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D.2.2. THE LOS ALAMOS ACCELERATOR DRIVEN TRANSMUTATION OF NUCLEAR WASTE (ATW) CONCEPT DEVELOPMENT OF THE ATW TARGET/BLANKET SYSTEM

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D.2.2.1. THE ATW OBJECTIVE

The objective of the ATW project is to provide a compelling, proliferation-resistant and economically viable alternative to partial reactor burning and geologic storage in the management of nuclear spent fuel and other plutonium bearing materials.

In the ATW concept, highly efficient and robust waste cleanup procedures produce a feed directly suitable for nearly complete subcritical burning, without significant additional waste generation. ATW technology will significantly reduce waste streams from the nuclear power cycle and essentially eliminate excess fissile materials. Plutonium and other hazardous components of nuclear waste will be destroyed to a high degree and with very low process inventories. Demands on permanent repositories will therefore be greatly reduced, and such repositories (containing no fissile plutonium or highly-mobile long-lived fission products) will become more technically feasible and politically acceptable and economically attractive.

The immediate technical goal is to demonstrate the feasibility of effective treatment of spent fuel and other types of plutonium-bearing materials (including residues and excess weapons plutonium), to eliminate fissile plutonium (utilizing it to produce energy and neutrons) and destroy by transmutation certain fission product nuclides of particular concern to long-term repository performance. Partitioning and isolation of plutonium will not be required to accomplish the project's goal. If ATW is to become the preferred option for nuclear waste management in the US, economic viability of the entire ATW-based fuel cycle must also be demonstrated.

The program will rely on the innovative application of proven technologies with emphasis on safety and anti-proliferation features.

For the past several years researchers at Los Alamos have been studying and analyzing a fundamentally different approach to nuclear power and the elimination of nuclear waste. At Los Alamos this approach is called Accelerator Driven Transmutation Technology (ADTT). We feel that ADTT has the potential for mitigating many of the difficulties inherent in present day nuclear power systems including nuclear waste disposal and the buildup of plutonium inventories [1]. In dealing with the problems of nuclear waste and management of excess plutonium, we believe that ATW has the potential to place nuclear power into a regime with significantly reduced waste streams and no accessible plutonium or highly enriched fissile materials.

ATW-based systems can consume nuclear waste (plutonium, minor actinides and long-lived fission products), burning down existing inventories and producing very little new actinide waste, while generating power and can further use the substantial excess neutrons that they are capable of generating to do useful work, such as transmute long-lived fission products or reconstitute uranium or thorium based nuclear fuel for further use in reactors. ATW comes to the scene with four unique attributes, which are the result of the combination of its subcriticality and use of liquid fuel. These are process robustness, poison insensitivity, neutron efficiency and completeness. The robustness of the ATW system processes allows it to accept a wide range of nuclear waste with significant variability in isotopic assay and chemical contamination, from defense spent fuel and scrap plutonium to commercial spent fuel. Since there is no need to reach and maintain criticality, the ATW systems are relatively insensitive to poisons and burnup fuel variation or poor fuel characterization. Because of the rapid elimination of fission product poison and the achievement of high neutron fluxes, ATW systems should be able to burn plutonium and most other components of nuclear waste efficiently , with low standing inventories, and to a high degree of completeness, so that the waste load destined for permanent storage in a repository might be reduced, and such repository (containing no significant TRU waste or mobile fission products) would become technically feasible and politically acceptable.

ATW can accept the spent fuel generated by any of the existing and conceivable future types of nuclear reactors. ATW's front-end processes produce a feed of unseparated actinides directly suitable for nearly complete subcritical burning in the actinide burn unit. The unit operates subcritically, driven by a large current LINAC. Power production to offset operational costs is optional.

In ATW, the excess neutrons generated by the accelerator and the fission of the higher actinides are used in a blanket/reflector containing long-lived fission products to be transmuted. The separated uranium is collected and sent to permanent storage. Plutonium and higher actinides are completely eliminated, as well as the most troublesome fission products

The successful implementation of ATW systems will lead to the elimination of plutonium, higher actinides and selected fission products from the nuclear waste stream. With the implementation of ATW systems the efficient (and eventually full) utilization of the existing uranium and thorium energy resources will become possible, with strong proliferation barriers.

D.2.2.2. GENERAL FEATURES OF ATW

Fuel preparation. Spent fuel coming from operating nuclear reactors (LWRs, HTGRs, CANDU, and others) is processed using the "Hydrofluorination+Electrowinning in Molten Fluorides" process conceived at LANL. The process allows the separation of the enriched uranium and/or thorium contained in the spent fuel without plutonium or actinide separation. The role of the fuel preparation part is to prepare the waste feed for use in the target/blanket cell. The process is designed to be relatively insensitive to the composition of the waste feed. A unique feature of ATW is that the system can accept a wide range of feeds with minimal chemical preparation. The use of stable high-temperature molten salts as the primary chemical medium is key to this opportunity.

Actinide and selected fission-product burn. Subsequent to fuel preparation, destruction of plutonium, higher actinides and selected fission products is accomplished in the passively safe, deeply subcritical blanket conceived at LANL. Neutrons to sustain the burn phase are produced in a liquid metal spallation target driven by a large-current proton accelerator operating in the 1- GeV energy range. When a proton beam is injected into the target, over 30 neutrons per incident particle per GeV are generated via spallation reactions. These neutrons are multiplied by fissions in the surrounding subcritical blanket containing the actinides. Electricity may be generated from the heat released by the fission processes, and a fraction of the electric power generated will be available to the commercial grid for distribution. The high-flux, low-inventory burn phase is assisted by the cleanup processes for fission product removal conceived at LANL for use in the ATW burner (sparging, electrowinning and reductive extraction). The burn is nearly complete and only certain fission products are discharged by the process.

At the end of ATW operations, practical processes will be used for removal and recycle of radioactive materials from the salt. The long-lived hazardous fission products, e.g. ^{129}I and ^{99}Tc, will be transmuted into stable isotopes by neutron irradiation. Because there are no significant amounts of actinides and technetium or iodine (the long-term hazards) in the waste stream, the waste need not be guarded for fear of diversion of fissionable material, and its long-term toxicity is greatly reduced. A general layout of the ATW components is illustrated in Fig. 1.

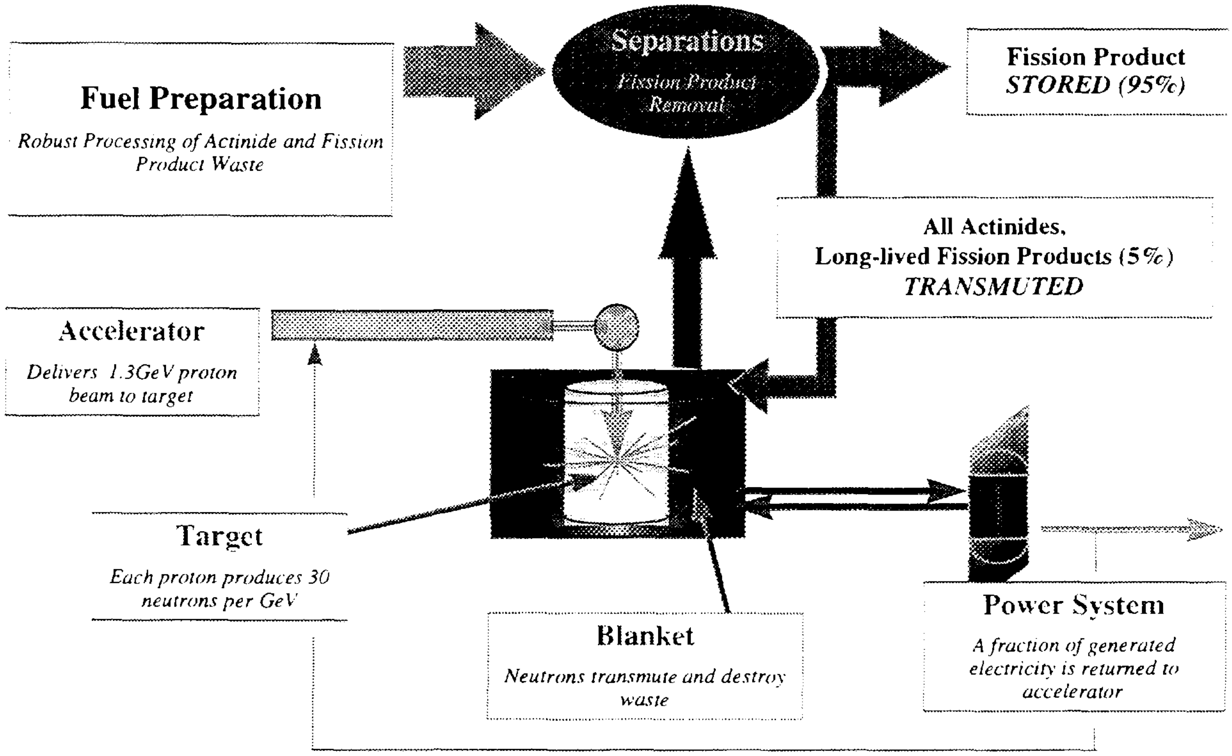


FIG. 1. A general layout of the ATW components

D.2.2.3. THE VALUE OF SUBCRITICALITY

The ATW system is operated in a subcritical mode, thereby precluding a self-sustained chain reaction regardless of whether the accelerator being on or off. From a dynamic standpoint, driving a system with an accelerator decouples the source of neutrons from the fissile fuel: subcritical systems can operate relatively unconstrained by the internal workings of the blanket. Critical systems, on the other hand, are driven by the internally generated delayed neutron "source." Their control mechanism depends on the absorbing properties of the control rods and their influence on the reactivity of the system, and is deeply affected by the internal conditions of the core, e.g. neutron flux and spectrum, and fission-product buildup. Critical reactors are therefore constrained by the flux, spectrum and internal composition tolerated in the core because these parameters determine the effectiveness (reactivity worth) of the control mechanism.

