Solving the Spent Nuclear Fuel Problem by Fissioning Transuranics in Subcritical Advanced Burner Reactors Driven by Tokamak Fusion Neutron Sources

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Abstract. The Georgia Tech concept of the Subcritical Advanced Burner Reactor spent nuclear fuel transmutation reactor and supporting analyses to date are summarized. SABR is based on the fast reactor physics and technology prototyped in EBR-II and proposed for the Integral Fast Reactor and the PRISM Reactor, and on the tokamak fusion neutron source physics and technology that will be prototyped in ITER. Preliminary fuel cycle calculations indicate that subcritical operation would enable a proliferation-resistant, up to 100% TRU aggregate, fuel reprocessing cycle, and that introduction of SABRs in a 1-to-3 power ratio with LWRs would reduce the required spent nuclear fuel High-Level-Waste-Repository capacity by a factor of 10 to 100. Preliminary dynamic safety calculations indicate that SABRs could be shut down to the decay heat level by turning off the plasma heating power without core damage in Loss of Heat Sink, Loss of Flow and Loss of Power accidents, but that additional decay heat removal capability is needed.in the case of total loss of primary or secondary system pumping power.

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I. INTRODUCTION

The group at Georgia Tech has worked over the past two decades to identify a practical, near-term concept for subcritical transmutation reactors that could be fueled 100% with the transuranics (TRU) from spent nuclear fuel, using a D-T fusion neutron source based on the plasma physics and technology that will be prototyped by the operation of ITER[1] in 2025-35. A D-T plasma producing the ITER design objective 500 MWth of fusion power will produce $S_{fus}^{500} = 2.1x10^{20}$ 14-MeV neutrons per second. If this fusion neutron source is distributed over a large surface area and surrounded by fissionable material in an assembly with neutron multiplication constant k and number of neutrons per fission ν, the total neutron transmutation (fission) rate will be[2] $TR_{fis} = kS_{fus}/(1-k)\nu$ fis/s, provided $k<1$. Thus, a constant transmutation (power production) rate can be maintained as k varies with fuel burnup by varying the fusion neutron source level.

The characteristics and performance capability of nuclear facilities driven by a tokamak fusion neutron source operating with essentially ITER-level physics and technology (but improved availability) have been characterized[2-6] and design concepts have been developed for various nuclear applications (transmutation of weapons-grade plutonium[7], tritium production[8], transmutation of transuranics (TRU) in spent nuclear fuel[9-18] and breeding of fissile plutonium[19]) operating with such tokamak D-T fusion neutron sources. Preliminary dynamic analyses of fuel cycle performance[20-27] and of safety[28-30] of the subcritical TRU transmutation reactors have been performed, and the work has been summarized at earlier stages in Refs 31-34.

The world's electrical energy supplies must grow substantially over the near term to keep pace with the world's population growth and to extend the benefits of plentiful electrical power to the developing and undeveloped regions of the world. As the dream of large-scale deployment of reliable baseline solar and wind power is slowly being confronted by the reality that these sources are environmentally problematical in their own ways and inherently intermittent in nature, with only niche practical applications[35], the realization is slowly sinking in that expanded use of fossil fuels and/or nuclear energy are the only realistic possibilities for maintaining our standard of living and extending it to the under-developed regions of the world. Recognition of the adverse climatic effects of burning of fossil fuels[35] is becoming widely accepted, which leaves nuclear power as the only practical environmentally benign option for meeting the world's growing requirements for electric power in the first half of this century[35].

However, nuclear power has an unresolved problem, one that fusion can help solve—the disposal of spent nuclear fuel containing radioactive TRU elements with extremely long half-lives of 100,000 + years. While disposal of this spent fuel by burial in secured repositories is technically feasible and probably not excessively expensive, this solution has been put on hold in the US (at least temporarily) for political reasons. The burial solution also wastes the substantial energy resource of the transuranics in the spent fuel. A better, but technically more difficult, solution is to separate the long-lived transuranics in spent nuclear fuel, which are fissionable in fast reactors, and use them as TRU fuel in special purpose fast "burner" or "transmutation" reactors, thus extracting additional energy from the uranium fuel resource while destroying the long half-life radioactive material.

There are technical reasons why such transmutation reactors would work better if operated subcritical with a neutron source rather than operated critical. (In a critical reactor the neutron fission chain reaction is maintained entirely by the neutrons produced in fission, while in a subcritical reactor the neutrons produced by fission must be supplemented by source neutrons in order to maintain the neutron fission chain reaction.) One advantage of subcritical operation is that the neutron source strength can be altered to maintain the neutron fission chain reaction (power) level as the fissionable material is destroyed (or increased), allowing a longer fuel residence time in the reactor and more transuranic destruction before reprocessing. Another advantage of subcritical operation is that the margin of reactivity error to a prompt supercritical power excursion is much larger in a subcritical reactor than in a critical reactor where it is related to the small fraction of delayed fission neutrons which are not emitted instantaneously. Since this delayed neutron fraction is much smaller for the transuranics $(\beta \simeq 0.002)$ than for uranium $(\beta \simeq 0.006)$, prudence dictates that only a fraction (about 20%) of the fuel in a critical reactor be transuranics. The much larger margin of reactivity error with subcritical operation $(\delta k_{sub} > 0.03)$ would allow the subcritical transmutation reactor to be completely fueled with transuranics, resulting in $\simeq 5$ times fewer subcritical than critical transmutation reactors being needed to "burn" a given amount of transuranics. Finally, the much smaller sensitivity to feedback reactivity of subcritical reactors makes the understanding of their response to various malfunctions more straightforward.

Substantial investigations of fusion driven subcritical reactors for the breeding of plutonium, the fissioning of transuranics in spent nuclear fuel and other purposes are being carried out in the Russian[36,37] and Chinese[38] nuclear programs.

It would appear that the transmutation (destruction by fission) of TRU in spent nuclear fuel in subcritical fast reactors driven by ITER-level fusion neutron sources provides both the best option for solving the spent nuclear fuel problem and the most promising opportunity for fusion to contribute to nuclear energy in the first half of the present century. Such a SABR program would seem to be overdue in the USA.

