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Journal
Nuclear Technology, 194(2)

ISSN
0029-5450

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Publication Date
2016-05-01

DOI
10.13182/nt15-40

Peer reviewed

NUCLEAR TECHNOLOGY · VOLUME 194 · 233–251 · MAY 2016

Thorium Fuel Cycles with Externally Driven Systems

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Received March 16, 2015
Accepted for Publication July 31, 2015
http://dx.doi.org/10.13182/NT15-40

Abstract — *Externally driven subcritical systems are closely associated with thorium, partially because thorium has no naturally occurring fissile isotopes. Both accelerator-driven systems (ADSs) and fusion-driven systems have been proposed. This paper highlights key literature related to the use of thorium in externally driven systems (EDSs) and builds upon this foundation to identify potential roles for EDSs in thorium fuel cycles. In fuel cycles with natural thorium feed and no enrichment, the potential roles are (1) a once-through breed-and-burn fuel cycle and (2) a fissile breeder (mainly ^{233}U) to support a fleet of critical reactors. If enriched uranium is used in the fuel cycle in addition to thorium, EDSs may be used to burn transuranic material.*

These fuel cycles were evaluated in the recently completed U.S. Department of Energy Evaluation and Screening of nuclear fuel cycle options relative to the current once-through commercial nuclear fuel cycle in the United States. The evaluation was performed with respect to nine specified high-level criteria, such as waste management and resource utilization. Each of these fuel cycles presents significant potential benefits per unit energy generation compared to the present once-through uranium fuel cycle. A parametric study indicates that fusion-fission–hybrid systems perform better than ADSs in some missions due to a higher neutron source relative to the energy required to produce it. However, both potential externally driven technology choices face significant development and deployment challenges. In addition, there are significant challenges associated with the use of thorium fuel and with the transition from a uranium-based fuel cycle to a thorium-based fuel cycle.

Keywords — *Thorium, accelerator-driven system, fusion-fission hybrid.*

Note — *Some figures may be in color only in the electronic version.*

I. INTRODUCTION

Subcritical nuclear fission reactors driven by an external source of neutrons have been considered for more than half a century. The external source of neutrons in proposed concepts is usually either a spallation source, in an accelerator-driven system (ADS), or a fusion source, in a fusion-fission hybrid system (FFH). This paper focuses on ADSs or FFHs intended for production of electricity (or other energy carriers) that also fulfill various missions in nuclear fuel cycles with thorium fuel. These systems may have the potential to generate clean energy and help close the nuclear fuel cycle.

Several roles have been proposed for subcritical externally driven systems (EDSs), including transmutation of minor actinides and as a breeder of fissile material using fertile resource feed without enrichment. Both ADSs and FFHs are considered examples of EDS. A significant example of an EDS is the Accelerator

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Transmutation of Waste (ATW) program,[1] a major U.S. Department of Energy (DOE) effort that was aimed at burning long-lived minor actinides. Another frequently cited example is the Energy Amplifier, which was proposed as an option to address[2] "(1) the breeding process in thorium fuel, (2) the burning of the self-generated actinides, (3) the plutonium (higher actinides) burning of spent fuel from ordinary reactors and (4) fuel reprocessing/regeneration." These examples utilize a spallation source of neutrons driven by a proton accelerator. However, FFHs have also been considered since the 1950s (Ref. 3). One recent example is the Laser Inertial Fusion Energy (LIFE) engine,[4] where an inertial confinement fusion (ICF) neutron source drives a subcritical blanket.

Externally driven systems are particularly relevant with fertile thorium fuel since thorium has no naturally occurring fissile isotopes. The external source of neutrons can bootstrap a breed-and-burn mode by initially generating fissile uranium (mostly ^{233}U) from thorium. The primary objective of an EDS considered in this paper is energy production, with EDS also potentially fulfilling secondary objectives that support the primary mission or fulfill other fuel cycle functions.

An FFH concept may exhibit distinct advantages over an ADS concept with respect to energy production because the fusion driver can be energy self-sufficient and thus send electricity from the fission blanket entirely to the grid, whereas accelerator drivers will always require some fraction of the electricity generated by the fission blanket to be used for internal power consumption. However, even an FFH will need to use a portion of its total energy production (from both fusion and fission components) to sustain its operation, including the laser driver, refueling system, and pumps, among other systems.

The missions reviewed in this paper include (1) a once-through breed-and-burn fuel cycle, (2) a fissile breeder (mainly ^{233}U) to support a fleet of critical reactors, and (3) a burner that is dedicated to burning plutonium and minor actinides, which is relevant if enriched uranium is used in the fuel cycle in addition to thorium. In this paper, each of these potential missions is explored via holistic analyses of examples of each of these fuel cycles, supported by peer-reviewed reactor physics analyses and mass flow data. Central to this paper is a comparison of the present U.S. nuclear fuel cycle, a once-through fuel cycle with low-enriched uranium (LEU) in thermal spectrum critical reactors, with thorium fuel cycles employing EDS in breed-and-burn, breeder, and burner missions.

The fuel cycle analyses and mass flow data in this paper were performed as part of a DOE Evaluation and Screening (E&S) of nuclear fuel cycle options.[5] A key goal of the E&S effort was to identify fuel cycle options that offer significant benefits relative to the current once-through commercial nuclear fuel cycle in the United States with respect to nine specified high-level criteria, including waste management and resource utilization. The objective of the E&S was also to inform on potential areas for investments in research and development (R&D).

Most EDS concepts that have been proposed contain a subcritical blanket surrounding the interface with the external neutron source to multiply the source neutrons and maximize their effectiveness. A subcritical blanket is essential for energy production or effective transmutation and is essentially a nuclear reactor that operates at a multiplication factor [k-effective (k_{eff})] $<$1.0 (k_{eff} = 1.0 for a critical "self-sustaining" system). Subcritical operation results in a number of potential positive attributes:

1. As long as the k_{eff} of the system is sufficiently below 1.0, no reactivity perturbation can lead to a runaway transient with potentially catastrophic consequences.

2. When the accelerator beam or fusion source is turned off, the system shuts down.

3. The blanket multiplies the source neutrons from the external source by a factor of $1/(1 - k_{eff})$ making missions like power production and transmutation feasible with an externally driven concept. The source power can thus be lowered if k_{eff} is increased proportionally to maintain a constant system power generation, which could reduce system cost and minimize technical risk. In an ADS this could minimize the required accelerator beam power and keep it within achievable ranges, while in an FFH the energy gain expected from the fusion component could be substantially reduced to relax the demands on fusion and thus enable lower yields per target or decreased repetition rates. However, if the k_{eff} is too close to 1.0, the potential benefits of subcritical operation may be compromised.

4. There are potentially enhanced flexibility in blanket geometry and loading of materials than in a critical reactor.

The fact that the blanket is effectively a reactor, albeit a subcritical one, brings with it the safety and licensing requirements of a commercial reactor, with the added complexity of the interface with an accelerator-driven or fusion neutron source. This introduces new and different potential accident initiators, event scenarios, and issues with defining an appropriate containment/confinement boundary while allowing the external source to drive the EDS and simultaneously ensuring isolation in case of an accident. Although a subcritical fission blanket will shut down rapidly when the fusion reactor or accelerator beam is turned off, there will still be residual power generation from delayed neutrons and decay heat from fission products (FPs) and actinides. One of the most challenging reactor safety problems, decay heat removal in a severe accident, remains an issue with EDS just as it is with critical reactors.

I.A. Fusion-Fission Hybrid

At a very high level, all FFH concepts consist of a nuclear fusion reactor coupled to a fission blanket; the fusion component produces an excess of high-energy neutrons, which are then used as an external neutron source to drive a subcritical fission blanket with fertile and/or fissionable materials. All FFHs share these basic characteristics; however, an incredible variety of concepts exist for the fusion system, the fission blanket design, and the approach used for coupling the fusion and fission systems.

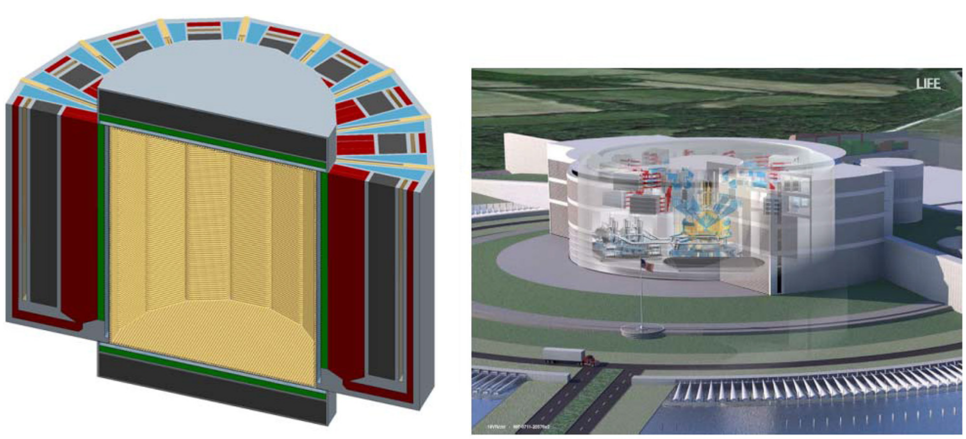
Fusion-fission hybrid systems exhibit a long history of research dating back to the 1950s in multiple nations,[6–8] with missions ranging from baseload electricity generation[9] to breeding fuel for fission reactors.[9,10] The majority of recent large-scale, well-developed fusion energy concepts use deuterium and tritium (D-T) as their fusion fuel and fit into two broad categories based upon how they achieve the conditions necessary for fusion reactions in the D-T fuel: ICF, in which neutral particle or ion beams directly or indirectly impart their energy upon the D-T fuel to create combined temperatures and pressures that support ample fusion reaction rates and confinement times in intermittent bursts, or magnetic confinement fusion involving electromagnetic systems that control ionic plasmas and heat them up enough to cause fusion reactions. As described in further detail below, the analyses performed for this work focused on FFHs using a solid-state laser-driven ICF concept. A notional possible realization of an FFH is shown in Fig. 1.