From a static point of view, the accelerator provides a convenient control mechanism for subcritical systems that is much faster than that provided by scram rods in critical reactors. Subcriticality itself adds an extra level of operational safety concerning possible criticality accidents.

Although additional safe-shutdown mechanisms might be required, accelerator-driven subcritical systems can work without control rods, and in principle are insensitive to flux, spectrum, fission-product buildup and core composition changes to a much higher degree than the equivalent critical reactor. The response to reactivity transients is also much less severe and higher neutron fluxes are possible. This inherent robustness allows subcritical systems to accept fuel that would not be acceptable in critical systems, and burn it with minimal processing requirements [2].

A considerably larger excess of neutrons per fission can be produced in deeply subcritical systems than it is possible to generate in critical (reactor) systems. Although these neutrons are not inexpensive, the substantial progress recently achieved in accelerator design and operation efficiency, allows the effective use of such neutrons for the transmutation of long lived fission products and other tasks.

Neutron economy of subcritical systems.

The basic equation relating source- and fission- generated neutrons and governing the behavior of subcritical systems is the following:

$$k_{eff} = \frac{\eta}{1+\alpha+P+L}$$

where k_{eff} is the reactivity of the system, related to its neutron multiplication M factor by the equation:

$$M = \frac{1}{1-k_{eff}}$$

Critical (self-driven) systems have keff =1 and infinite multiplication

η= average number of neutrons released by each fission

α= ratio of neutron-absorption-to-fission cross section in the active component of the fuel (plutonium and actinides in the case of the ATW burner)

P= number of neutrons parasitically absorbed in the system per fission

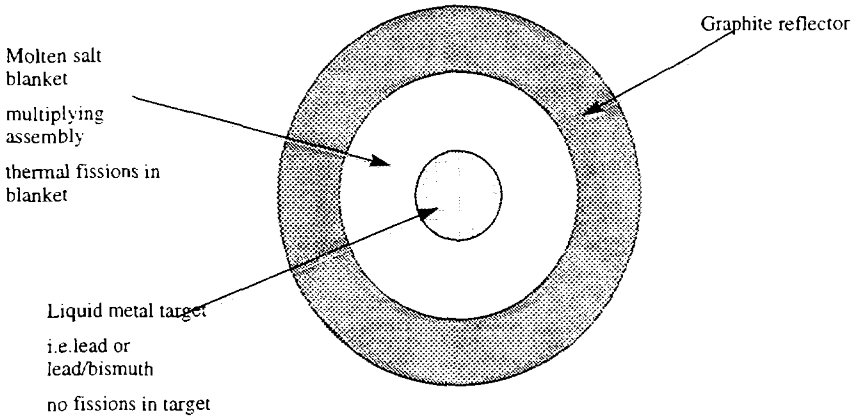
L= number of neutron that leave the system (leakage) per fission. These are the "excess neutrons" available for work (waste transmutation and fuel enrichment)

If the ATW burner is operated with a fast neutron spectrum then the values for the parameters are: α=0.6; P=0.2 ; η~3. If the ATW burner is operated with a more thermal spectrum (molten salt reflected by graphite), then the values for the parameters are: α=1.4 ; P=0.4 ; η~3. Values of n significantly larger than 3 may be possible in "hyperfast", deeply subcritical systems directly driven by spallation neutrons generated in the fuel, where a substantial number of fissions are initiated by high energy spallation neutrons instead of the relatively slow fission neutrons. These systems (which could be obtained in liquid lead/bismuth fueled ATW systems) have a spectrum substantially harder than that of fast reactors and offer special advantages in neutron-intensive applications. A schematic view of thermal and fast ATW burner is illustrated in Fig. 2. The number of available neutrons per fission (the L parameter) can be explicitly written as:

$$L = \frac{\eta}{k_{eff}} - (1+\alpha+P)$$

A cursory examination of this relation reveals the existence of two contribution to the number of neutrons available from a given system: the first term is determined by the subcriticality level, the second term by the neutron spectrum. The use of the accelerator drive (systems with lower k_{eff}) will improve the contribution from the first term. The use of fast neutrons will improve the contribution to the total number of available neutrons per fission from the second term. It should be noted that "hyperfast" systems (directly driven by spallation neutrons) could provide also additional neutrons by increasing the value of the parameter n. Applying the values for the parameters n, A, and P to the expression for L, it is apparent that more neutrons are available for use in the fast-spectrum system than in the more thermal system for the same value of k_{eff}.

ATW (molten salt, thermal spectrum)



ATW (liquid lead/bismuth, "fast" spectrum)

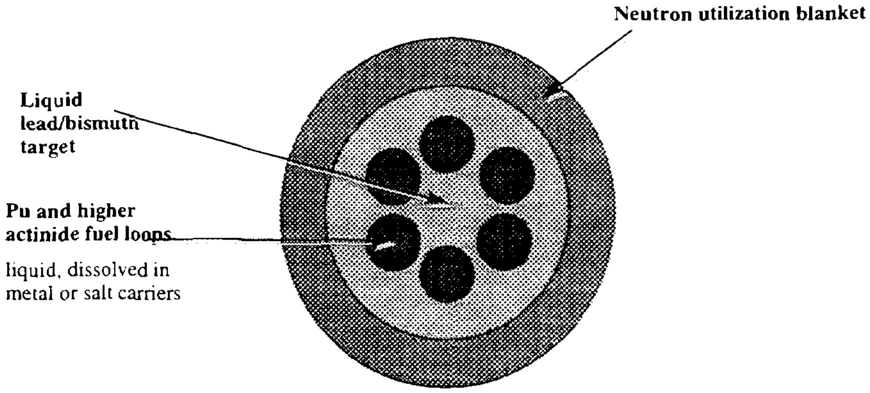


FIG. 2. A schematic view of thermal and fast ATW burners

Just how many neutrons per fission are necessary will of course depend on the extent of the work that is required from the ATW system. An important task for these systems is the transmutation of fission products. Some of the fission products targeted for transmutation remain in the fuel of the ATW burner and their transmutation is accounted for in the fuel capture parameter P. Some will have to be transmuted in special loops, by leakage neutrons. The number of neutrons per fission needed to destroy the long-lived fission products is directly related to the power produced in the ATW burner. The relation depends on the extent of the fission-induced power production in previous operation (fuel burnup), and the type of fuel cycle (whether uranium- or thorium-based). For uranium-based spent fuel, with a burnup of 33,000 MWd/t (typical of a majority of present-day spent fuel), 0.25 extra neutrons per fission in the ATW burner are needed to destroy the long-lived fission products generated during previous and present fission processes. This is the number of extra neutrons per fission that is to be made available through leakage in ATW systems where the transmutation of the long-lived fission products is sought in addition to the destruction of the plutonium and higher actinides). In a near-thermal spectrum configuration (molten salt reflected by graphite), a subcritical system (driven by the accelerator), operating at a keff =0.95 will free the additional neutrons to allow 0.25 leakage neutrons per fission to be used for fission product transmutation. It is true however that most of the usable leakage neutrons will have to come from the accelerator-driven source. A fast-spectrum system will have more than enough neutrons available even in a critical configuration (without accelerator) to perform the required transmutations of fission products. Unfortunately a critical system cannot be constructed to operate on pure plutonium and higher actinides, especially in the fast spectrum. Therefore some degree of subcriticality (accelerator drive) must be used also in fast-spectrum systems, its extent to be determined by safety considerations more than by neutron economy factors. More demanding applications, such as spent fuel reconstitution or the transmutation of a larger number of fission products, will require a larger number of neutrons available per fission. For these applications, fast or "hyperfast" subcritical systems, capable of supplying over 2 available neutrons per fission, are clearly preferable.