II. THE SABR TRU TRANSMUTATION REACTOR DESIGN CONCEPT

There has been a substantial technical investigation[9-12,16-18,20,22-33] of subcritical transmutation reactors based on tokamak fusion and sodium-cooled/metal-fuel fast reactor technologies. The reason that these technologies were chosen is that they are the most highly developed fusion and fission transmutation-applicable technologies, about which we know enough to make a realistic assessment of something that could be built in the next 25-30 years. The Subcritical Advanced Burner Reactor (SABR)[16,18] is based on ITER[1] fusion technology and physics, so in a sense ITER will be the prototype for the fusion neutron source for SABR.

The proliferation-resistant fission reactor physics and technology for SABR is based on the Integral Fast Reactor (IFR)[39,40] and the GE PRISM[41] designs, so the successful operation of EBR-II and its associated pyro-processing system[42-44] were the prototype for the fission system. A fast spectrum reactor was chosen to take advantage of the fact that the ratio of the fission to capture cross sections for all the transuranics increases with increasing neutron energy (the destruction of transuranics is via fission, not capture), and the metal fuel was chosen because it results in a harder neutron spectrum and thus a greater transuranic fission rate than oxide or carbide fuels. The IFR-type pyrolytic fuel cycle process envisioned for SABR has the property that all the transuranic elements are processed as an aggregate, and plutonium is never separated from the less reactive transuranics, some of which have significant spontaneous fission rates, which is a positive feature for non-proliferation.

It has been calculated that the ITER[1] tokamak magnetic and plasma support technology configuration, with a slightly smaller plasma operating with somewhat lesser performance parameters but with higher availability, could provide an adequate D-T fusion neutron source to maintain a 3000 MWth annular fast burner reactor surrounding the plasma[4,16]. The general configuration of the SABR fission reactor and fusion neutron source is described in Fig.1. In the first version[16] (SABR1) the fuel was located in four annular rings (0.62 m total radial thickness) of vertical fuel assemblies (in the region labeled sodium pools in Fig. 1) surrounding the toroidal plasma chamber on the outboard side, cooled by sodium loops. Calculations indicated that SABR1, operating for a period of 2800 full power days in a 4-batch fuel cycle, would accumulate 200 dpa (the design limit on the ODS steel fuel cladding) in the discharged fuel cladding.

In the more recent SABR2 design[18] (adapted to more readily accommodate refueling) the dimensions of which are shown in Fig. 1 and Table 1, the reactor core consists of 10 modular sodium pools (Fig. 2) containing essentially the same fuel assemblies as in SABR1 surrounding the toroidal plasma chamber on the outboard side.

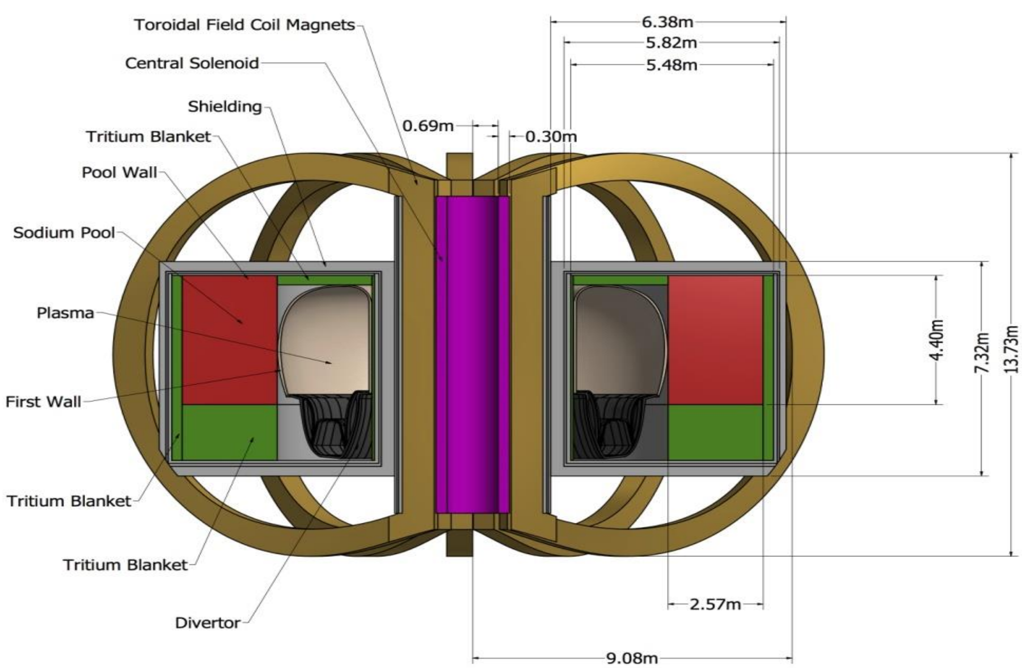


Figure 1 SABR2 configuration. (reproduced from Ref 18, with permission)

Table 1 SABR2 plasma physic parameters

Plasma	
Major radius	4.0m
Plasma radius	1.2m
Elongation	1.5
Toroidal magnetic field (on axis)	5.6T
Plasma current	10 MA
Inductive current startup	6.0 MA
Non-inductive current drive	4.5 MA
Bootstrap current fraction	0.55
Heating & current-drive power	110 MW (70 EC, 40LH)
Confinement factor H_{98}	1.2
Normalized β_N	3.2%
Safety factor at 95% flux surface	3.0
Max. and BOL fusion power	<500 MW and 233 MW
Max. fusion neutron source strength	1.8×10^{20} n/s
Fusion gain ($Q_p = P_{fusion}/P_{extheat}$)	2.1-4.5

The ITER magnetic, first-wall and divertor systems (the latter two converted to Na coolant) were used with minimal alteration, and the heating-current drive system was adapted from ITER[1].