I.B. Accelerator-Driven System

A generic ADS consists of an accelerator, a target to convert the accelerated particles into "useful" particles for a desired application, and potentially a surrounding blanket where these particles can interact further to achieve a desired objective. Each of these components is a complex engineered system. The design and operation of the individual systems is a significant challenge, and this is exacerbated by the need for them to operate synergistically as an integrated system. Notional design possibilities of an ADS are shown in Fig. 2.

Most ADS concepts considered assume a proton accelerator. In principle, both linear accelerators (linacs) and cyclotrons are viable candidates for ADS. A White Paper prepared for the DOE Office of Science concluded that for the high beam powers (tens of megawatts) generally required for transmutation or energy production, linacs were best suited.[13] Recent work in cyclotrons, such as fixed-field alternating-gradient machines, suggests that they may also be viable options, depending on the beam power required.[14] The interaction/dependence of the required accelerator beam power and the blanket is discussed further below.

The primary purpose of the target in most ADSs is to convert accelerated incident protons to neutrons that are the useful particles for subsequent applications (e.g., energy production, material science, transmutation, etc.). Therefore, the top-level requirement of the target is to produce the maximum number of neutrons per proton (n/p) and to leak them out of the target, with minimal parasitic losses or modification in energy spectrum, for subsequent utilization. Achieving this objective requires trade-offs between engineering, materials, safety, operational, and



(a) (b)

Fig. 1. Notional design possibilities for (a) an FFH fusion chamber design and (b) a full plant layout, as reproduced from Ref. 11 (LIFE concept).

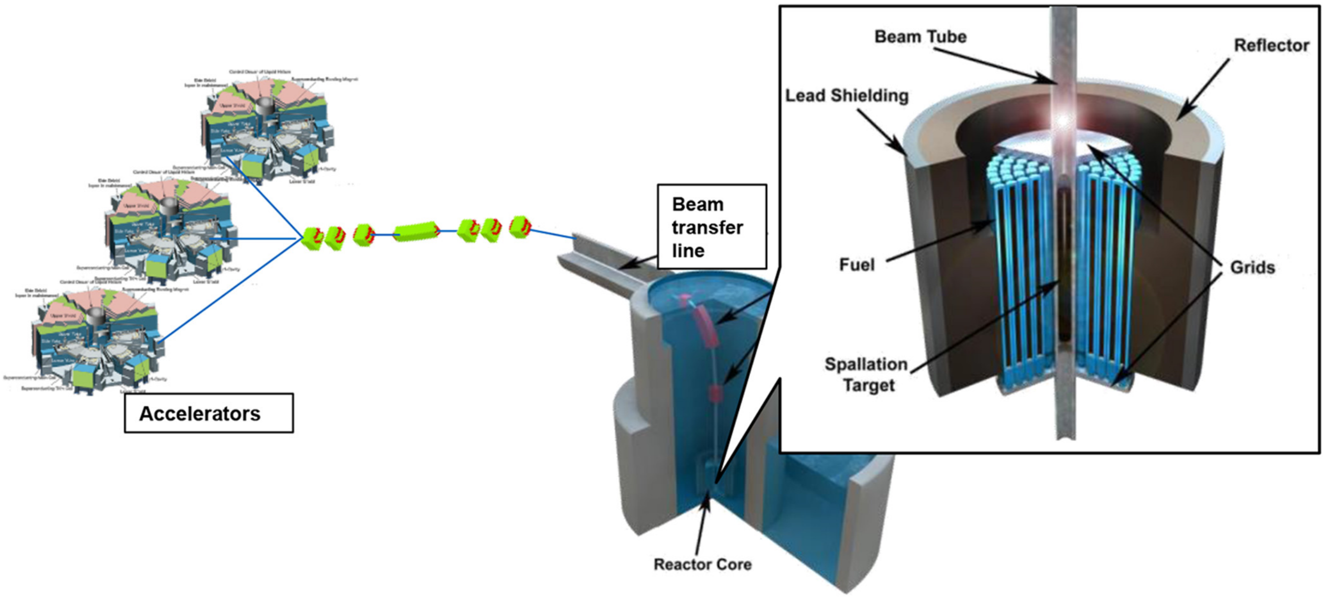


Fig. 2. Notional design possibilities for an ADS and subcritical blanket (portions adapted from the TRADE experiment[12]).

cost considerations. Candidate technologies for ADS targets are generally

1. solid target options, which consist of a solid material in the form of rods/spheres/plates to produce the neutrons and coolant flowing between the elements for heat removal
2. liquid target options, where a flowing liquid metal acts both as the source of neutrons and the heat removal media.

Tungsten, tantalum, and lead are the primary materials that have been considered, and/or used, for proton-driven solid spallation targets since the neutron yield (n/p) is proportional to the target atomic weight and density (number of target nuclei per cubic centimeter). While uranium and other actinides are also candidates and generally have a higher (n/p) ratio, they introduce engineering and environment, safety, and health complexities that have generally precluded their consideration unless the ADS contains a fissile blanket, since in that case similar issues must be addressed for both systems [e.g., FPs and transuranics (TRU) typical of a reactor]. The (n/p) ratio is also proportional to the energy of the incident protons and is roughly linear over the typical range of interest for ADSs (typically, ~0.5 to 3 GeV). Therefore, to generate a neutron source of sufficient intensity for a desired application, the energy of the protons impinging on the target or the current can be traded off against each other subject to accelerator performance and cost and engineering considerations. One example of the trade-offs would be that a higher proton energy will distribute the source in the direction of the proton beam, whereas a higher current will result in higher power densities near the front of the target as well as high damage fluence on the beam window.

The blanket designs typically proposed for an ADS will vary over a wide range of possibilities. These include the traditional fuel with cladding cooled by a flowing liquid, to fully liquid fuel variants such as a molten salt with dissolved fuel.

Some electricity will be needed to run the accelerator. Depending on the type of system and the objectives of the mission, there may be limited net production of electricity from this system under some scenarios, such as breeding of fissile material from a blanket that initially only contains natural thorium or uranium. Until sufficient fissile material is generated to raise the k_{eff} of the blanket, it will not produce power, and hence, the system will require power from the grid (i.e., a consumer of electricity part of the time). However, if the sole purpose is to serve as a prebreeder for subsequent systems, this could be acceptable. The production of fissile material will depend on the (n/p) and thus varies in direct proportion with the beam power of the accelerator.

I.C. Roles for Thorium EDSs in the Nuclear Fuel Cycle

Several roles for thorium EDSs in the nuclear fuel cycle have been proposed. One of the roles is in a breed-and-burn fuel cycle with either a thermal or fast neutron energy spectrum in the subcritical blanket. In this fuel cycle, thorium is the only natural resource feed. An FFH is a possible technology for once-through thorium fuel cycles featuring a thermal spectrum or a fast spectrum.

For these options, the feed fuel does not contain any fissile material (only thorium), and FFHs provide an abundance of neutrons for the breeding process while the fusion component produces enough energy to sustain itself.

Another role for a thorium EDS is as a breeder taking only natural thorium feed that supports a fleet of critical reactors. An ADS is a possible technology used to breed uranium (mainly ^{233}U) in a thorium blanket that will be used to support a thermal reactor fleet with thorium and uranium (mainly ^{233}U) fuel. The feed material is fertile thorium, with no natural fissile isotopes. Maintaining a heterogeneous blanket where several "batches" of fuel have fissile content will increase blanket self-multiplication and therefore increase the energy production, neutron flux, and bred fissile in the blanket.

The third mission considered is that of an EDS dedicated to the burning of TRU elements. The relevance is related to thorium fuel cycles that use enriched uranium support. These fuel cycles will produce significant amounts of TRU waste. An ADS, similar to that developed in the ATW program, is one potential technology option for burning TRU elements.

In this paper, each of these missions is explored via studies conducted as part of the DOE E&S study of fuel cycle options. Various reactor physics and mass flow analyses were conducted and reviewed systematically as part of the E&S effort. This paper summarizes some of these efforts and the key results in the context of these different missions for EDSs in thorium fuel cycles. In each case a comprehensive analysis of the entire fuel cycle was performed with the inventories in the reactor stage determined via detailed reactor physics analyses using either deterministic or stochastic tools.