In the following discussion we will concentrate on the ATW systems with application to actinide and long-lived fission product burning from spent fuel, and in particular to subcritical systems operating in the thermal neutron spectrum. As we have hinted and will show in more detail, the neutrons introduced into these subcritical systems by the accelerator, the absence of neutron absorbing reactivity control devices, and the possibility of achieving high neutron flux levels can improve the neutron economy to the point that fuels that were considered very difficult to use in the thermal spectrum due to their poor neutron economy, i.e. minor actinides, can be burned subcritically by thermal or near-thermal neutrons [3], with the ensuing advantage of lower inventory.

Liquid-fuel subcritical systems

While it is strictly correct that any type of reactor can be arranged in a subcritical configuration, the attractive features that subcriticality makes possible in ATW (fuel feed flexibility, completeness of burn, low sensitivity to changes in core parameters, relatively high flux, large number of excess neutrons) are best accomplished with liquid-fuel systems, where the burnup is uniform, continuous fueling is available, regular fuel cleanup is possible, and no excess reactivity is required. Furthermore, because of the strongly spatially non-uniform power distribution typical of deep-subcritical systems, fluid fuels are most likely to be used in high-power configurations [4].

Among liquid fuel carriers, both liquid lead and molten fluoride salts, similar to the fuel used in the Oak Ridge National Laboratory (ORNL) experiment [5], offer a good interface for the versatile, proliferation-resistant front-end process identified for the waste plutonium (molten salt hydro-fluorination and electrowinning). Both molten salt and liquid lead, because of their chemical and electrochemical stability, also can provide a natural medium for simplified back-end fuel cleanup processes. Both carriers are low vapor pressure liquids even at high temperatures, and therefore add to the safety and performance of the systems, allowing low-pressure operation of the blanket and high-efficiency power production. The complete absence of water in the primary system also eliminates the possibility of hydrogen generation and the attendant risk of accidental explosions. Designs based on both options are currently under investigation at Los Alamos. In the following chapters, the molten salt fuel carrier option will be explored. The lead-fueled system, which form the basis of the "hyperfast" system currently under investigation at LANL, and other variants where both fast and thermal spectra are used to minimize the required accelerator size, will not be described in this paper.

D.2.2.4. BASIC DESIGN CHOICES FOR THE ATW MOLTEN SALT CONCEPT

Molten salt reactors have been operated and elements of the molten salt fuel cycle were developed and deployed at ORNL during the 1970's. The addition of the accelerator drive and the absence of breeding requirements for the fuel cycle make the ATW system considerably simpler and more forgiving than the critical reactor proposed and carried to a considerable level of design detail in the ORNL work. After a long search through a broad parameter space of options [6, 7, 8, 9 and 10], the Los Alamos molten salt ATW concept has converged on a system reflected by graphite, a liquid lead-bismuth neutron production target, and pyrochemical separation processes.

High-current accelerators and liquid-lead target

The high-power accelerator technology required for ATW has been under continuous development for the past three decades at Los Alamos. Presently, Los Alamos is involved in a major project to develop a large accelerator (100 mA, 1300 MeV) for the production of tritium (APT) for defense applications.

Similar accelerator designs were reviewed by the Energy Research Advisory Board of the U.S. Department of Energy [11] and by the JASONS [12]. These reviews gave general endorsement of the proposed accelerator technology with the provision that appropriate pilot and demonstration steps be made along the way towards the construction of a full scale facility [13]. In October 1995 the DOE committed to the demonstration of the accelerator technology for application to tritium production (APT). The operating parameters of the proton linac envisioned for the APT project will be similar to those needed for the ATW system.

The average power needed for the very largest of systems we propose requires an accelerator power of around 200 MW (100 mA beam at 2000 MeV). This can most readily be achieved with a linear accelerator. At present, the highest-power, linear proton accelerator is at LANSCE (formerly called LAMPF), and operates at around 1 MW (1 mA at 800 MeV). At first, proposing an accelerator 100 times bigger might seem like a very large extrapolation, but it is not as demanding as it appears. First, the LANSCE linac is operated with only every fourth bucket filled with proton beam. Filling every bucket (by use of funneling) or every other bucket (if unfunneled) immediately increases the average power by a factor of four (or two). Also, the LANSCE linac is a pulsed machine operated at 6% duty factor: going to 100% duty factor would produce a factor of 17 increase in power. The charge in each microbunch can be increased by about a factor of four and still stay well within the stable space-charge regime. Therefore, an improvement by a factor of more than 200 (4 x 17 x 4) is possible by the extension of proven proton linac technologies. This should enable operation of 100-mA, 2000-MeV linacs based on current technology. Figure 3 illustrate the possibility of using LANSCE LINAC for ATW applications.

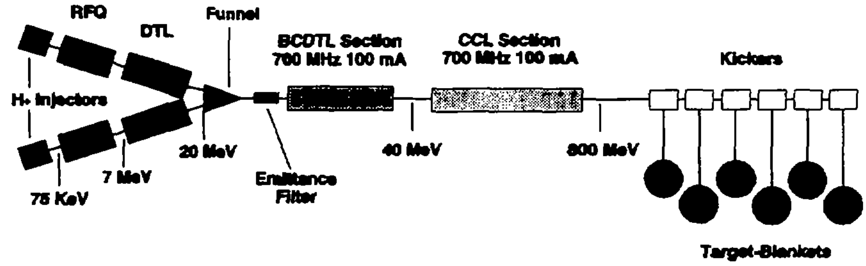


FIG. 3. The ADTT linear accelerator design for multiple targets.

The function of the target in the ATW system is to convert the incident high-energy proton beam to neutrons. Among the requirements for an ATW target are:

1. Compact size to enable good coupling to the surrounding blanket,
2. High power operation, up to 200 MW,
3. High proton-to-neutron conversion efficiency,
4. Reliable and low maintenance operation,
5. Safe operation,
6. Small contribution to the waste stream.

These system requirements are best met by molten lead or lead-bismuth eutectic (LBE). LBE has a melting point of 125°C (200°C lower than for pure lead) and considerable experience exists in the use of LBE in reactor systems. A significant problem with LBE, however, is the production of radioactive and highly mobile polonium from high-energy proton and neutron reactions on bismuth. This becomes a concern in accident scenarios where the polonium contained in the LBE is rapidly released at high temperatures. Irradiation of lead, on the other hand, produces much less polonium, but there is also considerably less experience for pure liquid lead systems in reactor operation.

The use of molten lead or lead-bismuth eutectic in the neutron production target provides the following advantages:

1. High power operation. For compact solid-target systems, the maximum incident beam power is limited by the ability to cool the targets. In liquid metal targets, the medium can be circulated to

external heat exchangers, eliminating the need for coolant in the target production volume.

2. Lead and LBE have good proton-to-neutron conversion efficiency because of their high atomic number, high density and low neutron absorption cross section. The neutron production of large-size lead targets has been measured at LANL to be 22 neutrons per 800-MeV proton, substantially better than any of the other materials tested [15]. The absorption cross sections are about one hundred times lower for thermal neutrons than that of tungsten .
3. Radiation damage and shock issues that exist in solid targets are less of a concern with liquid targets. The fluid container and window are the only parts of the target system which have radiation-induced lifetime limitations. The performance of the liquid part of the target is not affected by irradiation, so that the operational life of the lead or LBE fluid can extend far beyond the lifetime of the container.
4. Lead and LBE are low pressure liquids with boiling points greater than 1600 °C. They are solid at room temperature, reducing the potential for uncontrolled release of radioactive material when the target fluid is removed and cooled.

The beam/target/blanket interface (target structure) is the point where three systems converge and certainly will be the most stressed in the design from the materials point of view. The ATW design for this structure will have to "demonstrate ability to maintain system integrity at the beam/target/ blanket interface" for the anticipated time of operation. In the ATW system, the low pressure of the target and blanket systems and the liquid nature of the target material will be helpful in this regard. The target should be a totally enclosed structure, capable of being rapidly inserted and extracted from the blanket. This structure must be compatible with both the lead and molten-salt liquid interfaces and therefore might suffer from limited resistance to radiation damage due to the properties of the available structural materials. The use of liquid metal targets however, should allow for the softening of the very hard spallation spectrum at the target container walls to levels typical of fast reactor systems. The choice of structural materials can therefore directly extend to the large number of materials tested and satisfactorily used in fast reactors. Proton and neutron damage rates for the window and the rest of the structural parts of the target are not expected to exceed the operational limits established at the Los Alamos Neutron Science Center (LANSCE) [16]. Based on LANSCE experience and extensive experimental studies of materials damage by neutrons (HT-9, Cr-Mo alloys, and Inconel), the lifetimes of the parts of the target most exposed to the neutron flux will be one to two years [16]. These materials have also shown good compatibility and no corrosion problems with LBE as long as the temperature is kept below 400°C [17]. For higher temperature compatibility, the use of carbon based ceramic materials will need to be explored. Figure 4 illustrates a design for the liquid lead target currently under development at Los Alamos.