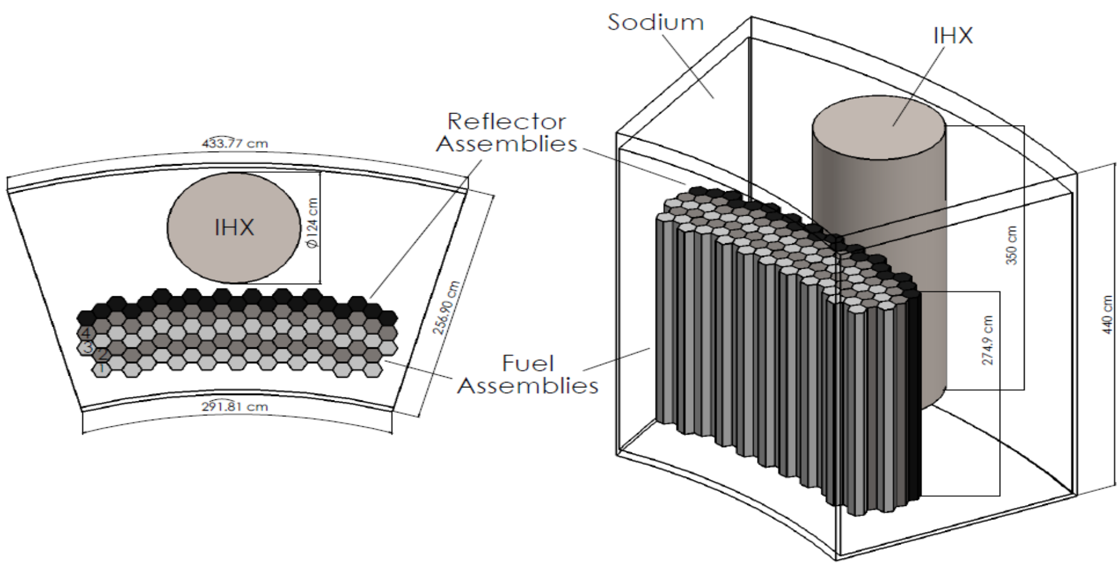


Figure 2. SABR2 modular sodium pool with reactor and intermediate heat exchanger. (reproduced from Ref 18, with permission)

The TRU-Zr fuel is clad with ODS steel in 0.54 cm OD fuel pins (Fig. 3), 469 of which are contained in each of the 80 fuel assemblies (Fig. 4), 13.9 cm across flats, in each of the 10 Na-pools. Each pool also contains an intermediate heat exchanger, as depicted in Fig. 2. A 3 mm thick SiC flow channel insert is placed within each assembly to prevent current loops connecting through the duct wall, which would increase the MHD pressure drop.

The fuel pin design is depicted in Fig. 3. The pin is about 2 m in height, with the TRU fuel in the lower third and a fission gas plenum in the upper two-thirds.

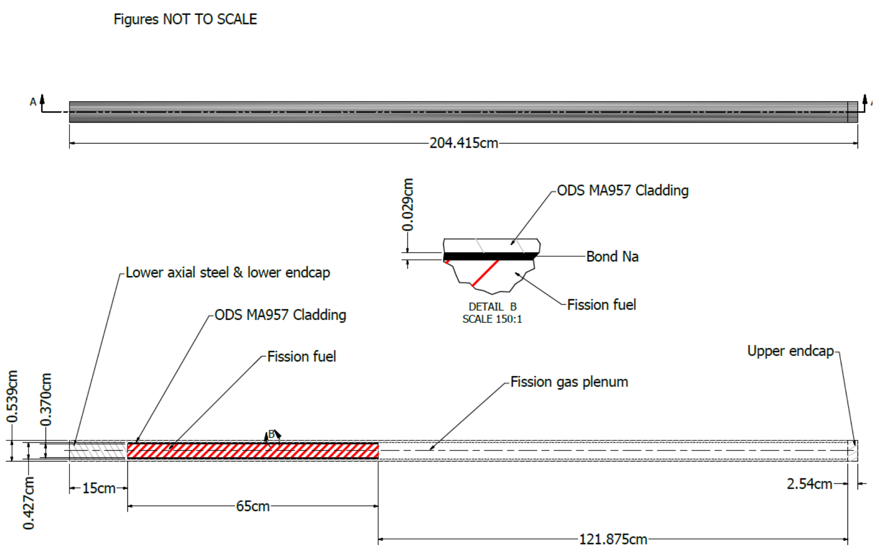


Figure 3. TRU fuel pin configuration. (reproduced from Ref 18, with permission)

The fuel assembly consists of 469 of these pins, arranged as indicated in Fig. 4. Note the SiC liner separating the fuels pins and sodium within the assembly from the ODS steel duct in order to prevent current loops that would cause MHD pressure drops.

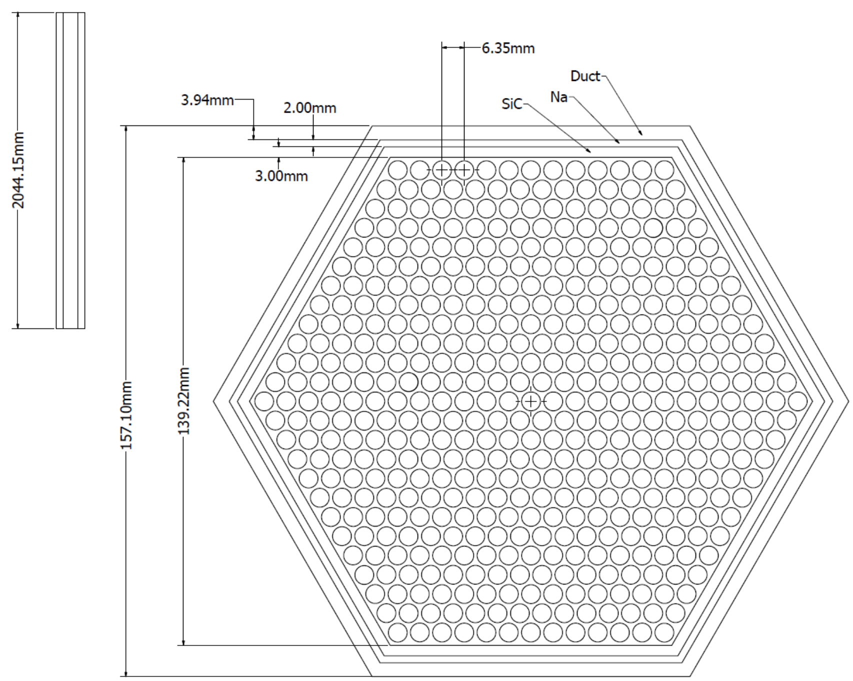


Figure 4. TRU fuel assembly configuration. (reproduced from Ref 18, with permission)

Fuel assembly calculations were made in 1968 group P_1 transport theory to homogenize the fuel assemblies for a 2D, S_8, 33-group ERANOS[45] neutronics calculation, and RELAP5[46] thermal-hydraulic calculations were made to obtain the parameters given in Table 2.

