 29

UCRL-JC-134396
PREPRINT

The Secure, Transportable, Autonomous Reactor System

N.W. Brown, J.A. Hassberger, C. Smith, M. Carelli, E. Greenspan,
K.L. Peddicord, K. Stroh, D.C. Wade and R.N. Hill

This paper was prepared for submittal to the
American Nuclear Society
International Conference on Future Nuclear Systems
Jackson Hole, WY
August 29-September 2, 1999

May 27, 1999

Lawrence Livermore National Laboratory

DISCLAIMER

UCRL JC 134396

THE SECURE, TRANSPORTABLE, AUTONOMOUS REACTOR SYSTEM[a]

Neil W. Brown, James A. Hassberger, Craig Smith
Lawrence Livermore National Laboratory
7000 East Avenue,
Livermore, CA 94550
(925) 424-4019

Mario Carelli
Westinghouse Science & Technology Center
Westinghouse Electric Corporation
1310 Beulah Road
Pittsburgh, PA, 15235
(412) 256-1042

Ehud Greenspan
University of California, Berkeley
4107 Etcheverry
Berkeley, CA, 94720-1730
(510) 643-9983

K. Lee Peddicord
Texas A&M University
129 Zachry Engineering Center
College Station, TX, 77843-3133
(409) 845-6443

Kenneth Stroh
Los Alamos National Laboratory
P.O. Box 1663
Los Alamos, NM 87545
(505) 667-7933

David C. Wade, Robert N. Hill
Argonne National Laboratory
9700 South Cass Ave.
Argonne, IL, 60439
(630) 252-0000

ABSTRACT

The Secure, Transportable, Autonomous Reactor (STAR) system is a development architecture for implementing a small nuclear power system, specifically aimed at meeting the growing energy needs of much of the developing world. It simultaneously provides very high standards for safety, proliferation resistance, ease and economy of installation, operation, and ultimate disposition. The STAR system accomplishes these objectives through a combination of modular design, factory manufacture, long lifetime without refueling, autonomous control, and high reliability.

I. INTRODUCTION AND BACKGROUND

A new concept for a nuclear power system to specifically address the needs of much of the developing world was first presented at Global 97.[1] Lawrence Livermore National Laboratory (LLNL), with the support of Argonne National Laboratory (ANL), Los Alamos National Laboratory (LANL), Westinghouse Science and Technology Center, the University of California at Berkeley (UCB), Texas A&M University (TAMU), and the Massachusetts Institute of Technology (MIT), has further evaluated this nuclear power system concept and named it the Secure, Transportable, Autonomous Reactor (STAR). It uses small nuclear power stations with the aim of reducing the proliferation concern associated with the introduction of nuclear power in developing countries.

Energy demand in the developing countries is increasing concurrent with their growing populations and economies. This demand will be met by a combination of fossil, renewable and nuclear power sources. However, the current mix of these energy sources does not, and cannot, satisfy all the demands for new energy. Fossil-fueled sources add to the accumulation of greenhouse gases and are of limited supply in many areas. Renewables, including hydro, are in limited supply, or are at an insufficient level of technical readiness to play a major role in meeting energy demands. Besides the growing demand for electricity, significant energy growth is also anticipated for desalination and district heating systems. Today's nuclear power systems are too large and expensive for many areas, especially those lacking the necessary institutional infrastructure, expertise, capital, or large established power grid. Small nuclear power systems could provide an ideal

[a] This work was performed under the auspices of the U.S. Department of Energy by Lawrence Livermore National Laboratory under contract no. W-7405-Eng-48.

solution for meeting the needs of many regions for which the current systems are inadequate.

Prior work[1,2] showed that a number of unique and distinguishing innovations are needed to meet the energy demands of most of the world's developing regions and address growing nuclear proliferation concerns. These technical innovations form much of the basis underlying the STAR concept and include:

- Eliminating on-site refueling and fuel access;
- Incorporating a systems approach to nuclear energy supply and infrastructure design, with all aspects of equipment life, fuel and waste cycles included;
- Small unit size enabling transportability;
- Replaceable standardized modular design;
- Resilient and robust design concepts leading to large safety margins, high reliability, and reduced maintenance;
- Simplicity in operation with reliance on autonomous control and remote monitoring;
- Waste minimization and waste form optimization.

The combination of these features, including small unit size (on the order of 300 MWt), makes such a system particularly attractive for siting in regions lacking critical infrastructures, such as large power grids or highly qualified construction and operating personnel.

II. STAR SYSTEM

The STAR approach requires a substantially new look at the architecture of nuclear power and presents a unique opportunity to optimize the overall system design. The STAR approach considers the entire system, including the manufacturing, operation, and ultimate disposition phases of the system life cycle. The safety, proliferation, environmental, and economic implications of the system are carefully addressed in each phase of the system and fuel cycle development as well as for the whole. This approach will help ensure that specific features are not unknowingly optimized at the expense of the overall system. Such an approach also offers promise for further reducing life cycle costs, addressing waste management and spent fuel disposition issues and concerns, and improving the balance of attention to nonproliferation issues across several segments of the fuel cycle and life cycle.