II. BREED-AND-BURN FFH

This section investigates the role of EDSs in a breed-and-burn thorium fuel cycle. As the fusion component is energy self-sufficient and provides an abundance of neutrons to the blanket, FFHs are particularly suitable for a once-through breed-and-burn operation mode where natural thorium is initially loaded in the blanket and fissile material is generated and burned in situ until operational limits are achieved. The FFH analyzed in this study is the LIFE engine developed at Lawrence Livermore National Laboratory.[15]

The FFH employs an ICF system based on a National Ignition Facility (NIF)–like illumination geometry and hot-spot ignition.[16] The fusion neutron point source is located at the center of a 250-cm-radius chamber filled with low-pressure gas that protects the first wall from ions and X-rays. The chamber is enclosed by a spherical oxide dispersion strengthened (ODS) ferritic steel first wall coated with tungsten. The system is completed by a series of shells: a dedicated liquid LiPb layer for first-wall cooling, an injection plenum for the fission blanket's coolant, a multiplier layer, the fission blanket, and a reflector (Fig. 3). Dimensions vary depending on the fuel type. A 3-mm-thick ODS wall separates each shell, and 48 entrance ports for laser beams penetrate each layer. The blanket's coolant flows radially outward starting from the injection layer and moves from one layer to the next through perforated walls. FLiBe (2 LiF + BeF_2) is the preferred coolant for the blanket, but other options may be considered. The multiplier layer is filled with metallic beryllium pebbles coated with ODS steel and cooled by FLiBe—60% and 40% volume, respectively. High-energy neutrons from the source are multiplied by means of (n,2n) reactions in beryllium. Both thermal and fast neutron spectrum configurations can be engineered varying the fission blanket design. The analyses reported herein considered a fast spectrum option with solid fuel (pebbles) cooled by F-Li-Be.

The ICF system is self-sufficient in breeding its own replacement tritium and producing power for its laser driver and balance-of-plant (BOP) equipment. A 13.3-Hz fusion power repetition rate with 37.5-MJ fusion targets produces a 500-MW fusion power source and $\sim 2 \times 10^{20}$ 14-MeV neutrons per second. The fission blanket provides an energy gain (thermal fusion power to total system thermal power) of 4 to 8, depending on the initial fuel.

In the fast spectrum option considered in this analysis, fuel is in the form of tri-structural isotropic (TRISO) particles dispersed in 2-cm-diameter carbon pebbles. As pure thorium (no fissile material) is used at start-up, the FFH initially operates below nominal power until enough fissile material is bred. This time is defined as ramp-up time. After this point, nominal power is kept constant by

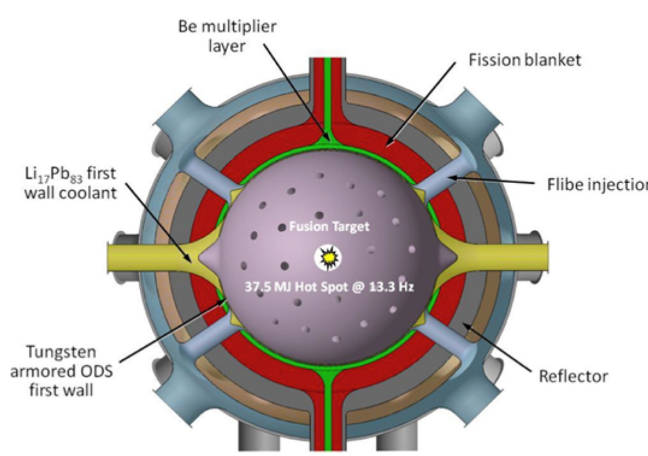


Fig. 3. Schematic cross section of an FFH based on NIF configuration.[16]

controlling the level of ^{6}Li enrichment in the blanket coolant. The entire fuel in the blanket is replaced with fresh fuel when the accumulated tritium to drive the ICF is exhausted (most common situation) or the blanket multiplication factor is too low to maintain an adequate blanket gain. Assuming a total power of 2000 MW(thermal) (blanket gain 4), a burnup of 729 GWd per metric ton heavy metal (MTHM) could be theoretically achieved in 55 calendar years or 53.2 effective full-power years (EFPY). Neither TRISO nor other forms of fuel have been proven capable to reach the burnup and radiation damage levels envisioned in this system, and extensive fuel development and qualification will be required.

The FFH in this case was modeled using MCNP5 (Ref. 17). The model includes detailed TRISO particles and pebbles, arranged according to a simple cubic and a body-centered-cubic lattice, respectively. Material densities and cross sections (ENDF/B-VII.0) are assumed at the nominal operating temperature. Scattering kernels apply to graphite, metallic beryllium, and iron. Depletion calculations were performed using MONTEBURNS (Ref. 18), which couples MCNP with ORIGEN2.2. A newly developed code called the LIFE Neutronics Code[16] controls ^{6}Li enrichment, adjusting its level at every depletion step according to tritium breeding ratio, tritium mass, and thermal power constraints.

II.A. Fuel Cycle Data

Figure 4 shows the material flow diagram for the breed-and-burn FFH fuel cycle described above, and Table I summarizes various performance parameters. The fuel form is thorium oxycarbide (ThOC). The thermal efficiency of the Brayton cycle was assumed to be 43%. The overall net efficiency was calculated assuming that the ICF would require 175 MW(electric) to operate the laser and the BOP would retain 20 MW(electric). FFHs present the unique characteristic among other technologies of including an energy-producing nonfission system: the fusion engine. With a 43% efficiency Brayton cycle, the 500 MW(thermal) derived from fusion is enough to cover laser and BOP needs, with 20 MW(electric) left for the grid. The blanket in this example is a fast spectrum blanket, which takes advantage of a small "fertile fission bonus" due to the fission of ^{232}Th via 14-MeV neutrons.

TABLE I

Fuel Cycle Performance Parameters*

Parameter	Value
Total power [MW(thermal)]	2000
Blanket power [MW(thermal)]	1500
Fusion power [MW(thermal)]	500
Brayton cycle efficiency (%)	43
Laser power [MW(electric)]	175
BOP power [MW(electric)]	20
Capacity factor (%)	90.0
Net efficiency (%)	33.25
Fuel form	ThOC
Discharge burnup (GWd/MT)	729
Specific power (MW/MTHM)	37.5
Fuel management	One batch
Fuel inventory in core (MTHM)	40.0
Fuel residence time (EFPY)	53

*References 19 and 20.

Material mass flow rates were also calculated on the basis of producing 100 GW(electric) · yr/yr of energy. Table II lists the mass of heavy metal (HM) per unit of electricity generated, including 0.2% fuel fabrication losses. Since fusion provides ~3% of the total FFH electricity [20 MW(electric) out of 665 MW(electric)], the mass flow of heavy metal is reduced by the same factor.

Besides thorium, the operation of an FFH requires the following material inputs: (1) deuterium as fusion fuel, (2) beryllium as neutron multiplier, and (3) ^{6}Li to breed tritium. An external feed of tritium is only required at start-up; after that, the plant will be tritium self-sufficient. An accurate estimation of the mass flows required to fuel the fusion component would require more detailed information on the performance of this system. Since these data

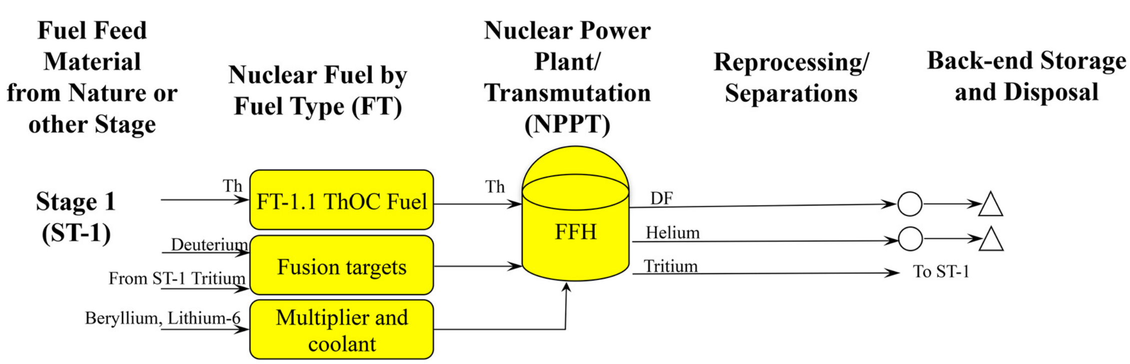


Fig. 4. Material flow diagram for once-through FFH thorium fuel cycles.

TABLE II

Fuel Mass Flow Data Based on EFPY

Parameter	Value
Fuel mass [MTHM/GW(electric) · yr]	1.164
Loss [MTHM/GW(electric) · yr]	0.002
Thorium mass [MTHM/GW(electric) · yr]	1.166

were not available, estimates of the minimum masses of deuterium and tritium required to operate the fusion system were determined using the number of fusion reactions necessary to produce fusion's energy share and assuming that each fusion reaction consumes one atom of deuterium and one atom of tritium. Similarly, the mass of ^{6}Li required to breed tritium was determined under the constraint that one tritium atom needs to be produced per fusion reaction to maintain the system self-sufficient and assuming negligible tritium production from other sources (e.g., Be and ^{7}Li). Helium is produced as a by-product of fusion and (n,T) reactions in ^{7}Li; therefore, two atoms of helium are involved with each fusion event. To balance the mass flow, neutron masses were also considered. Each fusion produces one neutron, but each tritium breeding reaction requires one neutron, and the two events cancel out in the mass balance. Estimated mass flows are summarized in Table III and are the same regardless of neutron spectrum in the fission blanket because both options use the same fusion driver. These are minimum theoretical estimations and do not include losses due to fabrication, fuel recovery, and decay of tritium.

Target, coolant, and neutron multiplier will be activated and are expected to generate a waste stream. Storage and disposal requirements for these activation products remain to be determined. FFHs represent an enabling path for this fuel cycle because the fusion source could potentially provide abundant neutrons at no net electricity cost. It was also assumed that the blanket uses solid fuel (TRISO particles carried in carbon pebbles) and operates in a batch mode. From start-up with no fissile content to discharge when ~75% of the initial heavy metal has been consumed, the neutron energy spectrum varies considerably.