Table 2 SABR2 modular sodium pool parameters

Sodium Pool	
Number of modular pools	10
Mass of fuel per pool	1510.4 kg
Mass of Na per pool	22,067 kg
Power per pool	300 MWth
Power Peaking	1.27
Mass flow rate per pool	1669 kg/s
Number of pumps per pool	2
Pumping power per pool (EM pumps)	20 MW
Core Inlet/Outlet temperatures	628 K/769 K
Fuel Max Temp/Max Allowable Temp	1014 K/1200 K
Clab Max Temp/Max Allowable Temp	814 K/973 K
Coolant Max Temp/Max Allowable Temp	787 K/1156K

Refueling (removal and replacement) of the fuel assemblies located in Na pools within the TF coil configuration is a challenging design issue that was addressed for SABR2 as illustrated in Fig. 5. Transport casks for removal of an individual Na pool to a hot bay are located between TF coils on the outboard at the locations of pools 1 and 6 in Fig. 5. First, the pools in these locations are removed radially, then the other pools are individually rotated to the nearest port and removed. The secondary heat removal system must be disconnected for the individual pool during its removal. The thermal capacity of the pool must absorb the decay heat during the removal to prevent damage, which places an upper limit on the allowed removal time, or an alternative way to handle decay heat during refueling must be designed. We estimate the procedure will work for 1 h removal time and decay heat equal to 7% of operating power, possibly requiring overcooling of the Na in the pool prior to disconnect from the external heat removal system.

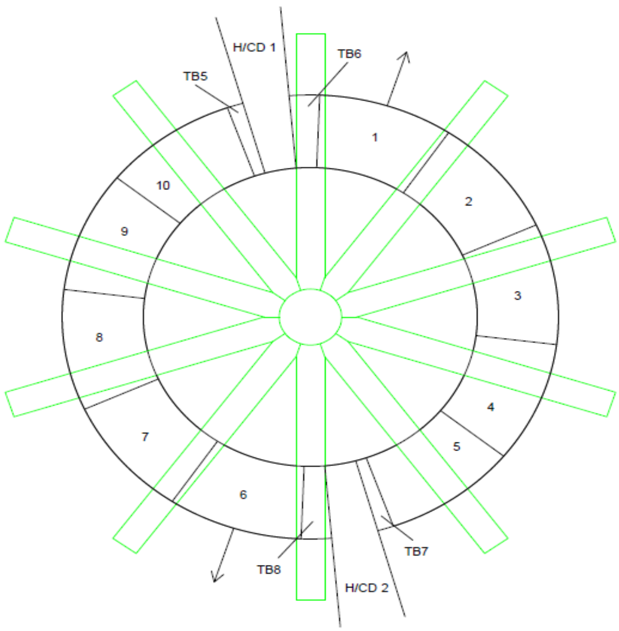


Figure 5. Removal of Na-pools from modular pool configuration. (reproduced from Ref 18, with permission)

III. SABR1 FUEL CYCLE ANALYSIS

Several fuel cycles based on pyroprocessing the fuel removed from SABR1 to separate the remaining transuranics in aggregate from the fission products and recycling of the TRU have been investigated[22-27]. The maximum fuel residence time in the reactor is limited to 2800 days = (4cyclesX700 days/cycle) by the 200 dpa radiation damage limit on the fuel cladding. A 4-batch fuel cycle is indicated in Fig. 6. The maximum effective multiplication constant was $k_{eff}^{BOL} \simeq 0.97$ at BOL and the maximum fusion power required to maintain the 3000MWth fission rate was < 500 MWth.

A SABR1, based on the sodium cooled, metal fuel technology developed at ANL[36.37] and proposed in the ANL IFR[40] and the GE PRISM[41] reactors, operating at 75% availability with a 4-batch out-to-in fuel cycle (with total fuel residence time limited by 200 dpa radiation damage in the clad) could destroy annually all the transuranics produced annually by three 3000MWth LWRs[22,23,25,27]. Thus, an equilibrium nuclear fleet could be envisioned in which 75% of the power is produced by advanced versions of the present LWRs and 25% is produced by SABRs burning the transuranics produced in the LWRs.

In an alternative fuel cycle in which the LWRs are phased out in favor of critical fast reactors, the Pu could be separated from the transuranics in spent fuel and used to fuel critical fast reactors, while the remaining “minor actinide” transuranics were used to fuel SABRs. One 3000MWth SABR could destroy annually all the minor actinides produced annually in 25 3000MWth LWRs[27].

With such SABR fleets, the relatively short-lived fission products (most with less than a few hundred year half-life), the few longer-lived fission products and trace amounts of transuranics would still need to be buried in secure repositories, but an order of magnitude fewer of them would be needed than for the direct burial of LWR spent fuel.