While the STAR approach focuses on small nuclear power systems for application in developing countries, many of the challenges and technology solutions required to meet these challenges will have far wider application. For example, development of the long-life fuels needed to achieve the STAR goals will also apply to the development of high-burnup fuels for more conventional nuclear power systems.

A. Research Requirements

The STAR system concept employs a number of innovative design features, the realization of which require research to demonstrate feasibility and ultimately to produce a practical system. These design features support a system well-suited to meeting the needs of the developing world. Table 1 identifies the major design features of the STAR system, lists the objectives they support, and outlines the primary advantages and issues associated with them.

These design features define especially demanding requirements, but they are more likely achievable with systems designed at the relatively low power level used in the STAR approach than with larger systems. Moreover, implementing these features will increase the security, safety, and public acceptance of the expanded use of nuclear power within developing countries. One of the major challenges will be to accomplish these goals with an economically viable system. The traditional approach to nuclear power economics has been economy of scale. The STAR approaches the economic issues from a different perspective: it relies on the economics of mass production, coupled with cost savings achieved from dramatically reduced on-site installation, interest, operation and decommissioning costs, reduced site infrastructure and licensing costs spread over a large production run. The approach also reduces project uncertainties by shortening the time required between placing an order and final commissioning.

B. Concept Development and Down Select

The initial phase of the STAR development would refine the four competing reactor concepts, leading to identification of the ultimately selected concept(s) at the end of that period. During this period, the system requirements and overall infrastructure architecture are developed, the scientific and technical feasibility of the architecture established, the most promising approach to realizing that architecture defined, and the R&D required to implement that approach identified. This first phase concludes with down-selection to a preferred system and possibly a second.

Table 1. Design Features of the STAR system

Design Feature	Supports Objectives	Advantages	Issues
Proliferation resistance Costs	Suitability for developing countries	Eliminates access to fissile materials Reduces local infrastructure Reduces containment size Eliminates much ancillary equipment Eliminates large PV penetrations Potential for reduced Safeguards and Security (S&S)	Requires long-life core No periodic core access for maintenance Need, potential for remote monitoring to replace traditional S&S Required enrichments
Delivered pre-assembled and pre-fueled	Proliferation resistance Costs	Minimizes site construction time, interest and uncertainties No access to fresh fuel	Transportation safety Mass, size limitations
Passively safe	Safety	Minimizes local infrastructure Minimizes potential for operator error Opportunity to improve regulatory practices	Demonstration Impact on regulations
Highly autonomous operation Minimized maintenance	Safety Costs	Minimizes local infrastructure Minimizes potential for operator error Facilitates innovative safeguards measures	How to achieve Reliability requirements and implications Need new approaches to reliability analysis Balance between automation (requiring substantial instrumentation and controls (I&C) and autonomous operations Defining "minimum-necessary" I&C
Replacement at end of life (EOL) Disposal "designed-in"	Waste management Costs	No/minimum site-generated wastes Optimized waste management integrated into design	Transportation safety Ultimate disposition Legal requirements of supplier country Ownership of wastes
Low power (50–150 MWe)	Costs Safety	Meets needs of many regions Facilitates safety, transportation designs Minimizes capital costs	Requires new economic model, manufacturing approach
Long life	Costs	Improves overall economics	What can be achieved Fuel, structural and control materials

The process for the down-selection involves three essential elements:

1. Working with a group of stakeholders representing technical experts, potential clients and others, the system goals, requirements and evaluation attributes will be established, and the overall system concept developed. Simultaneously, the down-selection decision tools that will be used to support the down selection will be developed.

2. A set of system and reactor alternative approaches will be developed and associated technical feasibility issues identified along with the R&D required for resolution.

3. The down-selection methods and the results of the feasibility assessments will be used to iteratively refine the alternatives, goals and requirements, evaluation criteria, and down-selection method. The refined system requirements and the evaluation methods and criteria will be used as evaluation tools to assist in determining the ultimately selected concept.

Each of the reactor concept development efforts would deal with the concept-specific feasibility issues identified through a technology road-mapping process. The four reactor development efforts will be continually kept aware of requirements and selection criteria development.

III. STAR—Light Water (LW)

Two reactor studies by Westinghouse (ES^2) and POLIMI (NILUS),[3–6] will be combined to form the basis for the development of the STAR-LW reactor. The Natural Circulation Integrated Layout Ultimate Safety (NILUS) reactor is a full natural circulation reactor originally designed for a 1000 MWt size (300 MWe), but which can very easily be scaled down to the 500 MWt range typical of the STAR-LW. The ES^2 focused exclusively on the core design while NILUS has also dedicated a large effort to characterizing the reactor systems, which were not addressed by ES^2. The Westinghouse effort had as its main objective the characterization of a core design capable of satisfying proliferation resistance requirements, which were not a recognized objective for NILUS. The objective of natural circulation/passive safety was recognized by both designs, but to a larger degree by NILUS which features full natural circulation. NILUS has adopted a less aggressive approach to design simplification and elimination of loss of coolant accidents (LOCAs).