Experience Base

- ***Solid targets and windows at LAMPF accelerator***
- ***Flowing Pb-Bi technology in Russian reactors***

Development Areas

- ***Adaptation of western and Russian molten metal technologies***
- ***Optimal integration with blanket***

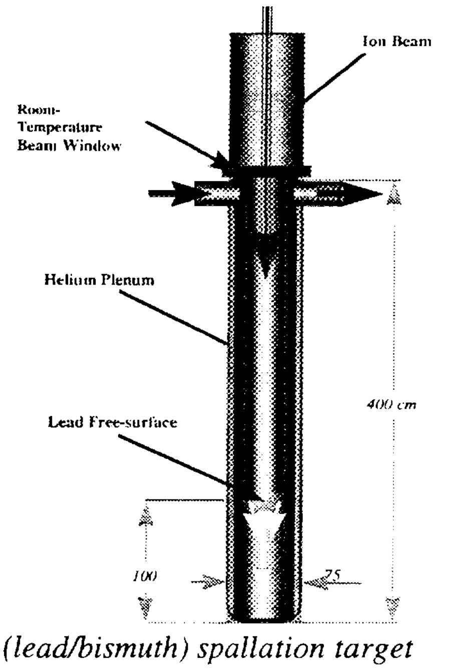


FIG. 4. A design for the liquid lead (lead/bismuth) spallation target

Molten-salt liquid fuel.

In the homogeneous molten salt fuel, as opposed to solid fuels or liquid suspensions (slurries), many fission products are readily dissolved into the liquid medium, and can be extracted readily without processing the fissile component of the fuel. The use of homogeneous molten salt fuel, in fact, allows the adoption of fuel preparation (front-end) and on-line cleanup (back-end) processes that do not require fuel partitioning and refabrication. These processes, which will not be detailed in this paper, except for the description contained in Fig. 5, do not give rise to extraneous waste streams and provide substantial proliferation and diversion barriers.

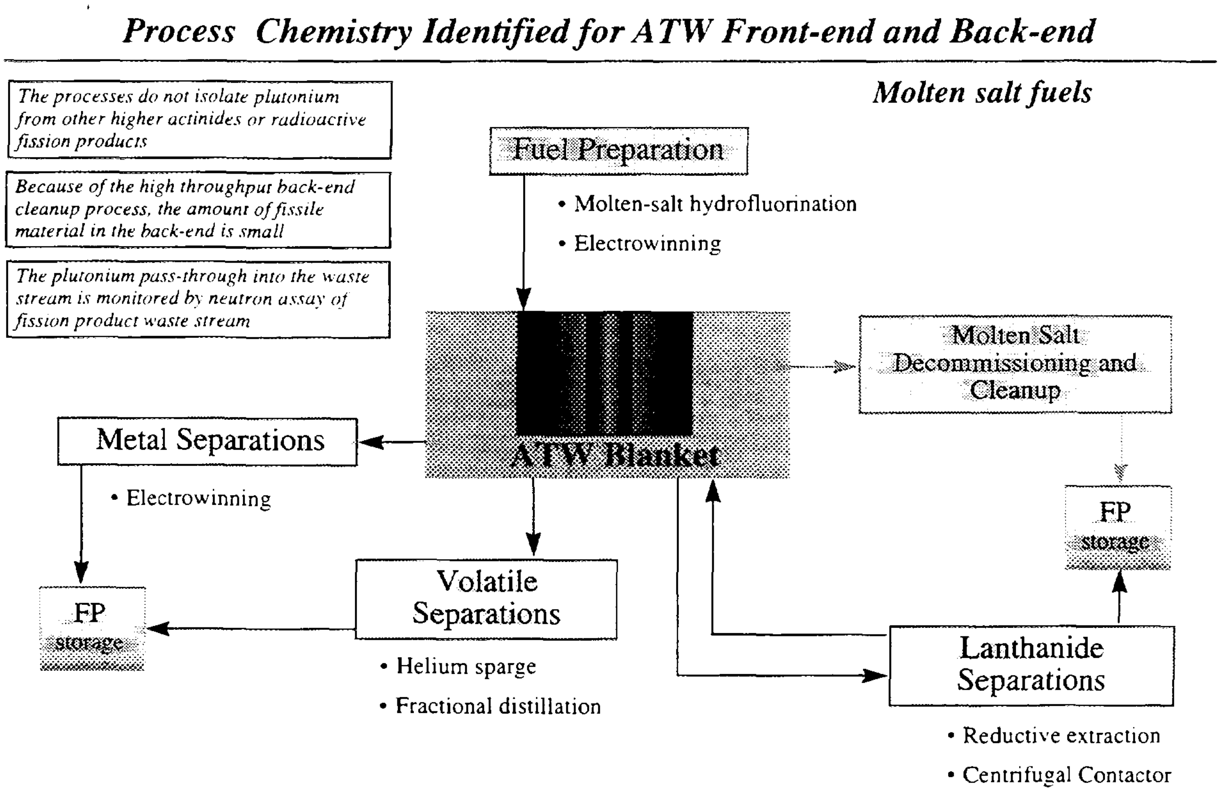


FIG. 5. Process chemistry identified for ATW front-end and back-end

The molten salt envisioned for the ATW system is a mixture of lithium fluoride and beryllium fluoride (^7LiF-BeF_2, 0.67:0.33), with melting point of ~450°C, would be used at between 600°C and 700°C. Its boiling point is 1700°C, and at 700°C the vapor pressure is only 10 milliTorr. Isotopically pure ^7LiF would be used because of its lower neutron absorption. All elements can be dissolved as fluorides in adequate amounts in the carrier salt which is inert in dry air and nitrogen at all temperatures. However, the melting point of ^7LiF-BeF_2 salt is relatively high, and the presence of lithium in it leads to the production of undesired tritium under neutron irradiation. The use of mixtures of other fluoride salts (e.g. NaF) to mitigate these concerns may be practical [18].

Molten salts as reactor fuels and as coolants have been under study and development for over 40 years, and their chemical, physical, and irradiation properties are well suited to reactor operation. The Molten-Salt Reactor Experiment (MSRE) at ORNL, with several years of successful operation, contributed significantly to molten-salt reactor technology. This 7.5 MW_t reactor was operated continuously from 1965 to 1969 with no major problems [19]. At that time, two industry/utility advisory groups concluded that an adequate molten-salt technology base existed and suggested that the Atomic Energy Commission should proceed with a molten-salt demonstration power plant [20].

The progress of the ORNL molten-salt design study is presented in the entire February 1970 issue of Nuclear Applications and Technology. The issue was devoted to a review of molten-salt reactor technology and to a description of a conceptual design for a molten-salt breeder reactor [21]. The program at Oak Ridge ended after the Liquid Metal Fast Breeder Reactor (LMFBR) was selected as a superior breeder. The successful operation of the MSRE and the substantial amount of research and development on molten-salt reactor materials and processes at Oak Ridge indicate that with adequate updating of that technology base, a prototype molten-salt ATW plant could be successfully constructed and operated.

Whereas most of the work done at ORNL involved the use of uranium dissolved in the salt, ATW would use primarily plutonium and higher actinides. This would introduce minor modifications to the solutions adopted in the Molten Salt Reactor program, as the oxidation/reduction potential associated with fissions in plutonium fluoride tends to be different than for uranium fluoride. Low molar concentrations of PuF_3 were added to the MSRE as makeup fuel during the latter portion of its operation. Binary and ternary phase behavior data exists for many of the components of interest [22, 23, 24, 25, 26, 27]. Only sparse data exists for quaternary and higher order systems.

Graphite reflector

Due to the presence of large amounts of light isotopes in the salt (lithium, beryllium, fluorine) the neutron spectrum for molten salt systems is always thermalized to a significant extent. To provide optimal moderation and neutron reflection, graphite can be used in the core because of its high-temperature compatibility [28, 5] with the fluoride molten salt fuel. Concerns existed early in the Molten Salt reactor program that seepage of fuel salt into small cracks in the graphite could lead to local overheating and crack propagation. However, there was no evidence that significant salt entrainment took place in the graphite.

In dealing with large amounts of actinides and essentially no resonance absorbers, a well thermalized spectrum will lead to large positive temperature reactivity coefficients. This is due to the presence of low-temperature fission resonances in the cross section of the higher actinides, especially the plutonium isotopes. A less moderated spectrum eliminates the problem in the ATW subcritical liquid fuel system. The ATW burner will use no internal moderation besides that provided by the lithium and beryllium in the base salt. Graphite will be used as a reflector and to protect the nickel-based hastelloy vessel from the damage of neutron flux.