TABLE III

Mass Flows for the Fusion System

Element	Type	Mass [MT/GW(electric) · yr]
Deuterium	Input	0.402
Tritium	Input/output	0.602/0.602
^{6}Li	Input	1.201
Helium	Output	1.599

III. PARAMETRIC COMPARISON OF BREED-AND-BURN ADS AND FFH

The objective of this study is to compare analyses of ADSs and FFHs for a breed-and-burn thorium EDS with a thermal neutron energy spectrum. In this case the thermal spectrum blanket selected was a system where the actinides are in a molten salt solution. This work was motivated by the desire to understand differences in fuel cycle performance between an ADS and an FFH. In the ADS, a proton accelerator drives a subcritical reactor with spallation neutrons that originate from the interaction between the proton beam and lead targets. These studies were performed using MCNPX (Ref. 21) and started with a multibeam ADS model. In the MCNPX model of an FFH, the accelerator in the ADS was exchanged for a fusion-generated source of 14.1-MeV neutrons. The lead targets in the ADS were voided and the fusion source was treated as a point source at the center of each fuel blanket. The goal of this study was to investigate the flux spectra in the fuel region in each blanket for two different external neutron sources for the subcritical thermal spectrum system.

III.A. Comparison of Flux Spectra

Figure 5 shows the normalized flux spectra in blanket fuel regions for both systems. The slight increase in flux in the highest energy bin in the ADS reflects the spallation neutron source energy. These source neutrons have energies >20 MeV (Ref. 22). Reference 22 shows that 16.8% of the neutron energies are >20 MeV and 3.3% are >150 MeV for a 1-GeV proton source. Figure 5 shows that in the thermal and epithermal regions the spectra between the two systems are very similar. They differ in the fast

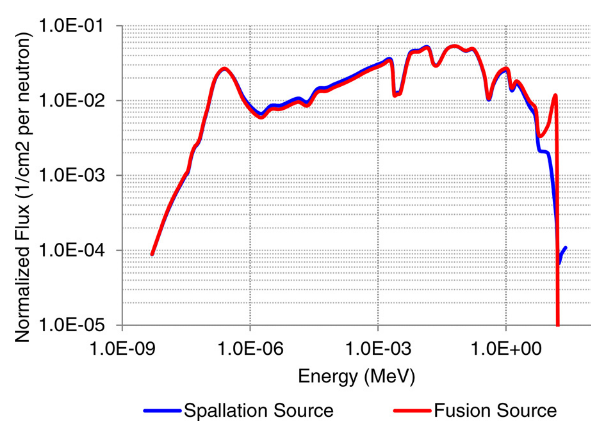


Fig. 5. Comparison of normalized flux spectra between the spallation source and fusion source systems.

energy range due to the varying neutron source energies. The purpose of this investigation was to study the thermal flux in the two systems, and this study revealed that the subcritical system provides a very similar thermal spectra, regardless of the neutron source.

III.B. Reactions in Selected Nuclides

The one-group, flux-weighted macroscopic cross sections for selected nuclides were calculated within the fuel region in both systems. Reaction rates for various reaction types such as $(n,2n)$, $(n,total)$, (n,γ), and $(n,fission)$ were computed using MCNPX tallies. The analysis in this section is presented for the average fuel zone since the spectra in all the fuel zones are similar due to the reflecting boundary condition on the edges of the core. Table IV shows selected isotopes with the greatest neutron interaction in the salt mixture within each fuel zone. As expected, total neutron interactions are greatest with ^{19}F, ^{232}Th, ^{9}Be, ^{7}Li, and ^{233}U. Also as expected, the results for the two systems are very similar.

Table IV shows differences in the one-group, flux-weighted total macroscopic cross sections for selected isotopes between the ADS and FFH. The relative percent difference presented are computed using the spallation source as the reference, as shown in Eq. (1):

$$\text{Relative \% Difference} = 100 \times \left(\frac{\Sigma_{i,fusion} - \Sigma_{i,spallation}}{\Sigma_{i,spallation}} \right) . \qquad (1)$$

For the isotopes with the largest total macroscopic cross sections (^{19}F, ^{232}Th, ^{9}Be, ^{7}Li, and ^{233}U) the relative percent difference in the total macroscopic cross sections obtained from the FFH and ADS is <2.6%. The largest relative difference is 9.0% for ^{240}Pu; however, the total macroscopic cross section is very small. The relative percent difference for total macroscopic cross section between the FFH and ADS for the fuel salt is only 0.9%. Therefore, this shows that the total behavior of neutron interactions in the fuel salt between the two systems is very similar.

Table V provides the one-group, flux-weighted macroscopic fission cross sections obtained from MCNPX. As expected, fission reactions occur primarily in ^{233}U and this percentage is comparable in both systems. The results also show that there is a small probability of fast fissions in ^{232}Th. Although there is less of a "fertile fission bonus" in the ADS, this has a small impact on the overall magnitude of the macroscopic fission cross section.

Table V also shows that the relative percent difference in the cross sections between the two systems is greatest for ^{232}Th, ^{238}U, and ^{240}Pu. This difference is due to the dissimilarity in the fast flux spectra between the ADS and FFH. The FFH has a harder spectrum in the fuel zone and this leads to more fast fissions in ^{232}Th, ^{238}U and ^{240}Pu.

Table VI shows the relative percent differences in the one-group, flux-weighted (n,γ) macroscopic cross sections in the selected nuclides for both the ADS and FFH. It shows that ^{232}Th is the largest source of (n,γ) reactions in the salt mixture. The isotopes that contribute to (n,γ) reactions in the salt is similar in both the ADS and FFH. The results also show that the differences in the cross sections is small for the isotopes that are a significant source of (n,γ) reactions in the salt mixture. The (n,γ) reaction probability is low in isotopes with a larger relative percent difference.

Table VII presents the $(n,2n)$ macroscopic cross sections obtained from both systems. The results show that there is a significant difference in the cross sections for

TABLE IV

One-Group Flux-Weighted Total Macroscopic Cross Sections for Average Fuel Material

	Total (l/cm)		
	FFH	ADS	Relative Percent Difference
^{232}Th	5.04E-02	5.15E-02	−2.0
^{233}U	3.49E-03	3.58E-03	−2.6
^{235}U	2.22E-04	2.25E-04	−1.2
^{238}U	5.64E-08	5.98E-08	−5.7
^{239}Pu	4.06E-07	4.08E-07	−0.4
^{240}Pu	1.39E-07	1.52E-07	−9.0
^{9}Be	2.67E-02	2.73E-02	−2.3
^{19}F	1.94E-01	1.95E-01	−0.6
^{7}Li	2.65E-02	2.60E-02	1.7
Fuel/salt	3.08E-01	3.11E-01	−0.9

TABLE V

Average Fission Macroscopic Cross Sections

	Fission (1/cm)		
	FFH	ADS	Relative Percent Difference
^{232}Th	4.84E-05	2.86E-05	69.2
^{233}U	2.51E-03	2.57E-03	−2.5
^{235}U	1.39E-04	1.39E-04	−0.4
^{238}U	1.23E-10	7.73E-11	59.6
^{239}Pu	2.37E-07	2.37E-07	−0.1
^{240}Pu	3.15E-10	2.73E-10	15.2
Fuel/salt	2.71E-03	2.75E-03	−1.6

TABLE VI

Average (n,γ) Macroscopic Cross Sections

	(n,γ) (1/cm)		
	FFH	ADS	Relative Percent Difference
^{232}Th	4.30E-03	4.49E-03	−4.4
^{233}U	3.27E-04	3.42E-04	−4.3
^{235}U	3.55E-05	3.67E-05	−3.3
^{238}U	1.52E-08	1.69E-08	−9.8
^{239}Pu	1.44E-07	1.45E-07	−0.5
^{240}Pu	1.19E-07	1.31E-07	−9.5
^{9}Be	2.59E-06	2.58E-06	0.4
^{19}F	4.27E-05	4.26E-05	0.2
^{7}Li	5.34E-05	5.34E-05	0.1
Fuel/salt	5.84E-03	6.12E-03	−4.5

TABLE VII

Average (n,2n) Macroscopic Cross Sections

	(n,2n) (1/cm)		
	FFH	ADS	Relative Percent Difference
^{232}Th	9.09E-05	7.29E-06	1146.9
^{233}U	2.23E-07	2.41E-08	823.5
^{235}U	4.20E-08	5.02E-09	735.4
^{238}U	3.86E-11	3.88E-12	893.6
^{239}Pu	8.61E-12	8.49E-13	914.1
^{240}Pu	5.17E-12	2.86E-13	1705.9
^{9}Be	9.15E-05	4.55E-05	100.9
^{19}F	1.49E-05	7.59E-07	1867.2
^{7}Li	7.60E-06	2.91E-07	2515.3
Fuel/salt	2.11E-04	5.41E-05	289.8

this reaction type. This is due to the difference in the fast spectra in the fuel zone for the two systems. The probability of (n,2n) reactions is very low in comparison to other reaction types. Therefore, the overall effect of this difference on the total cross sections for the selected isotopes in the salt mixture is very small.