It is envisioned that STAR-LW will be developed from the synthesis of the two designs, then modified as necessary to satisfy the STAR objectives. Since NILUS has reached a more mature stage of development, its solutions will provide a fallback position for the more aggressive features of STAR-LW, should they prove infeasible.

Two early Westinghouse designs having features germane to this effort, are the small size secure military power plant (SMPP) designed in the early 1980s for the Early Warning System and the light water breeder reactor (LWBR) which operated successfully in the late 1970s and early 1980s.

The SMPP design was based upon innovative applications of standard pressurized water reactor (PWR) for application to the Early Warning System, which required out-of-the-way, difficult-to-access siting, such as the Canadian tundra in the Arctic Circle.

The design objectives were very similar to those for STAR, with the exception that there was no requirement of proliferation resistance. The SMPP conceptual design, utilizing standard fuel assemblies with a 3.48% enrichment, had a cycle length of 3 years with a 90% capacity factor. The same core, with 5.0% enriched fuel, was shown capable of a fuel lifetime of 6 years

An attractive feature of the SMPP was its simplified control system. Reactivity control is provided by burnable poison absorbers and control rods, without the use of soluble boron and its accompanying complex concentration/dilution system. The absence of absorber in the coolant also meant that the core had a large negative power coefficient, an important inherent safety feature. In order to accommodate large peaking factors, it was necessary to reduce the average linear power density from 5.4 to 2.1 kw/ft, reducing the peak value from 12 to 9 kw/ft relative to a standard PWR. The fuel enrichment and burnable poison loading were optimized to meet lifetime requirements, while at the same time minimizing the swing in excess reactivity over lifetime. The total swing was only 3.5%, including over 2% for equilibrium xenon, simplifying control rod requirements.

Conventional light water reactors (LWRs) with a moderator to fuel volume fraction (Vm/Vf) of about 2, and a pitch-to-diameter ratio (p/d) of about 1.33, have a conversion ratio (CR) between 0.5 and 0.6. Calculations showed that this could be increased to much higher values, even exceeding 1, by hardening the neutron spectrum and changing the fissile material mix. Spectrum hardening can be done by reducing the water content in the core using a triangular pitch lattice, with the rods then bundled into hexagonal shaped cans. The Vm/Vf fraction in the cores examined generally varied from about 0.3 to 0.7, with corresponding p/d of 1.05 to 1.2. To attain a CR greater than 1, it was also necessary to utilize MOX fuel, with fissile plutonium enrichments up to 10%, or mixed enriched fuel (U-235 and Pu-239). Another effective way of achieving high CR is the use of a Thorium-based cycle. In fact, Westinghouse designed, built, and successfully operated from 1977 to 1982 a demonstration light water breeder reactor (LWBR) based upon the Th-U-233 fuel cycle, at Shippingport, PA.

The objective of most of these studies was to attain a CR greater than 1, i.e., a breeder, or at least very close to 1, to maximize fuel utilization. The CR in this case needs to be enough to limit the reactivity loss associated with the burning of the fissile fuel. An early example of this type of reactor was the large power reactor (LPR), a Th-U-233 seed blanket core designed by Westinghouse in the 1960s. A conversion ratio of about 0.9 enabled this reactor to have a reactivity lifetime of 10 years without refueling. This situation has changed dramatically, with current LWR (oxide) fuel rod designs able to reach burnups in excess of 60,000 MWd/t of heavy metal. Much higher burnups (up to 200,000 MWd/t) have been attained in fast reactors (FFTF, Rapsodie) by the mixed oxide fuel/stainless steel cladding fuel rods.

In addition to long cycle lifetime, there are other significant neutronic advantages associated with hard neutron spectrum of tight-lattice reactors. This includes low reactivity coefficients and excess reactivity requirements, simplifying control requirements. The major difficulty identified is the possibility of a positive void coefficient under certain conditions. This can be avoided by changes in the fissile fuel mix (Pu-239 and U-235), a decrease in the height-to-diameter ratio (H/D) of the core, or the use of heterogeneous core configurations. Another advantage is the higher volumetric power density at the same linear heating rate permitted by the more densely packed fuel rods, resulting in a more compact core.

A study by Kraftwerk Union[7] in 1988 focused on the development of a tight-lattice, high-conversion PWR designed to replace existing thermal PWRs with minimal plant changes, development costs, and licensing impact. The optimal design achieved had a homogeneous configuration with a P/D of 1.12, Vm/Vf of 0.52, and a conversion ratio of about 0.9. The power density was 150 W/cm^3, with an average linear heating rate of 4.9 kW/ft. MOX fuel was utilized, with an enrichment of 7.5%.