Graphite is subject to neutron damage. With increasing neutron fluence, graphite first contracts, then expands at a fast rate. This rapid growth rate represents a rapid decrease in density and the possible formation of voids and cracks. In this stage, mechanical properties will quickly deteriorate and the graphite might become permeable to the molten salt. A fast neutron fluence above 3×10^{22} n/cm^2 in graphite usually produces the onset of the rapid growth phase, with significant dimensional changes and marked degradation of its mechanical properties [29]. The neutron damage to the graphite is caused by the fast component of the neutron flux, which is usually defined as the flux above 50 keV. For a thermal spectrum system, neutrons are quickly moderated so that the neutron damage rate is proportional to the fission rate per cm^3. Therefore the neutron damage rate is proportional to the power density rather than the thermal neutron flux. Because of the significant complications involved with the removal of graphite from molten salt cores, it is highly desirable that the graphite reflector maintain its integrity throughout the plant life.

Because the fuel is kept relatively clean of fission product absorbers, even at the moderate power densities present in the ATW burner, the neutron flux is expected to be in the range 2-5×10^{15} n/s-cm^2 and very strongly peaked This flux could give reasons for concern about the effects of high cross-section fission product poisons, such as xenon, on the stability of the blanket operation. Xenon fixed in position such as by the constraints of solid fuel can cause the onset of spatial power instabilities. In the ATW system however, the fuel is continuously mixed, and therefore local concentrations of xenon in the fuel are not expected. Any unstable behavior would be limited to the amount of xenon that might become constrained by the graphite and would not extend to the bulk of the fuel. Since most of the xenon produced in ATW systems would be removed from the salt through sparging on a very fast time scale (1-2 minutes), and since there is no graphite moderator, but only a reflector, it is highly unlikely that xenon could be a source of unstable behavior. Xenon permeation into the graphite would be more of a waste cleanup problem because of the embedding of radioactive cesium daughter isotopes in the graphite, and should be dealt with accordingly by appropriate surface treatment (i.e. pyrolytic sealing) [28, 18].

Secondary coolant

An intermediate (secondary) coolant loop is required in the ATW system to transfer the heat generated in the molten salt fuel to a power-producing steam cycle. The coolant in this loop must have low vapor pressure at its operating range of 500 to 700 °C. The coolant must be compatible at one end with the fuel salt, and at the other end with the water/steam of the steam generator loop. The coolant must trap tritium produced in the core and keep it from reaching the steam plant. Materials compatibility must be assured and it is also desirable to provide an environment that is essentially free of hydrogenous materials for safety reasons. The coolant selected for the intermediate-loop in the MSBR design was a sodium fluoride/fluoroborate eutectic salt (NaF-$NaBF_4$). This was chosen because of its compatibility with Hastelloy and with the fuel salt, its low cost compared to ^7Li-bearing salts, its adequate thermophysical properties (melting point, viscosity, specific heat and conductivity), its ability to trap tritium and its value in shielding the heat exchangers from delayed neutrons coming from the irradiated salt. No nuclear experience exists with this fluoroborate, but extensive non-nuclear tests were performed on this salt under the MSBR program, and several concerns were uncovered. At high temperature, the fluoroborate decomposes and forms chemically hazardous BF_3. Fluoroborate also decomposes upon contact with fuel salt, and has a limited compatibility with steam, in that its corrosivity is greatly increased by the presence of moisture.

These issues do not rule out the use of the salt. One of the most important reasons for specifying fluoroborate as the secondary coolant salt is its ability to trap tritium. Significant quantities of tritium, in fact, are formed as a result of neutron capture (n, a) in ^6Li, which is always present in the salt because of (n,2n) reactions in ^7LiF, and tritium is also produced by (n,nα) reactions in ^7LiF. This tritium might migrate through the coolant system into the steam system, and from there to the environment. Simulation experiments using the fluoroborate salt, however, showed that the tritium could be effectively trapped by oxygen-bearing species ($Na_2B_2F_6O$) contained in the fluoroborate loop, as sodium hydroxy-fluoroborate ($NaBF_3OT$) [30]. Alternative coolants are also possible, such as lead and lead alloys or low melting point salts.

Structural materials and components

The structural material of choice for the molten salt blanket is Hastelloy-N, that was used at Oak Ridge during the MSRE work [31, 28]. Two problems, however, were discovered during the operation of the MSRE that required further alloy development:

1. Hastelloy-N suffered radiation embrittlement due to the accumulation of helium and, to a lesser degree, hydrogen at the grain boundaries [5]. The helium and hydrogen resulted from thermal (n, a) and (n,p) reactions on the unstable ^{59}Ni produced by thermal neutron absorption in the Hastelloy nickel. Helium was also produced by (n, a) reactions on boron impurities. To achieve a 30-year lifetime for the reactor vessel (made of Hastelloy N), the MSBR design had the inside surface protected by a 76-cm thick graphite reflector to reduce the thermal neutron fluence.
2. Post-operation testing of Hastelloy-N samples from the MSRE system revealed the presence of small cracks on the side exposed to fuel salt [5]. These cracks were later determined to be due to the presence of the fission product tellurium at the grain boundaries [18].

Experimental work following the shutdown of the MSRE laid the basis for eliminating these problems. It was found that modified Hastelloy-N, with fine carbide precipitants (Ti at first, then Nb, Zr and Hf) within the grains, could restrain the helium migration to the boundaries and mitigate the radiation embrittlement problem [32, 29]. Two solutions to the tellurium cracking problem also were investigated: adjustments to the salt oxidation potential to keep the tellurium from attacking the Hastelloy, and slight modification of the Hastelloy-N composition (addition of niobium). The combination of both essentially eliminated the tellurium problem [29]. The above-cited solutions to the problems observed in the MSRE have all been demonstrated in the laboratory environment. In the ATW system, tellurium would also be continually extracted from the salt.

The availability of a suitable structural material in the Molten Salt Reactor program allowed a considerable amount of engineering work to be conducted at Oak Ridge on the design and construction of components suitable for operation in fluoride molten salt fuel. Corrosion-resistant pumps, valves and heat exchangers capable of reliable operation at high temperature [19] were provided for the MSRE.

Pumps. The pumps for molten salt blankets must reliably circulate fluoride salts in the primary and secondary salt systems at temperatures as high as 750°C. The low electrical conductivity of molten fluoride salts hinders the use of electromagnetic pumps. However, centrifugal pumps, albeit of low power, were demonstrated to work very reliably in the MSRE [33]. For full-scale ATW systems, there is yet no experience in constructing and operating the much larger fuel pumps that will be required.

Valves. Valves are required in the ATW primary and secondary systems for isolating portions of loops, directing flow to alternate paths, and providing flow variations during startup, shutdown, and off-normal conditions. The only type of molten salt valve that was operated in the MSRE was the freeze valve. This is a type of valve suitable for on/off operations, such as fuel-salt drain. The fuel-salt drain valve used in the MSRE consisted of a flattened section of a two-inch pipe equipped with external heaters and coolers. It operated reliably. Although a large body of experience exists on several types of valves for operation in liquid metals at comparable temperatures, considerable more work is required for the satisfactory application of these designs to the molten salt environment

Heat Exchangers. The operation of the MSRE heat exchangers was reliable, although the capacity of the heat exchangers was somewhat lower than expected. There was no evidence of scaling or corrosion. The experience base with the proposed $NaF-NaBF_4$ secondary coolant derives from numerous tests for the MSBR design, but not from actual heat-exchanger design and construction. The experience base appears however to be sufficient to justify that the heat exchanger design be based on correlations for normal fluids, once the physical properties are accurately measured.

Overall, the MSRE fuel salt, $LiF-BeF_2$, its containment system and the associated hardware performed well throughout the life of the project [21, 5]. The ATW project will build on that experience, using the improved materials and techniques available in the 1990's.

D.2.2.5. ELEMENTS OF MOLTEN-SALT NUCLEAR SYSTEM DESIGN

In 1960's, the Molten Salt Reactor Program at ORNL worked on thorium-based molten salt breeding systems. The activity led to the design of the Molten Salt Breeder Reactor (MSBR) and was based on the design and operating experience of the Molten Salt Reactor Experiment (MSRE) and its technology support programs. In the early 1970's, ORNL proposed, with strong backing by industry and utilities, a major technology development program that would have culminated in the construction and operation of a demonstration reactor called the Molten Salt Breeder Experiment [5]. Later, in 1979, the Denatured Molten Salt Reactor (DMSR) was proposed for power generation with enhanced proliferation resistance [18].

Based on the evaluation of those activities, the National Laboratories of Los Alamos and Oak Ridge, within the ADTT program, have recently started to revisit the technology base for the design of molten-salt systems, in a framework stressing safety and reliability and with the different goal of waste and/or plutonium destruction replacing the original one of breeding [34, 35]. As a result of this effort, attractive features of molten salt systems were identified as well as special concerns which need to be addressed in the design of the molten-salt ATW concept.