III.C. Summary

The purpose of this parametric study was to analyze the differences in the thermal flux spectra between possible ADS and FFH thermal spectrum systems. A comparison of the macroscopic cross sections between the two systems show that the one-group, flux-weighted total cross sections for the selected nuclides compare well to each other. There are larger differences in the macroscopic cross sections for reactions that predominantly occur in the fast energy range, however, the probability of these reactions is small. This study shows that the reactor physics of the blanket is similar with either a fusion or spallation source. However the fusion source is expected to be the better choice for a breed-and-burn mission because it is expected that the fusion source yield more electricity delivered to the grid. No power produced by the subcritical blanket in the FFH will be used to power the fusion plant. The ADS assumes that a portion of the power produced by the blanket will provide energy to the accelerator. Therefore, the net power produced by the FFH is higher because it is expected that the fusion source will sustain itself.

IV. THORIUM ADS BREEDING URANIUM (MOSTLY ^{233}U) TO SUPPORT A CRITICAL REACTOR FLEET

In this example role of an EDS in a thorium fuel cycle, an ADS fueled with thorium breeds U3 (primarily ^{233}U) that is subsequently used in a pressurized water reactor (PWR) with U3 and thorium fuel to produce power. The ADS also produces power after a fuel cycle start-up transition.

The ADS stage consists of an accelerator, which accelerates protons that impinge on a molten lead target producing spallation neutrons. A blanket containing thorium surrounds the target in a coaxial manner. Since thorium has no fissile isotope, the blanket will have an essentially zero multiplication factor at initiation of the fuel cycle, and thus, to breed a significant quantity of fissile material in a reasonable time frame, the accelerator would require a high beam power to maximize the production of spallation source neutrons. The spectrum of the neutrons in the blanket will be particularly important in this case since it is desirable to maximize the production of ^{233}U. Therefore, to have a relatively hard neutron spectrum, the blanket fuel assemblies are cooled by liquid sodium. The window and target diameters (assuming a reference proton energy of 1 GeV) are determined largely by the accelerator power (product of beam energy and beam current). Higher beam energy will have implications on the axial length of the neutron producing volume while higher beam current will affect the damage to the window, which separates the spallation target from the vacuum of the proton drift tube. The irradiated fuel elements from the ADS are reprocessed, and the bred uranium (U3) is used as feed to a PWR in the subsequent second stage. Thorium is recycled in the ADS, and the PWR recycles the U3 and thorium as appropriate. Alternative approaches to the ADS start-up transition are possible to reduce the time required and/or the power requirements for the

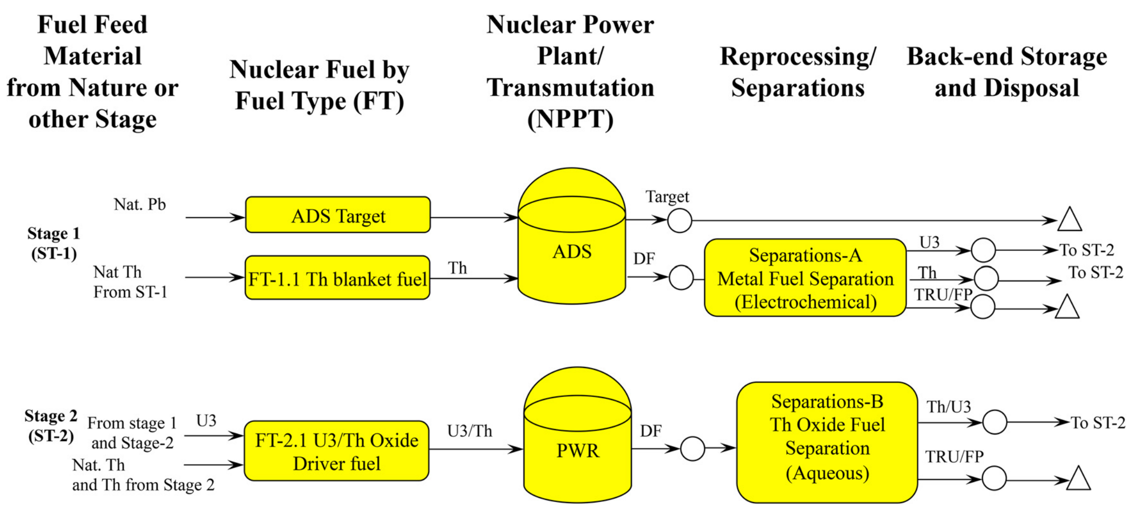


Fig. 6. Steady-state material flow diagram for example breeder mission.

accelerator, including blending the initial blanket loading with some fissile material to increase the multiplication of the source neutrons.

The PWR takes U3 and Th feed from the reprocessing of fuel discharged from the ADS (stage 1). A reprocessing step follows where uranium is coextracted with thorium and is fed as makeup fuel into the stage 2 PWR. The TRU and the FPs are disposed of as high-level waste (HLW). The material flow diagram that illustrates the example breeder mission for an EDS in a Th fuel cycle is shown in Fig. 6.

The overall approach used in the analysis of the ADS is based on the MCNPX Monte Carlo tool[21] and the CINDER-90 depletion analysis tool.[23] These tools are used in an iterative manner to track the changing composition of the target, the accelerator power required, and the power generated by the blankets.

The calculation approach is based on the MCNPX tool in the external source mode to determine the neutron spectrum when the proton beam is active and the criticality source (k-code) mode to check the changing multiplication factor between burn steps. Depletion calculations are carried out using the CINDER-90 tool. The reason for carrying out the two-step calculation is because MCNPX does not allow for depletion in the fixed-source mode. The calculation sequence is shown in Fig. 7. The calculation sequence begins with an MCNPX source calculation at beginning of cycle (BOC). This calculation simulates the proton beam, target, and blanket. This calculation results in the neutron energy spectrum in each blanket zone, and in addition, the fission heat is also tallied for each blanket zone. The depletion is then conducted in CINDER-90 utilizing the spectrum tallied in MCNPX. The revised material inventories are used in the MCNPX criticality source mode (proton beam off) to check the magnitude of the multiplication factor. If end of cycle (EOC) has been reached, the blanket fuel is shuffled. Finally, the process is repeated until equilibrium BOC and EOC k_{eff}'s are achieved.

The power generated in the blanket zone of an ADS was assumed to be entirely from fission of the fissile material. The accelerator power does not include the efficiency of the accelerating stages to accelerate the protons. This efficiency is assumed to be 50% in this paper. This efficiency was taken as a representative value, noting that accelerator "wall-plug" efficiency is a major consideration in the viability of an ADS (Ref. 13). Thus, to first order, the power required by the accelerator at any time will be twice the beam power.

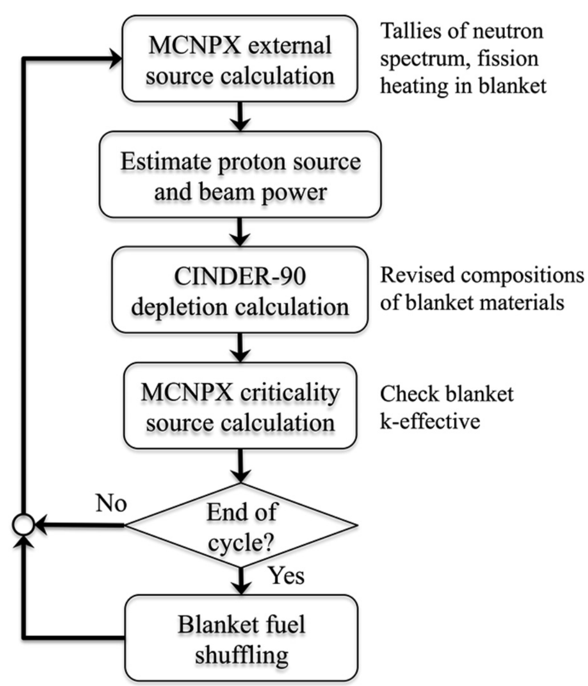


Fig. 7. Calculation flow diagram.

IV.A. Fuel Cycle Data

The parameters utilized in the analysis of this example option are shown in Table VIII.

The buildup from pure thorium fuel to a "near equilibrium cycle" will take many years due to the time required to build up fissile U3 and the low blanket self-multiplication. This will also depend on the power of the accelerator used in the ADS. This is due to a number of factors, including the fact that the multiplication factor in the blanket at initiation of the fuel cycle is substantially below unity. The source multiplication from the blanket is very low, requiring that most of the neutrons be supplied by the spallation reactions taking place in the target volume. These neutrons leak out of the target volume and convert the fertile material into fissile material, which increases the multiplication factor as a function of fuel transmutation. However, this transmutation process is a function of the neutron energy spectrum and flux, capture cross section of the fertile material, decay of the intermediate nuclei following neutron capture, and the capture cross section of the intermediate nuclei. The sensitivity of the ^{233}U buildup and EOC multiplication factor to the accelerator beam power is shown in Table IX. The sensitivity of the blanket thermal and electric power to the beam power is shown in Table X.