Actual operating experience in full natural circulation, light water reactors has been accumulated in the latest generation of Russian submarines. Fragmentary information indicates that some degree of boiling, possibly quite substantial, was allowed by design; special configuration fuel (of a multi-lobe design) was adopted to increase the heat transfer area. It is an objective of the STAR development to seek cooperation with such Russian organizations as the Kurchatov Institute (Moscow), the Special Design Bureau of Machine Building (Nizniy Novgorod) and the Bochvar Institute of Inorganic Materials (Moscow), which have been involved in the nuclear submarine development.

IV. STAR—Liquid Metal (LM)

For STAR-LM, the design approach being developed to achieve the objectives of the STAR system concept includes the following:

- Long refueling interval will be achieved using *demonstrated* high density, high burnup fast reactor fuel at a reduced core power density (derated configuration).
- For the derated power density, natural circulation cooling will be relied on to the maximum extent possible—this should improve the reliability of the heat removal, and reduce cost by mitigating forced convection (pumping) and emergency cooling system requirements.
- Natural circulation eliminates pumping power as an economic/design constraint. This makes heavy metals a viable design option. Because these heavy metals do not react with either air or water, the reactor will be further simplified by removing the intermediate heat transport loop employed in conventional (sodium-cooled) liquid metal reactors.
- The reactor will achieve passive safety through a low-pressure, pool-type primary circuit, natural convection decay heat removal, inherent reactivity feedback, and minimal excess reactivity.
- Internal conversion in the hard energy spectrum will be exploited to minimize burnup reactivity losses at low enrichment levels.
- The favorable thermal feedback characteristics of a small, fast reactor will be exploited to make the plant operation simpler and more autonomous.

To fully evaluate the merits of adopting heavy metal coolant, it is first necessary to explore how the mission objectives can be met using standard (sodium-cooled) technology. Thus, the STAR-LM development activities are divided into three phases. First, options would be developed which meet the STAR design criteria using conventional fast reactor technology (i.e., sodium coolant). The potential for natural circulation cooling would be investigated, system simplifications would be pursued, and the safety performance would be verified. In the second phase, design options utilizing heavy liquid-metal coolant would be explored. For this study, the focus would be on lead-bismuth alloy (Pb-Bi) as utilized in Russian heavy-metal cooled systems.[8,9] A consistent set of heat removal and safety performance analyses would be performed. In the third and final phase, a specific STAR-LM concept would be chosen based on the performance trends identified in the first two phases. This system would be developed and evaluated in greater detail to support down-selection among the various STAR reactor alternatives. This conceptual design would include the reactor plant layout, core design, heat removal systems, safety assessment, and manufacturability/modularization concepts for the proposed system.

A. Power Density Derating

The limiting constraint on the STAR-LM design is likely to be the requirement for a refueling interval of at least 15 years. For a fast reactor, fuel lifetime is not typically constrained by reactivity considerations, but is limited by damage to the fuel form as quantified by discharge burnup and discharge fluence limits. The strategies available for achieving lengthened refueling intervals can be understood by breaking the discharge burnup into several factors:

$$\text{Discharge Burnup (MWd / kg)} = \frac{\text{Power Density (MW / }\ell\text{ - core)}}{\text{Fuel Density (kg - fuel / }\ell\text{ - core)}} \text{Exposure (Days)}$$

or, rearranging,

Exposure (days) =

$$\text{Discharge Burnup (MWd / kg)} \frac{\text{Fuel Density (kg - fuel / } \ell \text{ - core)}}{\text{Power Density (MW / } \ell \text{ - core)}}$$

To achieve the design target of 15 years exposure, clearly it is desirable to use a fuel form with a high discharge burnup limit and design the core for a high fuel density. The upper limit of the burnup is dictated by the fuels technology; the upper limit of the fuel density is dictated by the fuel physical form and cooling considerations. Given the feasible limits (with current technology) for these two characteristics, the only way to further increase the exposure time is to derate the core power density (MW/liter).

Liquid metal cooled fast reactor technology offers several immediate advantages in addressing these three attributes. First, high burnup capability fuels have been developed and extensively used in fast reactors. Both MOX fuel pins and UPuZr metal alloy fuel pins in HT-9 ferritic cladding have achieved 200 MWd/kg peak discharge burnup and are qualified and demonstrated for 100 MWd/ kg average and 150 MWd/kg peak discharge burnup. Secondly, the fuel density (kgfuel/liter-core) in a fast reactor is comparatively high. The excellent heat removal capability of the liquid metal coolant allows tight pin spacing with fuel volume fractions of roughly 40%. In addition, metal and nitride fuel forms utilized for fast reactors have a fuel density significantly higher than conventional oxide fuel. Finally, the (non-derated) power density of fast reactors is in the 600 to 800 kW/liter range, compared to ~200 kW/liter for LWRs and < 100 kW/liter for gas reactors. Thus, the fast system can be derated by a factor of 3–8 to achieve the extended refueling interval, and still have a size comparable to alternative reactor systems at their full power density (and short refueling interval).