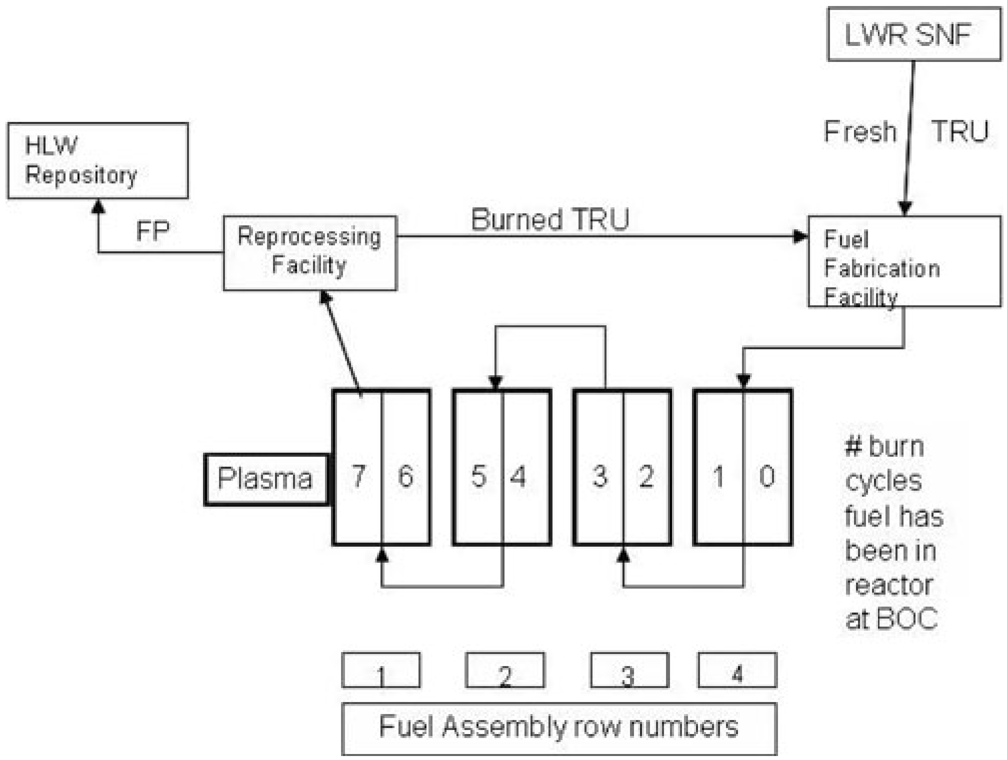


Figure 6 The SABR1 4-batch out-to-in fuel cycle. (reproduced from Ref 23, with permission)

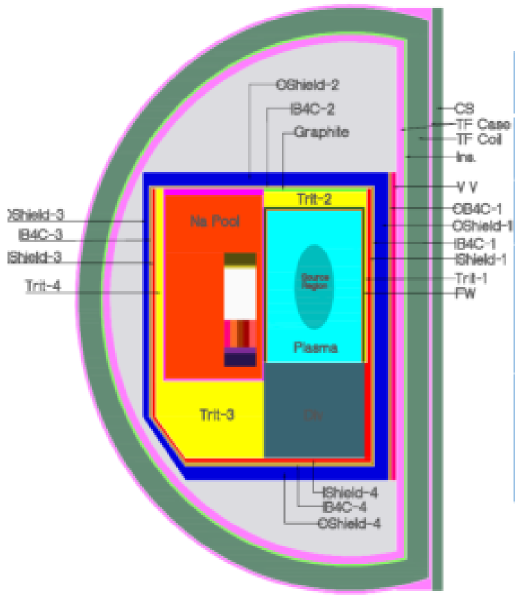
IV. TRITIUM SELF-SUFFICIENCY

In SABR2, modular sodium-cooled Li_4SiO_4 blankets are located i) above the plasma, ii) below the sodium pools, iii) outboard of the sodium pools and iv) in two locations 180^o apart in the ring of sodium pools shown in Fig. 5 (TB5, TB6, TB7, TB8). This tritium must diffuse out of micron grains and migrate through the blanket to helium purge channels, which requires a blanket temperature in the range 325 C < T < 925 C. Thermal-hydraulics calculations indicate that the nuclear heating can readily be removed to maintain temperatures in the blanket within this temperature window. Neutron transport calculations (R-Z, S_8, 33-grp) indicate that this configuration produces an average Tritium Breeding Ratio TBR = 1.12. A time-dependent calculation of the tritium inventory in the Li_4SiO_4 blankets, the tritium processing system, the tritium storage system and plasma demonstrated tritium self-sufficiency for an operational cycle based on one year of burn at 75% availability, followed by 90 days of downtime for the refueling operation[18,47]. SABR produces and consumes about 15 kg/yr of tritium.

V. SHIELDING

The SABR2 shield design[18] is indicated in Fig. 7. A 2D, R-Z MCNP-B Monte Carlo calculation confirmed that this shield design reduced the radiation damage to the TF Coils below the design limits shown in Table 3 for a 40 yr operational lifetime at 75% availability.

SHIELDING (SABR MEETS 30FPY DESIGN OBJECTIVE)



Radiation Limit	Value	Units
Insulation Dosage	1×10^{9}	rads
Nuclear Heating Rate	1.0 – 5.0	mW/cc
Neutron fluence (>0.1MeV)	1.0×10^{19}	n/cm^2
Copper Stabilizer Resistivity	1.2	nΩm

Figure 7: SABR2 Shield Design (reproduced from Ref 18, with permission)

Table 3: SABR2 Shield Design Limits to TF Coils

VI. DYNAMIC SAFETY ANALYSIS

A. Feedback Control of Power Excursions

The SABR must be designed to prevent or suppress any dynamic power surges that may inadvertently occur in the plasma neutron source or in the modular fission cores. The actions of various negative and positive reactivity feedback mechanisms for the fission cores are well-known, and in fact it was demonstrated in EBR-II that metal-fuel, sodium pool technology could be designed to be inherently safe (i.e. negative feedback mechanisms shut the reactor down without damage when pumps in the sodium pool and in the external heat removal system were intentionally shut down to increase the fuel temperatures[42-45]). However, feedback mechanisms are much less effective in subcritical than in critical reactors.

B. Dynamic Safety Analysis of SABR1

Earlier work[28] examined the transient response to loss-of-flow (LOFA), loss-of-heat-sink (LOHSA) and loss-of-power (LOPA) accidents, control rod ejection and neutron source excursion accidents in a single 3000MWth SABR1 core with a sodium-loop cooling system, using a “point” neutron kinetics model of the single SABR1 core and the RELAP5-3D thermal-hydraulics code[46]. It was found that (a) the core power can be reduced to decay heat levels in a few seconds by turning off the neutron source heating power; (b) a LOPA thus reduces the core to decay heat level in a few seconds and natural circulation prevents core damage; (c) natural circulation heat removal is sufficient that undetected LOFAs and LOHSAs up to 50% can be tolerated without core damage, and d) no core damage occurred in detected 100% LOFAs and LOHSAs in which the fusion neutron source power was switched off.