Low pressure. The very low vapor pressure of molten-salts even at high temperatures and the low pressure-drop design of the primary molten-salt circulation system allow for an overall low operating pressure in the primary system. This enhances passive and engineered safety, for example, through natural-convection flow capability, design simplifications, and increased system component and structure reliability. To fully take advantage of this feature, the pressure should also be kept low in the secondary coolant and care should be taken in the steam generator so that the high pressure of that subsystem remains isolated from the low pressure systems in the primary and secondary loops. This will require that pressure isolation devices be employed in the loops and that in the steam generator the pressure be kept as low as practical.

Pool design. The "pool" concept, used in the most recent sodium cooled reactor designs, has significant advantages in terms of a large heat sink and containment of coolant leaks, over the more traditional "boiler" type configuration, used in the large Pressurized Water Reactors (PWR). In the classical pool concept the core is located inside a vessel, together with pumps and heat exchangers. Hot coolant outside the core is separated from the cold coolant by an interface that directs the hot coolant to flow through the pump and heat exchangers back into the core inlet region without using piping. Such a configuration is not directly applicable to a liquid fuel reactor because the absence of clear flow paths would significantly increase the system fuel inventory. By the strategic use of graphite in the blanket and clearly identified salt flow paths, the ATW design can utilize a modified form of the pool concept, which preserves the advantages of the pool concept, especially in terms of heat sinks, natural convection and containment of fuel leaks, and does not lead to large fuel inventories.

Natural convection. The ATW system will be designed so that natural convection is enhanced by using up-flow circulation in the core and arranging the thermal centers for core, heat exchanger and steam generator at increasing elevations. A low pressure-drop design with enhanced natural convection flow will allow for easy passive decay-heat removal in case of a loss of flow, without recourse to fuel drainage.

Large heat sinks and passive safety. Under the worst-case accident scenario (complete loss of primary heat removal), no operator action would be required to remove the decay heat and prevent the release of radionuclides. Temperature increases will be slow because of the large heat capacity available in the primary system cell. To safely dissipate the decay-heat, large heat sinks could be provided surrounding the blanket, or built into the salt drainage system. In-situ heat sinks could passively remove the decay-heat generated by moderately-sized molten-salt blankets (up to 1000 MWt). Larger-sized units might have to be gravity-drained through a freeze valve from the primary system into a drain tank where the heat would be passively transferred to a molten salt coolant and from there to water/steam.

The high flux of ATW systems is not reached through high power density, but through low parasitic capture in the fuel because of fission product extraction. As a consequence, the power density in the blanket is likely to be a factor of 10 lower than in LWR systems. We have evaluated systems of up to 1000 MWt which are passively safe in the event of loss of coolant, even without drainage of the fuel. The main reasons for this special ability to handle shutdown heating are: 1) the liquid nature of the fuel allows natural convection in properly designed systems, 2) the heat source is not localized, 3) the graphite reflector has large heat capacity and heat conductivity; 4) the integrity of the blanket can be maintained for peak core temperatures greater than 1500 °C because of the excellent high temperature properties of graphite and Hastelloy and the high boiling point of molten salt, and 5) with continual removal of the volatile fission products, the decay heat source term is substantially lower than for solid-fuel systems in any time scale longer than a few minutes, while the core heatup characteristic time is of the order of hours. In the event of total loss-of-coolant events at higher power levels, i.e. 3000-MWt systems, fail-safe drainage of the fuel with attendant passive decay-heat removal into large heat sinks is possible.

Containment of radionuclides. Radionuclide containment should be provided through the primary system loop, the loop containing the pressure isolation devices and secondary coolant, and the loop containing the steam generator system. All three loops will be designed as closed-cell constructions located within an additional containment structure. Attention will be given to the penetrations from one cell into another to ensure that proper isolation can be maintained.

Reduction of in-core graphite use. The presence of the graphite moderator in the MSBR design provided the well-moderated neutron spectrum which was essential for the effective breeding of thorium. However, the presence of graphite elements introduces problems concerning (1) the neutron damage in graphite and the need for its periodic replacement in high power-density systems, (2) the need to dispose of highly radioactive graphite, and (3) the possibility of flow obstruction by graphite breakage following extensive radiation damage. Because breeding is not desired in the ATW, all these problems can be reduced by limiting the use of graphite moderator elements in the high power density regions of the core.

Design simplicity. Because of its liquid form, the molten salt fuel permits a greater level of design simplicity than solid fuel reactors. For example, the following items will not be required: refueling machines, fuel rods and assemblies, reactivity control systems and their backup, complex shutdown systems and extensive accident-recovery procedures. On the other hand, salt cleaning equipment, fission product handling apparatus and remote maintenance devices would be required.

Minimization of material problems. The achievement of low inventories during waste destruction is related to high neutron fluxes. Neutron flux and power density are not necessarily closely correlated. The proper selection of operating conditions and sufficient shielding inside the reactor vessel will be required to achieve both low actinides inventories and low radiation damage to blanket components, and to operate at as high a power density as possible to reduce costs. Some compromises will have to be found.

Spatial separation of primary system components. Because all primary system components contain fuel salt, any neutron leakage into components will have to be minimized because it would produce fissions in those regions, with consequent heat generation and fast neutron flux damage and activation.

Conservative operating conditions. Fuel and salt temperatures, flow rates, power densities, operating pressures in the primary system, intermediate system and steam generator system should be selected based on previous experience with the MSRE, the conceptual design work performed for MSBR, the laboratory experience with molten salt tests, and the experience of other operating reactors, most notably the liquid metal reactors. Above all, the selected operating conditions should emphasize safety and reliability.

Based on the previous points, Figs 6 and 7 illustrate the current thinking in the ATW conceptual nuclear design. The performance of such conceptual design, when coupled with the process chemistry which is being developed at Los Alamos for use in the ATW fuel cycle, should provide very small standing inventories and a limited waste stream. Fig. 8 based on preliminary neutronics and depletion calculations, illustrate the low inventory feature of the ATW actinide burner.