Typical fast reactor design parameters can be utilized to estimate the derating requirement. Assuming a 15-year exposure at an 80% capacity factor, roughly 4,400 days of exposure are required. A burnup limit of 100 MWd/kg as demonstrated by conventional fast reactor fuels is assumed. The specific fuel loading of 4.7 kgfuel/liter is calculated based on a fuel volume fraction of 1/3 with fuel density of 14.1 g/cc (high density metal fuel). Using the above formula, the required power density for such a system is ~100 kW/liter. This is a significant power derating compared to conventional liquid metal cooled reactor system.

B. Performance Impact of Heavy Liquid Metal Coolant

Preliminary studies were performed to compare lead-cooled and sodium-cooled core performance. For a compact, mid-size (900 MWt) sodium-cooled heterogeneous fast reactor core design, the sodium coolant was replaced with a lead-magnesium eutectic alloy. The driver fuel enrichment was adjusted to assure end-of-cycle criticality, and all other aspects of the core material and geometry configuration were retained. Five different assembly designs with pitch-to-diameter ratio (P/D) ranging from 1.18 to 1.54 and pin size from 0.25 to 0.32 inches were analyzed using both sodium and lead coolant. As discussed in the previous section, the fuel lifetime in this design is only 1,500 days; thus, significant derating from the 375 kW/liter high power density would be required to achieve the long-life target.

The results indicate that a better neutron economy is achieved in the lead-cooled case, particularly in the loose lattice case where the leakage fraction is reduced from 27% (sodium) to 22% (lead). This leads to a reduced enrichment requirement (by 2-2.5% fissile content). The reduced enrichment is advantageous because additional fertile material can be introduced which decreases the burnup reactivity swing by ~0.3%Δk. The reduced enrichment is also beneficial for proliferation considerations since the fissile content of the reactor fuel is reduced. However, the reduced enrichment leads to higher fluence in the lead cooled cases; for the loose lattice, the peak fast fluence increases from 2.5 to 3.0 x 10^{23} n/cm^2. This may be a concern if the discharge exposure of the STAR-LM concept is fluence limited. Power peaking is also slightly worse for the lead case; this is attributable in part to the fact that the core configuration was optimized using sodium coolant.

For the same configurations, whole core reactivity feedback coefficients were evaluated to assess the potential impact of the lead coolant on safety performance. The reduced neutron leakage in the lead cooled system yields a coolant void significantly more negative for the lead coolant (by 1% Δk/k or more). However, the reduced neutron leakage also reduces the magnitude of the expansion coefficients which give favorable reactivity feedback in most transient scenarios. In addition, the Doppler coefficient is reduced by ~10% because of the harder spectrum in the lead cooled system.

In addition to the above, UCB and ANL have developed a liquid metal cooled concept called the encapsulated nuclear heat source (ENHS)[10] that will be investigated as a very innovative alternative to the approach discussed above. The single most unique feature of this concept is that the fission-generated heat is transferred from the primary coolant to the secondary coolant through the reactor vessel wall, completely eliminating through-vessel fluid or mechanical connections. This enables the reactor to remain sealed throughout its lifetime. The reactor module and the steam generator modules can be easily installed and replaced. The combination of a reactor module, that is not mechanically connected to other components of the power plant and includes a long-life core, is like a "nuclear battery." The ENHS opens new possibilities for the design, fabrication, construction, operation, maintenance and refueling of nuclear power plants. This concept possesses some major challenges but may lead to the best approach for the STAR-LM.

V. STAR—Gas (G)

The approach to STAR-G exploits demonstrated high-temperature gas-cooled-reactor (HTGR) attributes with innovative features and operating strategies. Basic gas-cooled-system features include the use of an inert, single-phase, neutronically transparent gaseous coolant. These concepts also rely on the well-developed TRISO-coated particle technology for both fuel and fertile material, retaining fission products at the microsphere level. Inherent safety results from passive system features, ensuring that fuel temperatures in foreseeable accidents do not exceed the failure threshold of these coatings. An all-refractory reactor core is envisioned where the most temperature-limiting component is the fuel. This would prevent propagating failures that can exacerbate accident conditions and minimize the "investment risk."

An ultra-long-life core is based on a high-burnup uranium-thorium fuel cycle, considering the simultaneous maximizing of both the production and in situ utilization of bred fissile U-233. Sufficient quantities of separated U-233 will not be available when systems are introduced, thus necessitating initial fueling with fissile U-235 (the feed uranium enrichment likely limited to 19.9% to remain consistent with current United States (U.S.) practice).

Although all design choices will be open during the development of the design, strawman concepts are presented here to illustrate the approach. The presentation of these "strawman" concepts is not intended to predetermine the system characteristics, but to present discussion of some of the major challenges and offer some potential solutions. All design characteristics and tradeoffs will be addressed during the preconceptual design phase.