C. Nodal Neutronics and Plasma Source Dynamics Model for SABR2

The earlier loop-type cooling system has now been replaced by a modular sodium pool design in order to facilitate core refueling and perhaps to benefit from some of the inherent safety features demonstrated in EBR-II by such a pool system. We have now constructed[29,30] a coupled-core, or nodal, neutron and coolant dynamics model of the 10 fission cores of SABR2, the fusion neutron source and the associated heat removal systems in order to investigate the degree to which the safety features of the SABR2 modular core design and to investigate if the modular fission core configuration might be subject to spatial power oscillations.

Plasma Source Dynamics

Power excursions in the plasma neutron source, due either to instabilities within the plasma or to the inadvertent turn-on of a modular plasma heating unit or pellet fueling unit or the inadvertent opening of a gas fueling valve, etc. are a concern because they would produce power excursions in the fission cores. We have recently initiated an investigation of plasma burn control mechanisms that could limit unanticipated D-T plasma power excursions. We are motivated by the observation that the edge plasma parameters have a strong impact on the core plasma parameters to search for possible burn control mechanisms in the more accessible plasma edge. Experimental and theoretical observations encourage us that it may be possible to use injection of deuterium or a seeded impurity gas in such a way that the plasma would respond to an increase in edge temperature by momentarily dropping into the inferior L-mode confinement regime and thereby to terminate or at least limit any plasma power excursion.

We are presently assembling a dynamic plasma-impurities-neutrals edge transport code coupled to a global plasma dynamics code, a wall recycling model and a 2-pt divertor model for the purpose of investigating the ability of modulated gas puffing (D) or impurity seeding (Ne, Ar, Xe) to create edge conditions in which an increase in edge temperature causes a momentary H-L transition to suppress the power excursion. For example, impurities seeded into a plasma edge that was somewhat cooler than the temperature for which the impurity radiation is maximum would respond to an increase in temperature with an increase in radiative power, hence a decrease in non-radiative power across the separatrix below the threshold $P_{\min}^{H-L}$, causing the plasma to drop momentarily into L-mode in response to a positive edge temperature excursion and thus serving as a burn control feedback mechanism. Another negative feedback possibility that will be investigated is that an increase in edge temperature would produce an increase in ion orbit loss of energy that would terminate a power excursion.

Nodal Dynamics Model for SABR2

We have developed a coupled nodal neutron dynamics and thermal-hydraulics code GTDYN[29,30] for the power level and temperatures in the different sodium pools. A node is defined as all of the fuel assemblies and reflector assemblies in a given sodium pool. These nodal kinetics equations have been used to calculate the time-dependent power in each separate core during various accident scenarios such as Loss of Flow Accidents (LOFA) and Loss of Heat Sink Accidents (LOHSA).

The neutron dynamics equations for the neutron density n_j and for the 6 delayed neutron precursor densities $c_{i,j}$ in each of 10 nodes j are

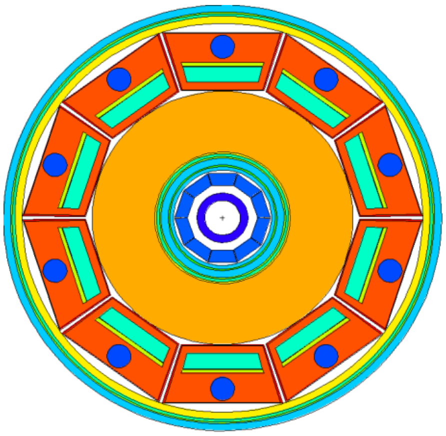
$$\frac{dn_j(t)}{dt} = \frac{(1-\beta_j)}{\Lambda_{fj}} n_j(t) + \sum_{i=1}^{6} \lambda_{i,j} c_{i,j(t)} + \frac{n_j(t)}{\Lambda_{2nj}} + S_{fus,j} + \sum_{k=1}^{10} \frac{\alpha_{k,j} n_k(t)}{l_{e,k}} - \frac{n_j(t)}{l_{e,j}} - \frac{n_j(t)}{l_{a,j}}$$

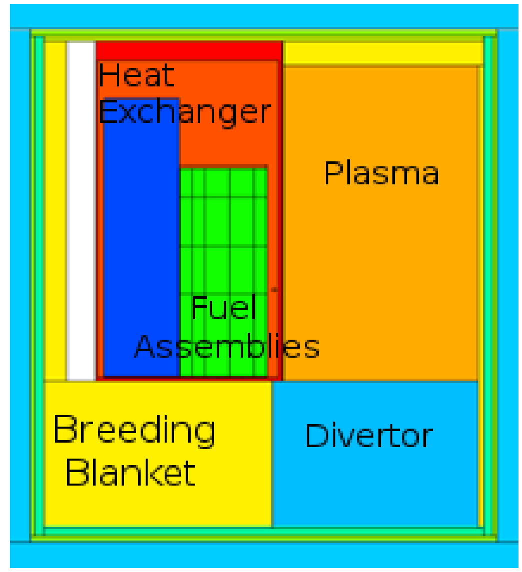
$$\frac{dc_{i,j}}{dt} = \frac{\beta_{i,j}}{\Lambda_{fj}} n_j(t) - \lambda_{i,j} c_{i,j} , \qquad \Lambda_{fj} = \frac{1}{V\upsilon\Sigma_{f,j}} \quad , \quad \Lambda_{2nj} = \frac{1}{V2\Sigma_{n2n,j}}$$

where β_j and $\lambda_{j,i}$ are delayed neutron fraction and precursor decay constant, Λ_{fj} is the fission neutron generation time, Λ_{2nj} is the n-2n generation time, l_a is the absorption lifetime, l_e is the escape lifetime, $S_{fus,j}$ is the rate at which fusion neutrons from the plasma enter node j, and $\alpha_{k,j}$ is the nodal coupling coefficient (i.e. the probability that a neutron leaking from node k will enter node j before entering another node). These kinetics terms are calculated using MCNP6[48] in a 3D geometry shown in Fig.8. We also include in the model a calculation of how these kinetics parameters change during various perturbations such as fuel Doppler broadening, sodium voiding, fuel rod axial expansion, core grid plate expansion, and fuel bowing. The MCNP6 parameters are precomputed and modified by table look-up at each time step.