It is desirable to breed sufficient fissile material to achieve a multiplication factor of ~0.95, implying a source multiplication of 20 at "equilibrium." At this stage a shuffling scheme is initiated in which fresh fuel is loaded into the outer volumes of the blanket and the irradiated fuel is progressively moved inward in a radial direction. The material in the innermost volumes is discharged from the blanket, cooled, and sent to a reprocessing facility. At BOC, after each fuel shuffle, the multiplication factor drops and then increases with burnup of the

TABLE VIII

Analysis Parameters for Example Breeder Mission

Reactor Stage		ADS	PWR
Core configuration		ADS with Th-Zr blanket fuel	U3/Th oxide
Core thermal power [MW(thermal)]		611.3	3400
Net thermal efficiency (%)		40 (23.6)[a]	33
Capacity factor (%)		90	90
Specific power density (MW/IHMMT)		20.4	41.8
Electrical energy generation sharing at equilibrium (%)		20.5	79.5
Fuel purpose		Blanket	Driver
Fuel chemical form		Metal	Oxide
Fuel physical form		Pin bundle, ducted	Pin bundle, ductless
Reactor average discharge burnup (GWd/MT)		138.0	62.5
Charge fuel composition	Initial nuclear material(s)	Th	U3/Th
	(^{235}U + ^{233}U)/total U + Th (%)	0	5.9
	Th/total HM (%)	100	94.1
	TRU/total HM (%)	0	0
ADS target materials		Pb	—
ADS target charge rate [kg/GW(electric) · yr]		11315.4	—
Nonfissionable target transmutation fraction (%)		0.416	—
Fuel residence time in reactor (EFPY)		18.5	4.1
Number of fuel batches		3	3

[a]Accounts for power required to run the accelerator.

TABLE IX

Sensitivity of ^{233}U Buildup to Accelerator Beam Power

Beam Power (MW)	Time to Peak ^{233}U (days)	k_{eff}	^{233}U Inventory (kg)
350	240	0.79064	160.8
250	330	0.81745	169.2
150	540	0.85249	173.5
50	1520	0.89560	187.5

TABLE X

Sensitivity of Blanket Power at EOC to Accelerator Beam Power

Beam Power (MW)	Blanket Thermal Power [MW(thermal)]	Blanket Electric Power [MW(electric)]	Accelerator Power Required [MW(electric)]	Fraction of Required Accelerator Power Generated by Blanket (%)
350	360	120	700	17
250	310	103	500	20
150	225	75	300	25
50	120	40	100	40

fuel, reaching a maximum at EOC. A "near equilibrium state," for the case under consideration, is achieved following ~28 years of accelerator operation. In a breakeven fast spectrum critical reactor with blankets, it takes close to 100 years to achieve "true equilibrium"; a "near equilibrium" is achieved in ~20 to 40 years. This also depends on the relevant quantity utilized to assess whether equilibrium has been achieved.

The buildup of ^{233}U, ^{233}Pa, and other uranium isotopes is shown in Fig. 8. This plot follows a single thorium assembly from the initial fueling until the final discharge. The calculation assumes three batches and a total fuel residence time of 21 years. This figure illustrates the fact that ~60% of the ^{233}U buildup occurs during the first cycle (i.e., the first 7 years of core residence). During the second and third cycles, relatively more ^{233}U is burned to support the self-multiplication within the blanket. This figure encapsulates many of the key challenges facing a reactor system with a pure thorium feed: very low initial multiplication for charge fuel, long fuel residence times and associated fast fluence, and the competing relationship between breeding and burning ^{233}U. The isotopic vector of the discharged uranium is shown in Table XI.

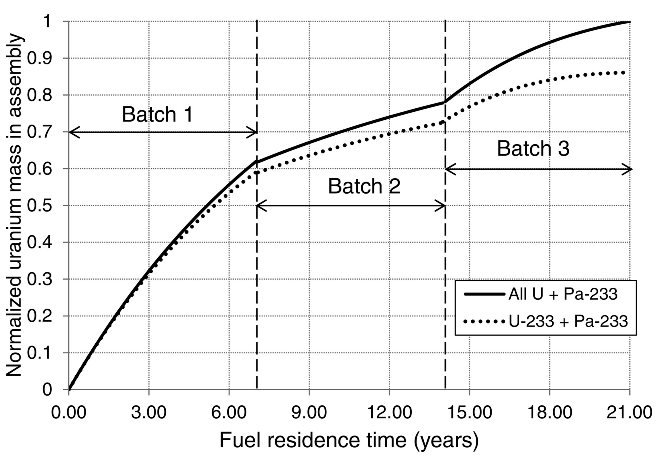


Fig. 8. U3 buildup in a representative ADS assembly.

Initially, the ADS requires full-power operation of the accelerator, assumed to be 350 MW of beam power, which requires ~700 MW of electric power from the grid assuming a wall-plug to beam power efficiency of 50%. However, as the fissile content in the blanket increases, the power requirements for the accelerator decrease, due

TABLE XI

Isotopic Vector of the Discharged Uranium from the ADS

Isotope	Mass (kg)
^{232}U	0.8
^{233}U + ^{233}Pa	1175.5
^{234}U	120.2
^{235}U	44.2
^{236}U	4.9
^{237}U	0.0
^{238}U	0.0

to the increase in the blanket source multiplication, and eventually, it might be possible for the ADS to generate sufficient electricity to be a net producer of electricity.

The reference reactor considered for stage 2 is a PWR with three-batch fuel management and 18-month cycles typical of current U.S. PWRs. The reactor had been loaded with a mixture of recovered uranium (RU) from the stage 1 ADS (mainly ^{233}U) as a main fissile material and RU from stage 2, in addition to recovered thorium (RTh).

The current study assumed a homogenous fuel loading; i.e, all 264 fuel rods of the 17 × 17 fuel assembly contained the same initial fuel loadings/compositions. The cycle length assumed for this analysis is 18 months with a three-batch fuel management scheme and 3% neutron leakage. To obtain the correct stage 2 RU vector (discharge composition), the burnup calculations were performed in two steps. In the first step, the initial heavy metal loading consists of 4.7 wt% of RU from the discharge ADS blanket fuel of stage 1 (which is mostly ^{233}U). In stage 2, the fresh fuel consists of 2.5 wt% of RU from stage 1 (makeup) mixed with 3.4 wt% of RU from stage 2. The resulting ^{233}U enrichment of the fresh fuel becomes 4.5 wt%. The RU mass balance between stage 1 and stage 2 was used to estimate the fraction of the electricity generated by each stage. Stage 1 generates 20.5% of the electricity, and stage 2 generates 79.5%.

V. BURNING OF TRU WASTE IN THORIUM FUEL CYCLES WITH ENRICHED URANIUM SUPPORT

The high-level objectives of this fuel cycle are to utilize existing thermal reactor technology, in this example case PWRs coupled with thorium fuel to extend uranium resources while burning TRU isotopes in an EDS, in this example case an ADS. Figure 9 shows the interpretation of the fuel cycle. Major challenges in the potential implementation of this fuel cycle include development cost, especially related to accelerator-driven subcritical reactor technologies, as well as fuel fabrication and separations technologies.

The first stage consists of a PWR. This PWR is based on state-of-the-art technology with three-batch fuel shuffling, but with two caveats: (1) the core is a heterogeneous mix of LEU uranium oxide seed fuel and thorium oxide blanket fuel and (2) the core specific power is derated by ~50% to reduce power peaking factors at beginning of life. In this calculation, the seed and blanket fuel pins were considered within a seed blanket unit configuration. The fuel is irradiated to a discharge burnup of ~61.7 GWd/MT. The discharged fuel from stage 1 consists of two streams: (1) discharged seed fuel and (2) discharged blanket fuel. The discharged uranium from the blanket fuel is coextracted with thorium. The RU—primarily ^{233}U—and some RTh are sent to stage 2, TRU are sent to stage 3, and FPs are sent to disposal. Some RU will also be sent to disposal. The discharged blanket fuel is separated, with RU (primarily ^{233}U) sent to stage 2 and RTh recycled within stage 1. Fission products are sent to disposal.

Stage 2 consists of a PWR based on the current state of the art with three-batch fuel shuffling but with RU (primarily ^{233}U) as the fissile material. The fuel is a homogeneous mixture of uranium oxide and thorium oxide. The fuel is irradiated to a discharge burnup of ~56.0 GWd/MT. The thorium oxide is the fertile material; some

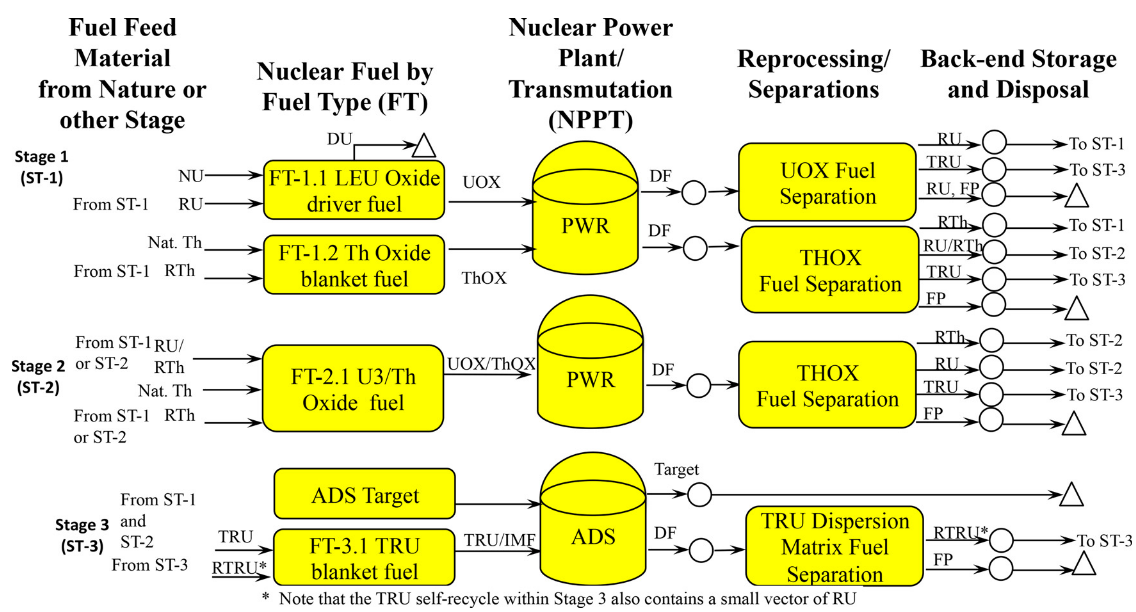


Fig. 9. Fuel cycle overview schematic with burner EDS.

fraction of the thorium is natural makeup feed, and some fraction is RTh recycled within stage 2. The recovered uranium oxide is recycled within stage 2, with makeup feed from stage 1. The discharged fuel is separated into a RU stream (recycled within stage 2), a RTh stream (recycled within stage 2), a TRU stream (sent to stage 3), and an FP stream (sent to disposal).

