOGY

- Six 20 MW ITER LHR launchers.
- Adapted ITER FW and divertor for Na and He coolant. Replaced SS with ODS steel. Confirmed heat removal with FLUENT code.
- FW lifetime 6.5 FPY at 200 dpa. Replace every 3rd refueling shutdown.
- Scaled down ITER SC CS and TF magnet designs, maintaining ITER standards.
- Multilayer shield. MCNP and EVENT predict > 30 FPY (40 yrs @ 75% avail) radiation damage lifetime for SC magnets.
Li$_4$SiO$_4$ Tritium Breeding Blanket

15 cm Thick Blanket Around Plasma (Natural LI) and Reactor Core (90% Enriched LI) Achieves TBR = 1.16.

NA-Cooled to Operate in the Temperature Window 420-640 C.

Online Tritium Removal by He Purge Gas System.

Dynamic ERANOS Tritium Inventory Calculations for 700 d Burn Cycle, 60 d Refueling Indicated More Than Adequate Tritium Production.
CONCLUSIONS

- The physics and technology performance parameters of ITER (many of which have been achieved already) will be more than adequate for a fusion neutron source for a FFH transmutation reactor. ITER will be the prototype.

- Additional R&D will be needed to obtain greater component and plasma reliability than demanded of ITER, tritium breeding technology, and a more radiation resistant structural material.

- The physics performance parameters and FW neutron and heat loads for a FFH are significantly less than are required for pure fusion electric power.

- The feasibility of deployment of a tokamak fusion neutron source, based on ITER physics and technology, in a FFH transmutation reactor by about 2040 is compatible with the nuclear power scenario for deploying transmutation reactors over roughly 2030-2080.

- Thus, FFH transmutation reactors (for the fissioning of the transuranics in discharged LWR fuel and hence the reduction of geological waste repository requirements) would seem to be the target of opportunity for fusion to contribute to solving the world’s energy problems starting in the first half of the present century.
R-Z Cross section SABR calculation model

Diagram showing the layout of a subcritical advanced burner reactor with various components labeled, such as First Wall, Four Active Fuel Regions, Structural Material, Divertor Removal Space, Pressure Vessel, Tungsten Shield, and Boron Carbide.
R&D FOR A TOKAMAK FFH NEUTRON SOURCE IS ON THE PATH TO PURE FUSION ELECTRIC 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-- prototype.
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. Remote Maintenance.
6. Fusion Nuclear Technology (tritium breeding, etc.) R&D
7. Advanced Structural Materials (200 dpa) R&D

FURTHER FUSION R&D FOR TOKAMAK ELECTRIC POWER
8. Advanced confinement and pressure limits physics R&D.
9. Advanced DEMO.
Promise of FFH transmutation reactors:
• Longer fuel residence time limited by radiation damage, not criticality.
• Larger TRU fuel fraction, not limited by TRU β.
SUB-CRITICAL ADVANCED BURNER REACTOR (SABR)

ANNULAR FAST REACTOR
1. Fuel—TRU from spent nuclear fuel. TRU-Zr metal being developed by ANL.
2. Sodium cooled, loop-type fast reactor.
   Based on fast reactor designs being developed by ANL in DoE Nuclear Program.

TOKAMAK D-T FUSION NEUTRON SOURCE
1. Based on ITER plasma physics and fusion technology.
2. Tritium self-sufficient (Li$_4$SiO$_4$).
3. Sodium cooled.
Fuel Cycle Analysis of SABR

• Purpose: To find a fuel cycle capable of transmuting greater than 90% of the TRU from LWR’s

• Several reprocessing fuel cycles were analyzed based on a fuel residence time limited by the radiation damage limit to structural materials

• Once through cycle was analyzed to determine the feasibility of a deep burn
SABR Model

- Eranos Model of SABR
  - RZ Geometry
  - 33 Energy Groups ranging from 0.1 eV to 20 MeV
  - Solved using discrete ordinates theory
  - 4 region annular fission core
  - 14 MeV neutron source inside the plasma region

- Reprocessing fuel cycles simulated
  - 100 dpa, 200 dpa, 300 dpa
## SABR Fuel Design

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Mass percent</th>
</tr>
</thead>
<tbody>
<tr>
<td>Np</td>
<td>17.0</td>
</tr>
<tr>
<td>Pu</td>
<td>1.4</td>
</tr>
<tr>
<td>Pu</td>
<td>38.8</td>
</tr>
<tr>
<td>Pu</td>
<td>17.3</td>
</tr>
<tr>
<td>Pu</td>
<td>6.5</td>
</tr>
<tr>
<td>Pu</td>
<td>2.6</td>
</tr>
<tr>
<td>Am</td>
<td>13.6</td>
</tr>
<tr>
<td>Am</td>
<td>2.8</td>
</tr>
</tbody>
</table>

### Isotope Mass Percent

- Np: 17.0%
- Pu: 1.4% (238), 38.8% (239), 17.3% (240), 6.5% (241), 2.6% (242)
- Am: 13.6%, 2.8% (243)