The strawman reactor concepts result from a selective blending of a broad range of design concepts and operating schemes developed since the birth of the nuclear age. Although no calculations have been performed on these concepts, all features proposed are either within the experience base of systems built and operated, or derived from systems that have been subject to level of design analysis giving adequate confidence in the validity of the basic assumptions. These concepts have the potential to realize an inherently safe, economical, environmentally friendly, and autonomously operable low-power system (50–100 MWe), that can operate without external refueling for 15 full-power years.

Up to some to-be-determined power level, an innovative fixed-core design is promising. Such a design would utilize features pioneered in the mid-1980s at LANL for small, inherently safe, autonomous gas-cooled reactors for defense applications and would combine features from the Rover/nuclear engine for rocket vehicle applications (NERVA) nuclear rocket program and nuclear process heat program. This concept utilizes an as-extruded (no machining), fully graphitized sleeve with finely divided cooling features to encase the coated particle fuel. Fuel particles in a center cavity are bonded together with a binder as in conventional fuel compacts. Geometry and design features cause the fuel to run much cooler than in a conventional prismatic-block HTGR. Further improvements could be achieved through full graphitization of the binder, which markedly increases the thermal conductivity of the fueled body. Increasing the temperature capability of the coated particle, which results from replacing the silicon carbide layer with zirconium carbide, could enable full graphitization of the binder. Higher temperature capability in this advanced coated particle would give the designer additional flexibility in achieving project requirements. The fueled sleeves are placed in holes in solid moderator blocks. (These could be unmachined, isostatically molded blocks as developed for the German prismatic-block HTGR, which would be markedly cheaper to manufacture.) Cross-block effective conductivity for a conduction cooldown is increased by these features, which, together with the lower operating temperature of the fuel and moderator, allow a cylindrical reactor core to be as safe as an equivalent-power, but larger, annular reactor core. The ability to separate the fueled sleeves from the bulk of the moderator offers significant benefits in fuel fabrication and ultimately in permanent disposal. The major materials challenges in an ultra-long-lived fixed-core design are the neutron fluence (managing net graphite expansion), the coated particle fluence, and the time at temperature. Additional design and safety issues arise from the management of core reactivity over the operating lifetime.

An alternate concept for higher power levels is a pebble-bed core with unique design features and fuel-management strategies. Pebble-bed cores are formed as a random bed of small, spherical refractory fuel elements cooled by pressurized gas flowing in the interstitial spaces. Pebbles are moved in and out of the active-core region to achieve the long interval between external refuelings (a pebble bed can be designed to flow in a reactor vessel much as sand flows through an hourglass). The basic strategy features recirculation of the pebbles through the bed (as in the German reactors), with real-time evaluation of each discharged pebble (via gamma counting and/or active neutron interrogation, weighing, and size gauging) combined with logic to determine if the pebble should be reinserted or sent to spent-fuel holding. Fresh fuel is added as needed from fresh-fuel holding. (These holding volumes are sealed and monitored remotely to prevent diversion.) Just-in-time fuel management maintains hot criticality (with a margin for power maneuvering). Elimination of large excess reactivity has inherent safety advantages, precludes the need for control absorber insertion at power, and eliminates the associated neutron loss. A seed-blanket fuel management strategy is envisioned where a curtain of fertile pebbles flows down the inner edge of the radial reflector, taking advantage of the neutron leakage from the core to breed U-233, while residing in a low-flux region that minimizes parasitic absorption in fissile precursor Pa-233. The conversion ratio is maximized by the ex-core time in the fuel cycle, allowing Pa-233 to decay to U-233. This scheme minimizes the fissile content of the fresh fuel, aiding proliferation resistance. Pebble cycling, with many passes through the core, also ensures very high average burnup of the fissile isotopes. As the fed U-235 burns out and the bred ^{233}U and fission-product absorbers build up, there may be advantages in hardening the neutron spectrum. Spectrum manipulation is accomplished by introducing (or removing) pebbles with different moderator-to-heavy-metal ratios and by introducing (or removing) unfueled moderator pebbles.

Although the focus of the proposed effort is necessarily on the reactor core, a sufficient power system design must be completed to determine requirements for reactor power, gas temperatures, and other constraints imposed by the balance of the plant. Such features also affect the potential accident scenarios to be considered in the design. STAR-G is assumed to utilize a closed-cycle gas turbine with magnetic bearings, with the generator submerged inside the reactor pressure boundary. This approach offers high thermal efficiency, which reduces reactor thermal power, eliminates the potential for bearing-lubricant ingress into the primary coolant, and eliminates the historically troublesome high-temperature steam generator. The operator-friendly characteristics of modest power-density reactors with large fixed heat sinks will facilitate autonomous operation. (LANL designed the Compact Nuclear Power Source in the mid-1980s, which was intended for unattended operation in the Arctic under fault-tolerant computer control.) Advanced gas-cooled reactors (AGRs) in Great Britain have even demonstrated a degree of load following. This behavior should result in predictable plant response and allow for relatively simple and slow-acting controls and instrumentation.