Stage 3 consists of an ADS based on the ATW design.[1] Within stage 3 the fuel is TRU in a zirconium metal dispersion with a zirconium matrix. The fuel is irradiated to a discharge burnup of ~195.0 GWd/MT. The feed for stage 3 is recycled TRU from stage 3 with makeup TRU from stage 1 and stage 2. Fission products are sent to disposal. There is also some RU vector within the recycled TRU, due to the radioactive decay of the TRU. This RU vector is recycled within stage 3. Additional waste steams include the Pb target utilized within the ADS. The discharged target may contain a considerable amount of radioactive spallation products. The target characteristics have not been determined at the time of this draft.

V.A. Fuel Cycle Data

The calculations for stage 1 were performed utilizing the SCALE package.[24] The TRITON control tool and NEWT discrete ordinates tool were utilized to analyze the PWR on a single-assembly basis. A polynomial reactivity model was utilized to estimate discharge burnup and cycle length, due to the highly nonlinear reactivity behavior of the heterogeneous assembly.[25] The utilization of higher-order reactivity models is warranted in such circumstances, and higher-order reactivity models have been extensively utilized.[25,26] A leakage penalty of 3% Δk was assumed. The specific power density (SPD) was derated to 17.5 MW/MT. The convergence of the RU vector was determined by the relative error in the RU vector from cycle to cycle. The maximum single isotope relative error in the discharged uranium vector after eight recycles was 0.4% in the ^{234}U composition. The cycle length converges after only three or four cycles. The number of seed and blanket fuel pins was not fully optimized for this study; the number of seed fuel pins per assembly is 108, and the number of blanket fuel pins is 156. The assembly model utilized in the stage 1 analysis and the convergence of the infinite multiplication factor are shown in Fig. 10 for the first eight recycles. The seed and blanket power share is roughly 50% each at discharge. The postirradiation cooling time is 5 years, and the fabrication time is 2 years. The analysis assumes a 1% loss in reprocessing and 0.2% loss in fabrication.

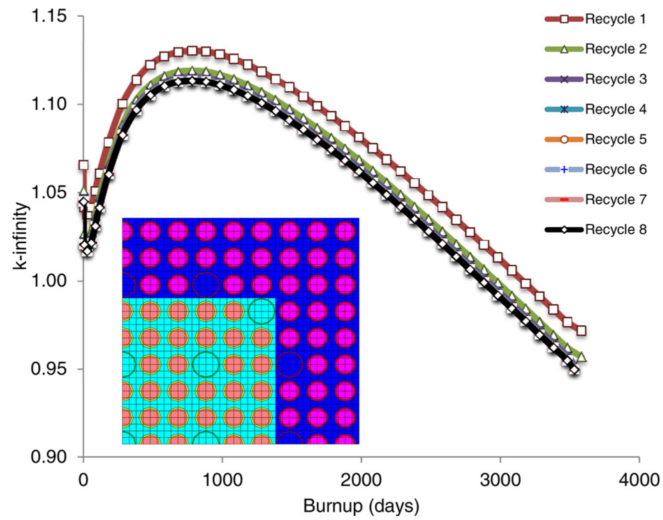


Fig. 10. Stage 1 assembly and *k*-infinity.

The calculations for stage 2 were also performed utilizing the TRITON/NEWT tools, utilizing the linear reactivity model with 3% leakage. The specific power was 36.6 MW/MT. The convergence of the RU vector was determined by the relative error in the RU vector from cycle to cycle. The maximum relative error in the discharged uranium vector after 12 recycles was 5.2% in the ^{238}U composition. However, because the RU here is primarily ^{233}U and the fraction of ^{238}U is very small, this was deemed acceptable. The relative cycle-to-cycle discrepancy in the discharge weight fraction of ^{234}U was 0.01% after 12 recycles. A small amount of TRU is also generated in this stage. A significant fraction of the TRU generated is ^{237}Np, but there is also a small amount of several Pu isotopes and other TRU elements. The makeup vector is taken from stage 1. The postirradiation cooling time is 5 years, and the fabrication time is 2 years. The analysis assumes a 1% loss in reprocessing and 0.2% loss in fabrication.

The ADS in stage 3 is analyzed utilizing the REBUS3/DIF3D tool set in the external source mode.[27–29] The TRU input vector to the ADS burner is a boundary condition from stage 1 and stage 2. The tools utilized are not capable of generating the necessary spallation products (DIF3D is a neutron diffusion code), so the target transmutation is undetermined at this time. The ADS simulation utilized ENDF/B-V cross sections from ATW design studies.[27] The thermal power of the blanket was 840 MW. A 1-GeV proton beam of varying source intensity was utilized to maintain constant blanket power. The source definition used was from ATW design studies. For the purposes of this paper, the beam power was assumed to be 60 MW(electric). The makeup vector is taken from stage 1 and stage 2, in proportion to the relative power share. The peak discharge fast fluence is 3.86 × 10^{23} n/cm^2.

Some other characteristics of the ADS are shown in Table XII. The postirradiation cooling time is 5 years, and the fabrication time is 2 years. The analysis assumes a 1% loss in reprocessing and 0.2% loss in fabrication.

The characteristics of the fuel cycle are summarized in Table XIII. The initial heavy metal mass is calculated via the thermal power of the reactor and the SPD. The units of thermal power are megawatts thermal, and the units of SPD are megawatts per initial heavy metal metric ton (MW/IHMMT). The required natural thorium for stage 1 and stage 2, ^{233}U vector for stage 2, and TRU for the ADS are derived from the reactor physics calculations. The fraction of electricity generated is calculated via an algebraic mass balance of the three stages.

The mass flow data are normalized in MT/GW(electric) · yr for each stage so that the data can be scaled to match a total electricity production of 100 GW(electric) · yr, which is roughly the size of the nuclear fleet in the United States. The electricity generation is 68.5% in stage 1, 23.9% in stage 2, and 7.6% in stage 3. The resource utilization of this fuel cycle could be further improved by optimizing the fissile inventory ratio of stage 2. The selected geometry was utilized because it fits within the form factor of existing PWR designs.

TABLE XII

TRU-Burning ADS Performance

Parameter		Value
Thermal power [MW(thermal)]		840
Capacity factor (%)		90
Multiplication factor	BOC	0.9700
	EOC	0.9333
Specific power density (MW/MT)		279.2
Power peaking factor (BOEC/EOEC)		1.68/1.65
Discharge burnup (GWd/MT)		194.9
Peak fast neutron fluence ($\times 10^{23}$ n/cm^2)		3.86
Heavy metal consumption rate (kg/yr)		284.4
Total fissionable heavy metal mass (kg)		3008.8

VI. COMPARISON OF FUEL CYCLE PERFORMANCE FOR VARIOUS EDS MISSIONS

There are diverse potential applications for deploying EDS in thorium fuel cycles. The examples shown in this paper are intended to demonstrate how an EDS might fill some of these roles, specifically in a breed-and-burn fuel cycle, as a breeder or as a burner. As part of the comprehensive E&S of nuclear fuel cycles,[5] each of these options was compared to a variety of other examples of possible fuel cycles and also to an example intended to represent our present once-through fuel cycle with LEU fuel in critical thermal spectrum reactors. In each fuel cycle, 0.2% of the material was assumed to be lost in fuel fabrication and 1.0% in separations. The assumed capacity factor was 0.9. These loss rate assumptions were standard boundary conditions in the E&S (Ref. 5).

In this paper, the E&S metrics performance from the E&S is reviewed for each of the example EDS missions in thorium fuel cycles: breed and burn, breeder, and burner. This metrics performance is compared to the present

TABLE XIII

Fuel Cycle Characteristics

Parameter	Stage 1	Stage 2	Stage 3
Power [MW(thermal)/MW(electric)]	1500/495	3000/1000	840/216
Fuel form	UO_2/ThO_2	UO_2/ThO_2	TRU-Zr[a]
Capacity factor (%)	90	90	90
Discharge burnup (GWd/MT)	61.7	56.0	194.9
Power density (MW/MT)	17.5	36.6	279.2
Number of batches	3	3	6
Tails (%)	0.25	N/A	N/A
Heavy metal in core (MT)	85.79	81.97	3.01
Charge mass per batch	28.60	27.32	0.50
Residence time (yr)	10.74	4.66	2.13
Cycle length (yr)	3.58	1.55	0.35
Electricity generation [GW(electric) · yr]	68.5	23.9	7.6

[a]Dispersion fuel.

once-through fuel cycle with critical thermal reactors and LEU fuel, and the metrics values are binned as described in the E&S report to accommodate uncertainties in the analyses.