### Components
- **LiNbO3**: t=0.3 mm
- **ODS Clad**: t=0.5 mm
- **Na Gap**: t=0.83 mm

### Fuel Assembly Dimensions

- Diameter: 8.95 cm
- Height: 15.5 cm
SABR Fuel Cycle

• Four fuel cycles analyzed
  – 100 dpa, 350 full power day burn cycle with reprocessing
  – 200 dpa, 700 full power day burn cycle with reprocessing
  – 300 dpa, 1000 full power day burn cycle with reprocessing
  – Once Through Cycle deep burn approximately 90%

• Pyrometallurgical Reprocessing used
  – 99% Recovery Rate of TRU
  – 1% Fission product contamination of reprocessed fuel
SABR Fuel Cycle

- Repository
- Reprocessing Facility
- Fuel Fabrication Facility
- Plasma

# burn cycles fuel has been in reactor at BOC

Fuel Assembly numbers:
1  2  3  4
# Fuel Cycle Performance

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Units</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cycle</td>
<td>200 dpa</td>
</tr>
<tr>
<td>Burn Cycle Length Time</td>
<td>FPD</td>
</tr>
<tr>
<td>4 Batch Residence Time</td>
<td>FPY</td>
</tr>
<tr>
<td>$K_{\text{eff}}$(BOL/BOC/EOC)</td>
<td>0.972/0.894/0.868</td>
</tr>
<tr>
<td>$P_{\text{fus}}$(BOL/BOC/EOC)</td>
<td>MW</td>
</tr>
<tr>
<td>TRU Content (BOC/EOC)</td>
<td>MT</td>
</tr>
<tr>
<td>Power Peaking (BOC/EOC)</td>
<td>1.80/1.98</td>
</tr>
<tr>
<td>TRU Burned per Residence</td>
<td>%</td>
</tr>
<tr>
<td></td>
<td>kg</td>
</tr>
<tr>
<td>TRU Burned per Year</td>
<td>MT/FPY</td>
</tr>
<tr>
<td>LWR Support Ratio</td>
<td>3.54</td>
</tr>
<tr>
<td>Average Core Flux Across Cycle</td>
<td>n/cm²-s</td>
</tr>
<tr>
<td>TRU to Repository</td>
<td>kilograms</td>
</tr>
<tr>
<td></td>
<td>31.4</td>
</tr>
</tbody>
</table>
SABR Power Distribution

- Highly peaked power distribution in the inner fuel regions
- Only fuel shuffling accounted for
- Power peaking less than 2.0
Decay Heat to the Repository

- Long term integral decay heat limiting repository factor
- Reduction of long term decay heat by transmutation of TRU
- Higher burnup → less TRU in repository
Summary

• 76% Availability → 3.0 LWR’s supported by SABR
• Greater than 90% burnup can be achieved in SABR
• TRU to the repository dependent on burnup and separation efficiency in reprocessing method
• Once through cycle is not feasible
## Fuel Cycle Performance

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Units</th>
<th>100 dpa</th>
<th>200 dpa</th>
<th>300 dpa</th>
<th>OTC</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cycle</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Burn Cycle Length Time</td>
<td>FPD</td>
<td>350</td>
<td>700</td>
<td>1000</td>
<td>4550</td>
</tr>
<tr>
<td>4 Batch Residence Time</td>
<td>FPY</td>
<td>3.83</td>
<td>7.67</td>
<td>10.95</td>
<td>49.8</td>
</tr>
<tr>
<td>$K_{\text{eff}}$(BOC/EOC)</td>
<td></td>
<td>0.940/ 0.916</td>
<td>0.894/ 0.868</td>
<td>0.887/ 0.834</td>
<td>0.784/ 0.581</td>
</tr>
<tr>
<td>$P_{\text{fus}}$(BOC/EOC)</td>
<td>MW</td>
<td>155/218</td>
<td>240/370</td>
<td>286/461</td>
<td>1012/1602</td>
</tr>
<tr>
<td>TRU Content (BOC/EOC)</td>
<td>MT</td>
<td>29.6/28.6</td>
<td>29.0/27.1</td>
<td>28.3/25.4</td>
<td>22.6/9.5</td>
</tr>
<tr>
<td>Power Peaking (BOC/EOC)</td>
<td></td>
<td>1.68/1.78</td>
<td>1.80/1.98</td>
<td>1.82/2.04</td>
<td>1.97/1.59</td>
</tr>
<tr>
<td>TRU Burned per Residence</td>
<td>%</td>
<td>16.7%</td>
<td>23.8%</td>
<td>31.6%</td>
<td>87.2%</td>
</tr>
<tr>
<td>TRU Burned per Year</td>
<td>MT/FPY</td>
<td>1.04</td>
<td>1.05</td>
<td>1.05</td>
<td>1.05</td>
</tr>
<tr>
<td>LWR Support Ratio</td>
<td></td>
<td>3.27</td>
<td>3.54</td>
<td>3.62</td>
<td>3.82</td>
</tr>
<tr>
<td>Average Core Flux Across Cycle</td>
<td>n/cm^2-s</td>
<td>4.73E15</td>
<td>4.81E15</td>
<td>5.28E15</td>
<td>1.82E16</td>
</tr>
<tr>
<td>TRU to Repository</td>
<td>kilograms</td>
<td>37.4</td>
<td>31.4</td>
<td>19.7</td>
<td>87.4</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>12</td>
</tr>
</tbody>
</table>
SABR Power Distribution