VI. STAR—Molten Salt (MS)

The molten salt reactor technology was one of the options considered in the initial LLNL study. The literature on molten salt reactor (MSR) concepts was researched, with the objectives of developing a concept and identifying constraints to the use of this technology in a STAR System. The viability of the MSR was continually challenged as the STAR team participated in further development of each of the concepts.

Graphite lifetime was the life-limiting factor in early design studies, but by increasing the core size (and therefore reducing the neutron flux) the designers were able to reduce the graphite damage and achieve a thirty-year design life. This approach was also used by a Japanese team in their proposal for a small MSR with a power level of about 155 Mwe, called FUJI-II.[11] The FUJI-II approach can be innovatively extended to a more compact system that incorporates many of the auxiliary systems within a single vessel.

The combined weight of the FUJI-II reactor itself (vessel, graphite, and fuel) is over three hundred tons. While this is a substantial weight, articles of this size are readily transported around the world. Even though the FUJI-II concept is not nearly as compact or integrated as that required to meet the STAR objectives, and little or no attention was given to the issue of securing the new and spent-fuel materials, the FUJI-II indicates that such an approach may achieve the STAR objectives. It appears reasonable and possible to develop a complete system assembled as shown in Figure 1, which might weigh in the range of 500 to 800 tons, also of transportable size.

One of the innovative approaches planned is the possibility of delivering a fully assembled system, with the fuel already loaded and frozen into the system. A number of issues must be resolved during development, including where the fuel is stored and how it is thawed. If the fuel is frozen in the reactor vessel, nuclear heat can be used to thaw it. If the fuel is stored in the dump tank, then auxiliary heat will be required. A number of other issues must also be resolved, such as chemical stability during the storage and transport periods.

Reactor start-up could consist of a combination of fuel insertion and withdrawal of control rods. It would be desirable to simplify the design by eliminating the control rods. Reactor control over the operating life can be achieved through a negative temperature coefficient and adding negative reactivity (e.g., by adding thorium salts) or positive reactivity (adding U-235 salts). These additions would be made automatically by a supply system sealed within the secondary fuel containment.

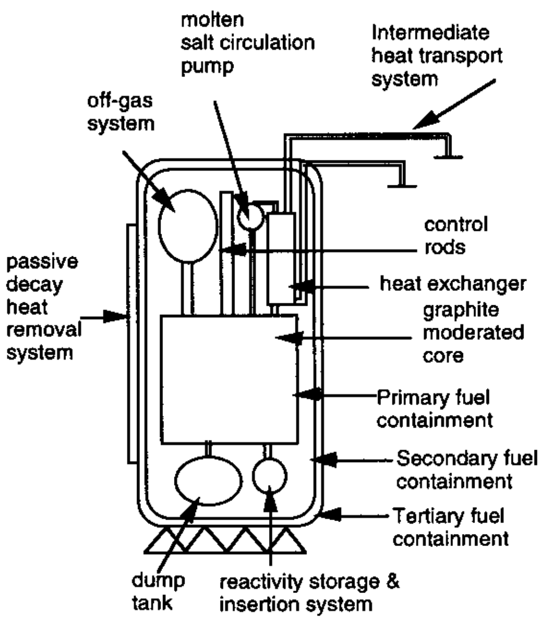

Figure 1. Schematic Concept of STAR-MS

Corrosion is minimized in a molten salt reactor by monitoring and adjusting the chemical reactivity of the salt. Adjustments can be made by altering the fluoride ion concentration in a manner similar to that used to control pH in a water solution. Adding varying amounts salts of the appropriate valence (either F_3 salts or U-235 F_4 salts) will control the chemical reactivity, but this must be done correctly to balance the nuclear reactivity.

Reactor decommissioning at end-of-life is another issue unique to the concept. The preferred approach is to empty the secondary salt from the system, then separate the nuclear assembly, with the fuel in it, from the secondary piping, and remove a large single package for shipping. This is a major challenge and is highly dependent on the size of the system, which is now quite radioactive. However, the molten salt system offers many options for addressing this operation. The fission products of group I (volatiles) would already be in activated charcoal containers. The group III fission products (noble and semi-noble metals that are insoluble in the molten salt) would for the most part be removed and located on filters. An opportunity therefore exists to remove some of the highly radioactive, non-fuel pieces of the plant in more manageable assemblies. Removing the whole assembly is most consistent with the STAR objectives, but also may introduce unnecessary expenses associated with removing large, heavy components, such as the tertiary outer containment, which is not damaged and can be left for reuse. The options associated with end-of-life removal are an important part of the proposed work.