The thermal hydraulics component of the dynamics model utilizes COMSOL's Pipe Flow Module and Heat Transfer in Solids Module[49]. We model each of the ten pools separately[29,30]. When modeling a pool, we consider the entire primary loop (including the temperature distribution in the fuel, cladding, and primary heat exchanger), and we consider a small part of the secondary loop. In the secondary loop, we only model the flow of sodium through the primary heat exchanger. We place boundary conditions just outside the heat exchanger's secondary side inlet and outlet. We assume the remaining power conversion and heat rejection cycles match that of a typical pool type, sodium cooled fast reactor design.

We use the Pipe Flow Module to create what is essentially a loop-type model of a SABR modular core. The structure of the model is similar to a RELAP5[46] model. It is a loop composed of sodium filled pipes. Various boundary conditions simulate heat transfer from the core, across the heat exchanger, and into the sodium pool. The Pipe Flow Module solves the fluid time-dependent continuity, momentum, and energy equations for the sodium in 1D.





Top down view of sodium pools in MCNP model   Side view of sodium pools in MCNP model

Figure 8 3D Geometry of Nodal Neutron Dynamics Model (reproduced from Ref 29, with permission)

For each core, we model a single characteristic fuel pin and a single characteristic heat exchanger tube using the Heat Transfer in Solids Module. The Heat Transfer in Solids module solves the solid time-dependent energy equations for the fuel, cladding, and heat exchanger tubes in axisymmetric 2D. We couple this tube-and-pin model to the sodium loop model by doing two things. First, the tube-and-pin model sends the surface temperatures to the loop model. The second part has the loop model calculate the heat transfer rate between the tube/pin surfaces and the sodium coolant and sends that rate back to the tube-and-pin model.

In each core, there are 5,700 tubes in a heat exchanger and 37,520 fuel pins. We are modeling one of each, so we scale the mass flow rate proportionally to accurately model the system. For the core side of the loop, we scale the flow rate to a value corresponding to a single fuel pin. This is 1/37520th of the core's primary mass flow rate. For the heat exchanger side of the loop, we scale the flow rate to a value corresponding to a single heat exchanger tube. This is 1/5700th of the core's primary flow rate.

We account for the additional heat capacity of the sodium pool using a MATLAB[50] function. As sodium in the primary loop exits the heat exchanger, the Pipe Flow Module calls this MATLAB function to bring the sodium to thermodynamic equilibrium with the current pool temperature. We set this new temperature as the inlet temperature for the core. We use the ANL correlations[51] for the sodium properties.

D. Dynamic Safety Analyses for SABR2

The above 3D nodal neutron kinetics model (with precomputed MCNP neutronics parameters and feedback due to Doppler, Na expansion and voiding, and fuel expansion) and a COMSOL T-H model of the sodium pools and secondary systems has been applied to simulate various accidents[29,30]. For 50% (failure of 1 of 2 pumps in secondary (LOHSA) or primary (LOFA) system) the core shifts to a slightly lower power level (99.9% for LOHSA and 97.2% for LOFA) without failure (no fuel melting or Na boiling). For a Loss of Power (LOPA) accident the core power shuts down to the 7% decay heat level in about 30s ---without any corrective action being taken. For 100% (failure of both pumps) LOFAs and LOHSAs and no corrective action, coolant boiling (1156 K) and fuel melting (1473 K) occur in the present design at about 27s and 36s, respectively, after primary system pump failure for the LOFA, and at about 56s and 84s, respectively, after secondary system pump failure for the LOHSA, unless corrective control action is taken before this time, which is possible.

Switching off the power to the fusion neutron source was found to be a much more effective control action than scramming control rods, so the primary corrective action for SABR is turning off the heating and current drive power to the fusion plasma when the pump failure is detected. Calculations indicate that the plasma fusion neutron source can be reduced so that the fission power is reduced to the 7% decay heat level within < 10s. We conservatively assume that the pump failure is detected within 2s and that another 3s is required for the control system to switch off the plasma power, and we then conservatively model the plasma neutron production rate decreasing exponentially from 100% with a time constant of 10 s. This corrective action is sufficient to prevent fuel melting and sodium boiling during the core fission power shutdown to decay heat level for the 100% LOFA and LOHSA (and occurs automatically for the LOPA). However, natural circulation is not sufficient to remove the decay heat and prevent subsequent failure unless at least one pump in each the primary and secondary systems can be restarted, indicating a need for a more effective decay heat removal system. The issue of decay heat removal in the absence of pumping power has not yet been addressed for SABR, and this is clearly an important next step in development of the concept.

An unfavorable neutron source feedback mechanism was identified--the increased transmission of source neutrons to the fuel assemblies when the intervening sodium pool density was reduced. This can be designed around to some extent by moving the fuel assemblies closer to the inner (front) wall of the sodium pool in Fig. 2.

VII. ECONOMICS

Over the past two years, the SABR research group has collaborated with Georgia Tech’s Scheller College of Business and the law school at Emory University to investigate the economic viability of the SABR concept and explore potential commercialization strategies[52].

This investigation initially focused on the United States as the target market for SABR, however the team later expanded to consider other markets (countries) around the world. The primary attributes for any suitable SABR market are: (1) a dependence on nuclear energy generation; (2) the existence of a “commercial” reprocessing program; (3) a favorable regulatory environment and public opinion; and (4) status as an ITER Party. Reprocessing, which is a method that separates the transuranic components of spent nuclear fuel (SNF), is a critical technology for any suitable SABR market.