The metrics performance for nuclear waste management, environmental impact, and resource utilization is considered in this paper. The metrics were calculated based on an assumption of 33% thermal efficiency, with the thermal efficiency of ADS stages adjusted to account for the energy requirements of the accelerator. Key values of these metrics are shown in Table XIV. Versus the reference U.S. nuclear fuel cycle, all EDS missions have at least an order of magnitude reduction in the mass of spent nuclear fuel (SNF) and HLW disposed per unit energy generated. This is due to the very high burnup attained in breed-and-burn missions and continuous recycling of material in the breeder and burner missions. Significant HLW will be generated by either an FFH or ADS. One pertinent example is spallation products in an ADS. All of the thorium fuel cycle examples with EDS here significantly improved mass of HLW metrics over the present once-through fuel cycle.

Short-term activity (100 years) is similar for each of the fuel cycles. This metric is dominated by FPs, which are similar per unit energy generation for each fuel cycle. It is notable that the breed-and-burn fuel cycle exhibits a marginal advantage versus the present U.S. fuel cycle as well as the breeder and burner missions. This is due to the very long fuel cycle length assumed in the analysis (53 years), which allows some of the FPs to decay.

Long-term activity is similar for the present U.S. once-through cycle as well as the breed-and-burn FFH. Marginal benefit is seen in the breeder ADS feeding a critical reactor fleet. The fuel cycle that employs the burner ADS is intended to continuously recycle and burn all uranium bred from thorium, as well as higher actinides produced during the fuel cycle, so it is expected that this fuel cycle would have the lowest long-term activity metric value. It is notable that the half-life of ^{233}U is 159200 years, so this is an important contributor to 100000-year activity in some thorium-based fuel cycles and was seen as a key discriminator between thorium-based and uranium-based fuel cycles in the E&S (Ref. 5). The mass of depleted uranium (DU), RU, and RTh disposed per unit energy generated is relevant to the fuel cycles with uranium enrichment, which both generate large amounts of DU.

The volume of low-level waste (LLW) is higher for the FFH EDS because that tritium ICF fuel must be produced and deployed at an industrial scale. In addition, the values are higher for the fuel cycles with ADS versus the reference once-through fuel cycle with LEU because these are continuous recycle fuel cycles and because the continuous processing of fuel material will generate additional LLW.

The environmental impact metrics (land use, water use, carbon emission, and radiological exposure) are not significant discriminators for any of these fuel cycles. One source of additional water use in a fuel cycle with an EDS is the additional cooling required for an ICF device or spallation target. Fuel cycles without enrichment, for example, the breed-and-burn once-through EDS or breeder EDS, have lower carbon emission because mining and enrichment operations are significantly curtailed.

Resource utilization is a key discriminator: The breed-and-burn EDS achieves very high burnup with only natural thorium feed, and the breeder EDS achieves similarly high resource utilization because enrichment is not required and all fuel is continuously recycled. The fuel cycles with enrichment, namely, the reference once-through fuel cycle with LEU-fueled critical reactors and the fuel cycle employing enriched uranium support and a burner EDS, exhibit resource utilization two orders of magnitude worse than the options without enrichment. The total natural uranium required by the fuel cycle with the burner EDS is marginally improved relative to the current once-through fuel cycle. However, this incremental improvement is not considered significant relative to the transformational two-order-of-magnitude improvement observed in the fuel cycles without enriched uranium.

VII. CHALLENGES FACING EDSs

There are many potential challenges facing EDSs. In all blanket designs the effects of higher-energy neutrons (higher than the traditional fast reactor neutrons) need to be understood. In addition, in ADS there is the possibility of protons, created during the spallation process, escaping the target volume and irradiating the closest fuel elements. The high-energy neutron fluence and the fluence of leakage protons on the fuel elements need to be resolved, and the impact on the fuel and structural material survivability needs to be determined.

There are no regulatory guidelines that cover the licensing of proton accelerators and fusion plants with the specific aim of producing fissile material and potentially generating electricity. EDS will be subject to the full regulatory scrutiny. In addition, there are new issues related to safety that need to be addressed. These include various failures of the external neutron source. Some specific examples for an ADS include beam collapse or failure of the rastering system resulting in the full power of the beam being focused down to a very small beam

TABLE XIV

Metrics Performance of the Fuel Cycles in This Paper Versus the Present U.S. Nuclear Fuel Cycle*

Criterion	Metrics	Once-Through LWR (LEU)	Breed-and-Burn FFH (Th)	ADS (Th) Supporting LWR (Th/U3) Fleet	LWR (Th/U3 with LEU Support) Fleet with Pu/MA ADS Burner
Nuclear Waste Management	Mass of SNF + HLW disposed [MT/GW(electric) · yr]	21.92/E	1.62/A	1.47/A	1.40/A
	Activity of SNF + HLW (at 100 years) [MCi/GW(electric) · yr]	1.34E+06/C	9.09E+05/B	1.49E+06/C	1.30E+06/C
	Activity of SNF + HLW (at 100000 years) [10^{-4} MCi/GW(electric) · yr]	1.65E+03/C	1.69E+03/C	1.28E+03/B	7.63E+02/B
	Mass of DU + RU + RTh disposed [MT/GW(electric) · yr]	166.67/E	0.00/A	0.00/A	114.17/D
	Volume of LLW [m^3/GW(electric) · yr]	398.84/C	826.47/D	592.98/C	677.49/D
Environmental Impact	Land use per energy generated [km^2/GW(electric) · yr]	0.175/B	0.109/B	0.094/A	0.130/B
	Water use per energy generated [Ml/GW(electric) · yr]	23891/B	33640/C	27306/B	24623/B
	Carbon emission—CO_2 released per energy generated [kMT CO_2/GW(electric) · yr]	44.1/B	25.4/A	29.4/A	41.7/B
	Radiological exposure [mSv/GW(electric) · yr]	1.10/B	2.93/B	1.49/B	1.14/B
Resource Utilization	Natural uranium required per energy generated [MT/GW(electric) · yr]	188.63/D	0.00/A	0.00/A	114.85/C
	Natural thorium required per energy generated [MT/GW(electric) · yr]	0.00/A	1.62/A	1.51/A	0.75/A

*Values from Ref. 5.

spot, failure to shut off the beam when problems in the target/blanket are detected, containment boundary issues due to the large penetration into the blanket for the beam and target, etc.

The development and deployment risks of any of these thorium fuel cycles with EDS should be considered high. No such combination has ever been demonstrated, licensed, constructed, or operated. All of the EDSs proposed here have a low technology readiness level, are highly novel, and utilize an unconventional fuel, implying additional risk. In all of these fuel cycles, material damage issues due to high-energy neutrons and protons will play a role in determining the life of various structural components and the fuel. The potential damage mechanisms will be different from the current fast reactor experience. Both FFHs and ADSs have unique challenges that would require significant R&D investments and the demonstrated incentive to make those investments.

Fuel cycle analysis data were generated for the once-through thorium fuel cycle in a fast spectrum system. FFHs represent an enabling path for such cycle because the fusion source could potentially provide abundant neutrons at no net electricity cost. It was also assumed that the blanket uses solid fuel (TRISO particles carried in carbon pebbles) and operates in a batch mode. From start-up with no fissile content to discharge when ~75% of the initial heavy metal has been consumed, the neutron energy spectrum varies considerably.

VIII. SUMMARY

There are various potential roles for an EDS in thorium fuel cycles. This paper explores three example roles for the FFH and ADS:

1. a once-through fuel cycle with breed-and-burn of thorium fuel in a thermal or fast neutron spectrum EDS to very high burnup without enrichment (breed-and-burn mission)
2. a fuel cycle with pure thorium natural resource feed where all materials other than FPs are recycled and a fast neutron spectrum EDS is used to breed makeup fuel (mostly ^{233}U in thorium) to support a fleet of critical reactors (breeder mission)
3. a fuel cycle where LEU is used to support a fleet of critical reactors with LEU and Th fuel, all materials other than FPs are recycled, and TRU actinides (primarily from the LEU support but with a small fraction from thorium fuel) are burned in a fast neutron spectrum EDS (burner mission).

All of these fuel cycles offer some benefits to the present once-through U.S. fuel cycle with LEU in critical thermal neutron spectrum reactors. Enrichment of uranium leads to poor uranium resource utilization, and the two example thorium fuel cycles without enriched uranium offer significant potential benefits in terms of resource utilization. All of the example thorium fuels cycles here exhibit substantially less volume of HLW than the present U.S. nuclear fuel cycle due to the relatively high burnup or the fact that the resource materials are recycled continuously (or both, in the case of the EDS burner). Long-term (100000-year) activity metrics are less favorable for the once-through breed-and-burn thorium fuel cycle due partially to the long half-life of ^{233}U. Most of these example fuel cycles exhibit substantially higher masses of LLW due to either continuous recycling of fuel materials or production of fusion target fuel.

Significant R&D challenges face thorium fuel cycles with EDS. Realization of these concepts would require high levels of sustained R&D investment to potentially achieve possible benefits.

Acknowledgments

Some analyses described in this paper were supported by the DOE Office of Nuclear Energy Fuel Cycle Options Campaign. This manuscript has been authored by employees of Brookhaven Science Associates LLC under contract DE-AC02-98CH10886 and of Oak Ridge National Laboratory, managed by UT-Battelle LLC under contract DE-AC05-00OR22725 with the DOE.

Dedication

This paper is dedicated to the memory of our deceased colleague, Dr. Hans Ludewig, of Brookhaven National Laboratory. Dr. Ludewig contributed significantly to many studies of externally driven systems throughout his distinguished research career.

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