Proliferation resistance is further enhanced by the inherent characteristics of the Th fuel cycle, or fueling with preirradiated or recycled and/or denatured (less than 20% enriched uranium) fuel. The Th/U-233 fuel cycle is considered by security experts to be less attractive for nuclear explosives than the U-235/Pu fuel cycle. Using uranium with less than 20% U-235, although it will produce some Pu, may be preferable to using highly enriched U-235 in the Th, where it is susceptible to chemical separation. Using preirradiated highly enriched uranium is another possibility that may be more practical in the STAR-MS than in other concepts. The molten salt fuel, with its unique handling and fabrication technology, provides an opportunity to reconsider approaches to making fuel material less attractive from a proliferation perspective. In addition, no capability is provided in STAR-MS for removing U or Pu from the enclosed reactor system. The potential clearly exists for achieving inexpensive, secure, remote monitoring of these barriers. A lifetime supply of fissile and fertile fuel would be loaded at startup, with the makeup portion added to the primary molten salt loop, on-line, automatically. This makeup fuel would be contained in the primary system and would be inaccessible, short of shutting down and dismantling the reactor's primary system. Alternatively, lesser amounts of makeup fuel could be included in the primary loop for use at start-up only, and other reactivity control means, such as burnable poisons, leakage, and absorber control rods, would be employed (at the cost of lower conversion ratio).

VII. THE NEXT STEP

The STAR system approach is continuing to be discussed with the U.S. Department of Energy and with those in the international community who have expressed interest. The STAR-LW and the highly innovative concept for the ENHS (using liquid metal coolant) have been selected for further study under the Nuclear Energy Research Initiative (NERI). The other elements of the STAR approach, which are necessary to optimize the selection of a concept that addresses the complete fuel cycle, are not yet funded. We continue to believe that a broad systems approach has an advantage, particularly when seeking a proliferation-resistant nuclear power system for developing countries. We are therefore in the process of restructuring our program plans to recognize the contributions that may come from NERI while seeking funding to implement the full scope of the project.

REFERENCES

1. N. W. Brown, J. A. Hassberger., E. Greenspan, E. Elias, "Proliferation Resistant Fission Energy Systems," *International Conference on Future Nuclear Systems, Global '97*, Yokohama, Japan, October 5–10 (1997).

2. N. W. Brown, J. A. Hassberger, "New Concept of Small Power Reactor Without On-Site Refueling for Non-proliferation," Lawrence Livermore National Laboratory, UCRL-JC-131317. Presented at *International Atomic Energy Agency Advisory Group Meeting on Propulsion Reactor Technologies for Civilian Applications*, Obninsk, Russia, July 20–24 (1998).

3. Lombardi, A. Mazzola, and M. E. Ricotti, "Natural Circulation and Integrated Layout Pressurized Water Reactor," *Proceedings of the 5th International Conference on Nuclear Engineering ICONE-5*, Nice, France, 26–30 May (1997).

4. Lombardi and M. E. Ricotti, "The NILUS Project: Preliminary Study for Medium and Small Size Innovative PWRs—The NILUS-1000," *Proceedings of the 6th International Conference on Nuclear Engineering ICONE-6*, San Diego, USA, May 10–14 (1998).

5. M. E. Ricotti, "The NILUS Project: Design Features of the Small Size, Natural Circulation, Integrated PWR for Heat and Electricity Production," *Proceedings of the 6th International Conference on Nuclear Engineering ICONE-6*, San Diego, USA, May 10–14 (1998).

6. Achilli, G. Cattadori, R. Ferri, A. Cammi, C. Lombardi, and M. E. Ricotti, "An Innovative Start-Up Device for the Passive Heat Removal System of an Integrated-Layout PWR," *Proceedings of the 1st European-Japanese Two-Phase Flow Group Meeting, 36th European Two-Phase Flow Group Meeting*, Portoroz, Slovenia, June 1–5 (1998).

7. Maerkl, C. Goetzmann and H. Moldaschl, "KWU's High Conversion Reactor Concept," *Nucl. Tech. 80*, **65** (1988).

8. R. N. Hill, J. E. Calahan, H. S. Khalil, and D. C. Wade, "Development of Small, Fast Reactor Designs Using Lead-Based Coolant." Submitted to *International Conference on Future Nuclear Systems, GLOBAL '99*, Jackson Hole, WY, Aug. 30–Sept. 2 (1999).

9. Branover, A. Barak, E. Golbraikh, E. Greenspan and S. Lesin, "High-Efficiency Energy Conversion Cycle for Lead Cooled Reactors," *Proc. 9th International Conference on Emerging Nuclear Energy Systems, ICENES'98*, pp. 617-625, Herzlia, Israel, June 28–July 2 (1998).

10. A. S. Boloori, M. Frank, E. Greenspan, E. Hill, D. M. Hutchinson, S. Jones, X. Mahini, M. Nichol, B. H. Park, H. Shimada, N. Stone and S. Wang, "Once-for-Life Fueled, Highly-Modular, Simple, Super-Safe, Pb-Cooled Reactors," Submitted to Submitted to *International Conference on Future Nuclear Systems, GLOBAL '99*, Jackson Hole, WY, Aug. 30–Sept. 2 (1999).

11. Furukawa, N. Nakamura, K. Mitachi, Y. Kato, and K. Komatsu, "Simplified Safe Small Molten-Salt Reactor—"FUJI"—for a Global Measure of Greenhouse Effects," *9th Miami International Congress on Energy and Environment*, Miami Beach, Florida, Dec. 11–13 (1989).