- Highly peaked power distribution in the inner fuel regions
- Longer residence time in reprocessing cycles higher power peaking
- OTC EOC lower power peaking
A Safety and Dynamics Analysis of SABR

T. S. Sumner, W. M. Stacey and S. M. Ghiaasiaan
Georgia Tech
November, 2009
SUB-CRITICAL ADVANCED BURNER REACTOR (SABR)

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1. Fuel—TRU from spent nuclear fuel. TRU-Zr metal being developed by ANL.
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   Based on fast reactor designs being developed by ANL in DoE Nuclear Program.

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1. Based on ITER plasma physics and fusion technology.
2. Tritium self-sufficient (Li$_4$SiO$_4$).
3. Sodium cooled.
Safety and Dynamics Analysis of SABR

• Purpose: To find the transient response of the subcritical, sodium-cooled, TRU-fueled, fast transmutation reactor design concept.

• Several initiating accidents were simulated to determine the magnitude of each accident that could be tolerated without resulting in permanent damage to the fission core.

• How much time would be available to take corrective measures to prevent damage to the reactor?
Thermal Hydraulics Model

• Coupled fission core power level with the thermal hydraulics of the thermal heat removal system using RELAP5-3D.

• Fusion neutron source accounted for by the source term in the point kinetics equations.

• ATHENA version of RELAP5-3D allows for liquid metal coolants.
Thermal Hydraulics Model, continued…

- Accidents simulated:
  - Loss of Power Accident (LOPA),
  - Loss of Flow Accident (LOFA),
  - Loss of Heat Sink Accident (LOHSA), and
  - Accidental Increase in Fusion Neutron Source Strength.
- Coolant Boiling Temperature: 1,156 K
- Fuel Melting Temperature: 1,473 K
LOPA

- A Loss of Power Accident results in zero power to SABR’s auxiliary systems.
  - All sodium coolant pumps shut off,
  - Heating power to the plasma is lost
LOFA

- A Loss of Flow Accident results in one or more of the primary coolant loop pumps failing. (neutron source remains on)
LOHSA

- A Loss of Heat Sink Accident was modeled as the failure of the intermediate coolant loop EM pumps. (Neutron source remains on)
Accidental Increase in Fusion Neutron Source Strength

- What is the maximum allowable fusion neutron source strength increase that could be tolerated before either coolant boiling or fuel melting occurs?

<table>
<thead>
<tr>
<th></th>
<th>BOL</th>
<th>BOC</th>
<th>EOC</th>
</tr>
</thead>
<tbody>
<tr>
<td>Steady-state Fusion Power Level (MW)</td>
<td>73</td>
<td>242</td>
<td>370</td>
</tr>
<tr>
<td>Fusion Power Leading to Coolant Boiling (MW)</td>
<td>121.2</td>
<td>398.1</td>
<td>563.9</td>
</tr>
<tr>
<td>Maximum Tolerable Increase in Fusion Power (MW)</td>
<td>48</td>
<td>156</td>
<td>194</td>
</tr>
<tr>
<td>Maximum Fuel Temperature (K)</td>
<td>1265</td>
<td>1336</td>
<td>1337</td>
</tr>
</tbody>
</table>

(Note: Coolant boiling occurs at 1,156 K and fuel melting occurs at 1,473 K)

- Above fusion neutron source strength increases resulting in coolant boiling would surpass the Troyon Beta Limit and terminate the source excursion.
Summary

• Fission core power can be reduced to decay heat levels in a couple of seconds without source neutrons
• SABR is safe against LOPA
• SABR can tolerate up to 50% LOFA and LOHSA without sustaining permanent core damage
• Accidental increases in neutron source strength could be limited by operation near inherent density and beta limits.
The “natural” coordinate systems for tokamak plasma physics computations is the set of nested magnetic flux surfaces. In general applications, these flux surface coordinate system must be determined by numerically solving the Grad-Shafranov equations. However, an analytical solution (the Miller equilibrium) for the flux surface coordinates has been developed in terms of the elongation, triangularity and major radius of the displaced centers of the flux surfaces. The metrics for such a representation have been worked out, and the plasma fluid equations have been explicitly represented on this flux surface coordinate system, thus eliminating the need for a numerical solution to the Grad-Shafranov equation prior to solution of the transport equations in flux surface geometry.