There are significant challenges in assessing the economic viability of a potential SABR implementation, not the least of which is the significant amount of uncertainty in the construction cost of a SABR and associated reprocessing plants. The determination of whether a SABR scenario could be economically viable vis-à-vis direct burial of SNF in HLWRs cannot be based on a simple cost comparison because, unlike geological repositories, SABRs would produce electricity, change the number of traditional nuclear reactors in operation, and even affect the cost of fuel for those traditional nuclear reactors.

To reduce these complex considerations to a single number, the energy industry uses the Levelized Cost of Electricity (LCOE). The LCOE can be thought of as the current price of electricity per kilowatt-hour from a power plant which, when adjusted for inflation throughout the lifetime of the plant, would result in the plant meeting all debt obligations incurred in constructing the plant and provide a reasonable return to equity investors. Calculations of the LCOE for nuclear power typically stop short of accounting for the true societal cost of the spent nuclear fuel (SNF). Instead, they use the waste fee that nuclear plants are required to pay to the federal government. The team suggested modifying the calculation to compute a true, post-waste LCOE, which is better suited to evaluating the effects of a SABR program on the complex nuclear industry.

It is not feasible at this time to estimate the cost of a SABR and its prorated share of the associated fuel processing and re-fabrication facilities—a *SABR+* cost. Instead, we are taking the approach of determining a "break-even" *SABR+* cost with respect to direct burial of SNF in HLWRs. We will first determine the true LCOE of nuclear power with direct burial of SNF in HLWRs that can be secured indefinitely into the future. We will then determine the "break-even *SABR+"* cost for each SABR and its prorated share of the associated reprocessing and fuel fabrication facilities for which the electricity revenue and cost savings resulting from avoiding the construction of the majority of the repositories will result in a similar overall LCOE for nuclear energy, compared to the direct SNF burial in HLWRs (i.e. Yucca Mountain) scenario. A lower than "break-even" construction cost for a SABR and its prorated associated reprocessing and fuel re-fabrication facilities would result in a lower LCOE for nuclear power with SABRs than with HLWRs. If plutonium is separated during the reprocessing stage and used as fuel in critical fast reactors to produce additional electricity, the LCOE for the SABR scenario can be reduced; however this would mitigate the non-proliferation aspect of the reference SABR fuel cycle in which all the transuranics (including the plutonium) are processed as an aggregate metal. These issues remain to be worked out.

VIII. DISCUSSION

The work summarized above quantitatively characterizes the physical features and the fuel cycle, tritium self-sufficiency and safety performance characteristics of a subcritical fast transmutation reactor based on the existing Na-cooled, metal-fueled fast reactor fission physics/technology demonstrated in EBR-II and on the existing tokamak D-T fusion physics/technology, in both cases assuming successful extension to industrial scale (which has been done to varying degrees for the fission physics/technology, if not for the TRU-Zr fuel and processing technology, and which is being done by ITER in the case of the tokamak fusion physics/technology).

The Na-pool, TRU-Zr fuel fission physics/technology was chosen for SABR because most of the experience with fast reactors is with Na-cooled fast reactors, because of of the performance features of metal fuel exhibited by EBR-II, and because of the preliminary development of TRU-Zr fuel by ANL. The completion of the development of TRU-Zr fuel and the associated pyrolytic processing and fuel fabrication on an industrial scale is certainly a necesary next step. In particular, a confirmation that the high burnup obtained with U-19Pu-10Zr fuel in EBR-II by limiting the initial fuel smear density to <75% can also be achieved with TRU-Zr fuel, so that the fuel lifetime is set by the 200 dpa fuel pin clad limit, is a high priority next step.

Tritium self-sufficiency is essential for SABR. While the work to date indicates that it is possible, a more detailed analysis of trititum self-sufficiency of the SABR design is an important next step.

Another high priority next step is to extend the decay heat removal system design concept to handle the decay heat 1) in the case of severe 100% LOCAs, LOHSAs and LOPAs and 2) during refueling.

There are also other possibilities for the fast fission reactor coolants and fuels, and a comparative investigation of these in a SABR application would also be a useful next step.

Qualitative and semi-quantitative arguments have been made for the inherently superior transmutation and safety features of the inevitably more complex and costly subcritical reactors relative to critical reactors based on the same fission technology. An obvious necessary next step is to quantify these arguments for the Na-pool, metal fuel fission reactor physics/technology and the tokamak fusion neutron source physics/technology of the type discussed in this paper, by carrying out comparative fuel cycle and safety studies of "comparable" critical and sub-critical transmutation reactors.

The tokamak fusion neutron source based on ITER-level magnetic confinement fusion physics/technology suggests itself because it is the leading magnetic fusion physics/technology, which is being developed collaboratively by the international magnetic fusion program, and because the tokamak configuration provides for a large surface area distributed neutron source which is compatible with the objective of introducing source neutrons broadly distributed within the fission reactor. Investigation of the intrinsic or active burn control of possible plasma thermal excursions is a necessary next step.

There are, of course, other possible neutron sources. The concept of a subcritical reactor driven by other magnetic fusion confinement concepts, by a laser-driven inertial fusion plant and by an accelerator driven spallation neutron source (ADS, ATW) have all been suggested, and there is a major international effort on ADS. The problem with a neutron source based on other magnetic confinement fusion concepts is that these concepts are probably 25 years behind the tokamak in the status of physics development and that the overwhelming majority of the present international development effort is concentrated on the tokamak. The problem with the ADS is not in its development, which is quite advanced, but in that it provides a highly localized, essentially "point", source of neutrons that is not at all distributed over the reactor and that will create large radiation damage in the region of the source. The problem with the inertial fusion neutron source is that it too is a point source of neutrons, plus that the physics understanding does not seem to be as developed as was recently thought. A comparative study of the fuel cycle and safety performance of "comparable" subcritical transmutation reactors based on the same fission physics/technology, but driven by either tokamak or accelerator spallation neutron sources, would also be a useful next step.

The economics will be an important consideration in consideration of the development of a SABR. Development of a breakeven Levelized Cost of Electricity for a US Nuclear system with SABRs plus associated reprocessing and fuel fabrication facilities with a US Nuclear system with HLWRs only would be an important further step in development.

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