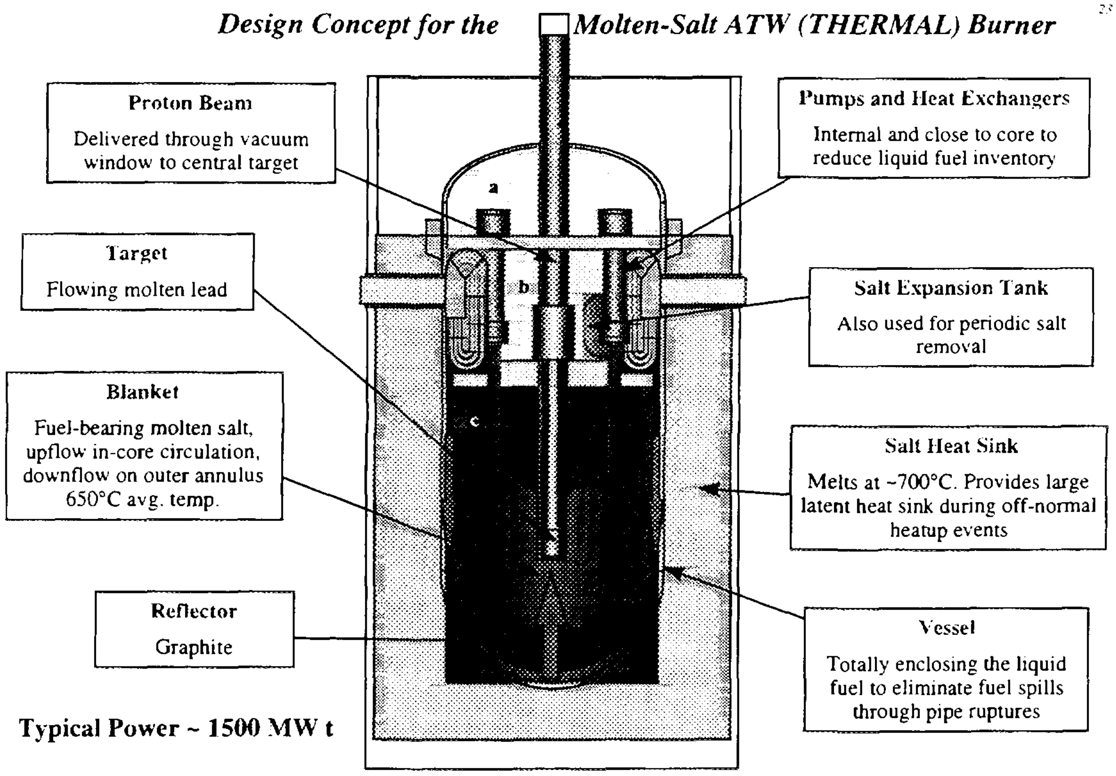


FIG. 6. A design concept of thermal ATW burner

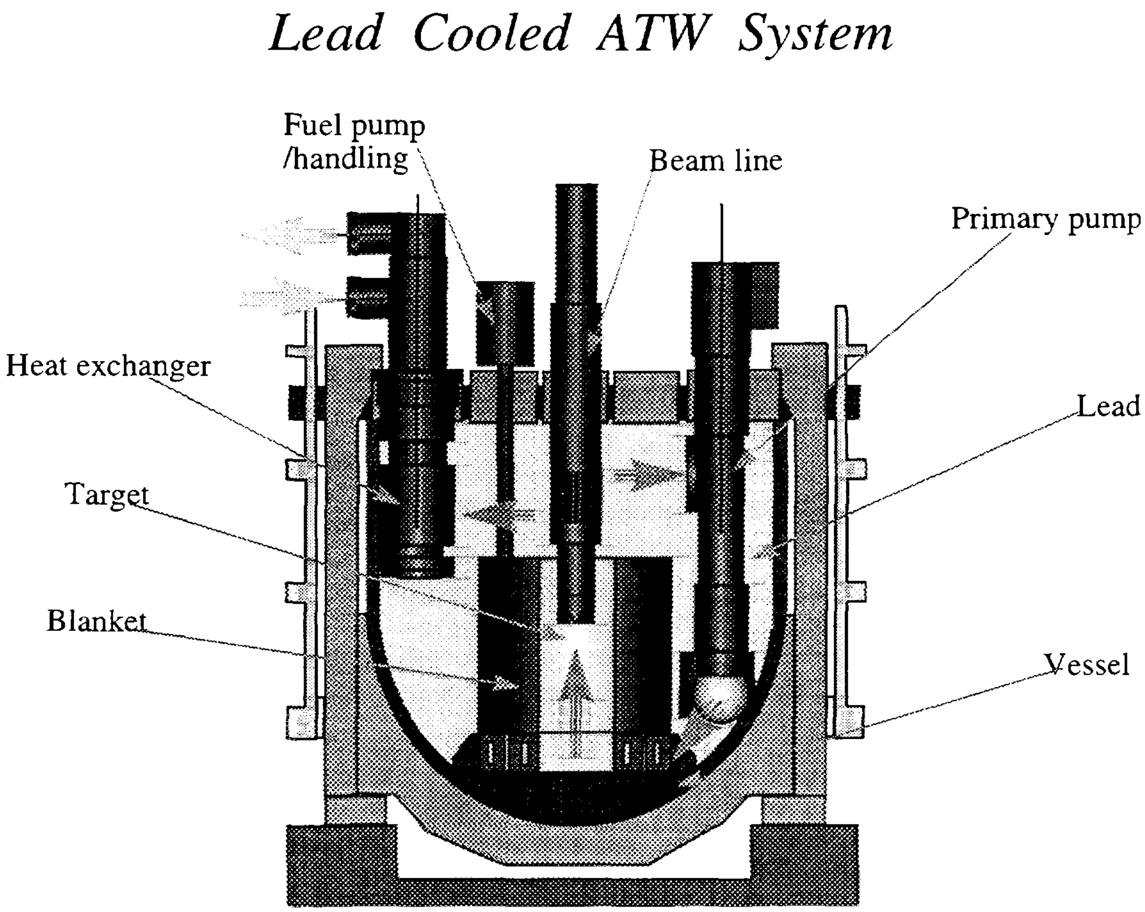


FIG. 7. A design concept view of fast, lead-cooled ATW burner

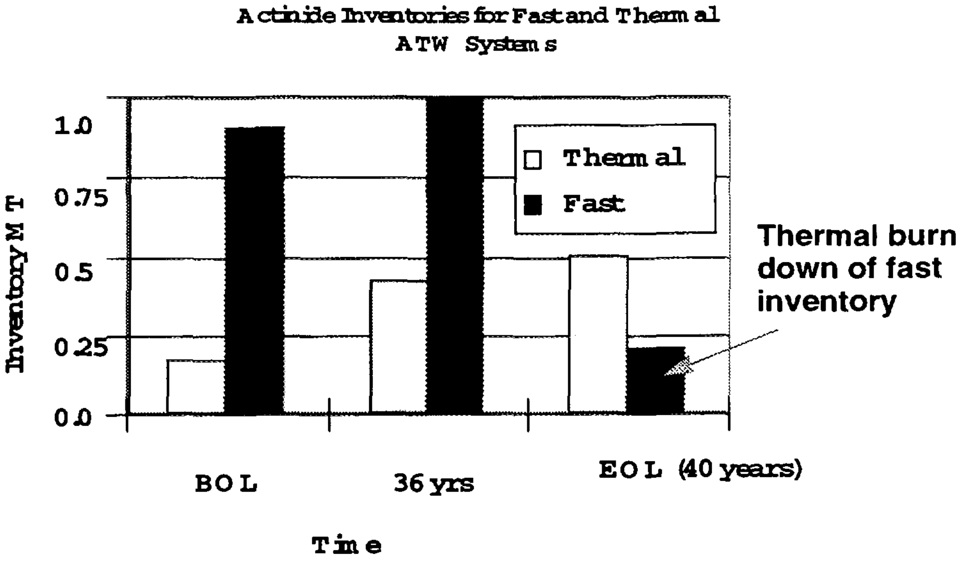


FIG. 8. Life-time actinide inventory for fast and thermal ATW systems

D.2.2.6. THE SAFETY PERSPECTIVE

Accelerator-driven subcritical systems involve sustained fission processes that will produce radiation and fission product levels comparable to conventional power reactors. As such, a nuclear safety approach similar to that used for reactors must be employed.

The fundamental nuclear safety objectives applied to the design of fission reactors are [36]: 1) control of fission power, 2) adequate cooling, 3) containment of radioactive materials, 4) prevention of inadvertent criticality and 5) control of personnel and public exposure. Also, for an ATW system to be licensed in the United States, it would need to have characteristics consistent with those delineated by the Nuclear Regulatory Commission (NRC) in its policy statement on regulation of advanced nuclear power plants [37]. These characteristics include: simplicity, slow response, passive system reliability, reduced system interdependencies, reduced severe-accident concerns, assured defense-in-depth barriers, and clarity in safety analyses: these features provide the safety robustness and margins that are desired for advanced systems, which initially will have had little operational history.

Control of fission power

The response of a neutronically critical system to reactivity insertions is highly nonlinear and rapid: the initial response is in fact related to the prompt-neutron generation time and the longer-term response is related to the addition of delayed neutrons from particular decaying fission products. Indeed, it is these delayed neutrons that make reactor control practically feasible. Reactivity control systems must be capable of preventing power levels that exceed specific limits, which are set to ensure fuel stability, cooling sufficiency, and first-barrier integrity.

To obtain a general sense of the responses of critical and subcritical systems to reactivity insertion events, simple point-kinetics calculations have been performed [14] and their results are shown in Fig. 9.

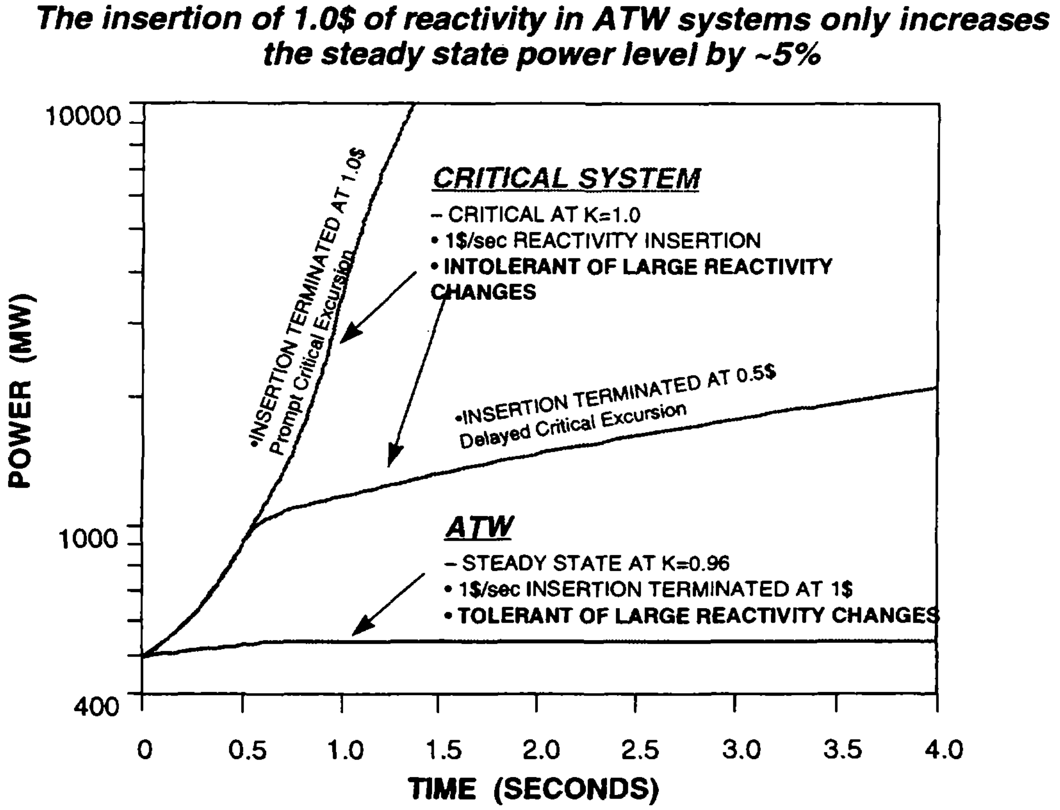


FIG. 9. Kinetics of the critical and subcritical nuclear systems. Responses to reactivity insertion events: 1$/s reactivity insertions.

In these calculations, the reactivity insertion rate was 1 $/s, and its duration was assumed to be either 0.5 sec or 1 sec. For a typical plutonium-fueled molten-salt system, where up to ½ of the delayed neutrons may be released beyond the boundaries of the core in pipes or heat exchangers, a dollar of reactivity is approximately equal to a change in k_{eff} of 0.002. No reactivity feedbacks were assumed in either case to simplify the understanding of the results. The initial multiplication in the subcritical system was assumed to be 0.96, and the initial power for both systems was assumed to be 500 MW. The predicted performance is plotted in figure.

As expected for the critical system, the power rises rapidly and continues to rise after the reactivity insertion is completed. Without negative reactivity insertions from inherent negative feedbacks and/or addition of neutron absorbers, the transients are unterminated. The response of the subcritical system is markedly different in that the power changes very little. The power is inversely proportional to the inverse of the degree of subcriticality of the system (initially 1-0.96 or 0.04). At the end of the 1-sec reactivity insertion transient, one dollar of reactivity was gained, making the final multiplication of the system equal to 0.962 and the degree of subcriticality equal to 0.038. The power change is approximately 5%, and a new steady state is established. A subcritical system appears to be robust in accommodating neutronic upset conditions, because the system's response is predictable and relatively insensitive to the reactivity changes (small power changes for large reactivity changes), as long as the degree of subcriticality is substantial, i.e. large compared to the delayed-neutron worth. Even if the system was assumed to be initially at a multiplication of 0.99 and the same reactivity insertion event occurred, the power would only change by approximately 25%. To put things into perspective, a dollar worth of reactivity insertion would be equivalent to a plutonium feed error of a factor of ten at beginning of life (the most sensitive condition) [38].

Although these large margins and the decoupling of power changes from reactivity changes are attractive in preventing and limiting potential power excursions, and make subcritical systems inherently stable, it is also true that other (desirable) feedbacks, such as those from system temperature changes, only weakly affect the power for subcritical systems. In subcritical systems, power cannot be easily adjusted by means conventionally used in reactors. For example, active control with absorber rods would not be particularly effective in subcritical systems because of the weak coupling between reactivity and power. The obvious power control would be by changing the intensity of the driving neutron source by varying the intensity of the proton beam current.

A second objective for nuclear systems is the highly reliable and rapid termination of the fission process if the control system were to fail or unforeseen reactivity increases were to occur (scram). Normally, reactors have mechanically inserted neutron absorber components (shutdown rods, safety rods, absorber balls, etc.) to provide the scram function. To achieve the required high reliability of inserting these absorbers on demand, redundancy and diversity of components and systems are employed. Because of the delays associated with absorber insertion, the power continues to rise above the scram point: this power over-shoot must be predicted and included in the design of the shutdown system to ensure that safety limits are not exceeded. The design must in turn take into account the reactivity properties of the fuel, moderator and coolant. As a result, fuels containing mainly plutonium and higher actinides, for which there may be little inherent negative reactivity feedback are typically not manageable in critical systems.

If a scram event were to occur in an accelerator-driven subcritical system, the delays would be limited to those associated with detection of the condition, and interruption of the proton beam. The sensing and interruption system could be entirely passive, based for example on fuses for the accelerator beam source, placed in the target/blanket. Once the proton beam is interrupted, the neutron production stops nearly instantaneously and the power in the subcritical system drops with a decay time given by the product of prompt neutron lifetime and multiplication. The shutdown can be accomplished quickly, predictably, and reliably, following detection of abnormal conditions, in milliseconds instead of seconds. The need for in-core neutron absorber insertion with the associated mechanical complications is eliminated. The complexities associated with "managing" the power over-shoot are also eliminated, and inherently unstable fuels like plutonium become acceptable [14].

Analogously, the quick restart of a subcritical system can be performed following a forced shutdown in virtually any situation. A typical problem in reactor operation following shutdown is the buildup of samarium-149, a large cross-section daughter of the fission product promethium-149, which has a 53-hour half-life. Shutting down a reactor (especially a high flux system) can lead to the growth of such large amounts of ^{149}Sm that it may not be restartable until the ^{149}Sm has been removed. No such constraint exist for subcritical accelerator-driven systems: these systems can always restart and burn the accumulated samarium because they do not need to reach criticality in order to start producing neutrons and destroy the poison.

Adequate cooling

Assuring adequate cooling in all situations is essential for all fission reactors. Because the integrity of the first barrier is a key element in the defense-in-depth strategy for preventing the release of fission products, and because this integrity is strongly linked to its temperature, adequate cooling of the first barrier must be assured for normal power operations and for a variety of accident situations, including off-normal shut-downs with the associated decay heat from the fission products. For conventional solid-fuel reactors fuel cladding is the first barrier, and with high power densities and water coolant, the cladding temperature can rise rapidly if cooling is interrupted locally. For the molten salt system, the first barrier is the vessel. The thermal response of this first barrier is therefore linked to the heat capacity of the entire primary system enclosed by the vessel, which is substantially larger than that of the individual fuel pins or fuel assemblies of an LWR. The rate of temperature rise in molten salt systems, therefore, would be much slower than in solid fuel systems. This slow, system-wide response provides substantially increased opportunities to sense inadequate cooling conditions and to respond appropriately.

The molten salt system also exhibits robustness from the standpoint of heat transport in and out of the core region. Again the point of reference is a solid-fuel, rod-type core. In this system, the heat is generated in the solid fuel material and is transferred out of the fuel, across the fuel-clad gap, through the clad, and to the surrounding coolant. The heat transfer processes include conduction, radiation, and convection. The overall heat-transport process is sensitive to dimensional changes, fuel restructuring, fission gas content in the gap, coolant pressure, coolant subcooling, and coolant velocity. Many of these aspects can change substantially during core life, for different operating modes, and for off-normal and accident conditions. Although the process is complex, it must be known reasonably well to assure adequate cooling and protection of the first barrier. In a molten salt system, on the other hand, the heat is generated directly in the molten salt, which is also the heat-transport medium. Heat-transport is therefore greatly simplified and rapid dissipation is possible in the event of inadequate cooling. This arrangement also appears to have a self-limiting characteristic in that if cooling in the core is inadequate, the coolant/fuel salt overheats, eventually boils and is removed from the core [14].

The final aspect of assured cooling is decay heat removal. The challenge is to get the decay heat to an ultimate heat sink before the increase in temperature can damage the integrity of the core. The heat produced in the compact cores of solid fuel systems must be removed into the primary heat transport system. In the molten salt system, instead, the decay heat is generated throughout the

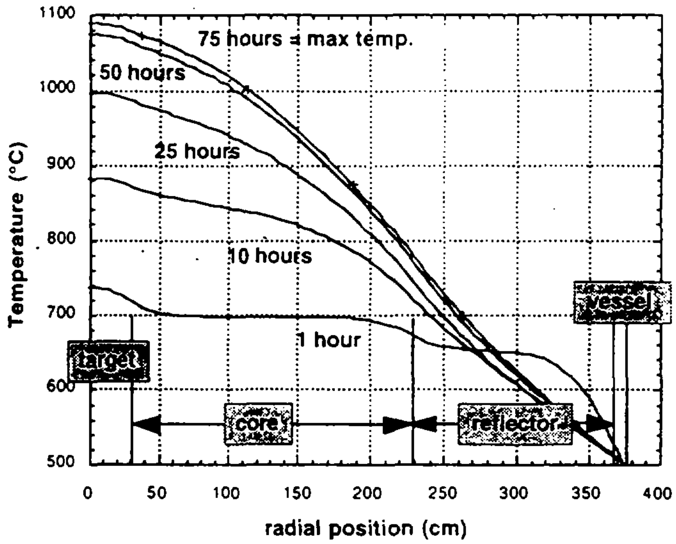


FIG. 10. Temperature rise of a prototype 500 MW_t ADTT blanket in a loss of coolant accident. Assuming only convective cooling and a passive heat removal mechanism, the maximum temperature of 1100 °C is reached in 75 hours.

salt inventory in the primary system. Thus, this system has the possibility for predictable natural convection cooling to the primary system boundary and the possibility of efficient passive heat rejection from the primary system boundary to an external heat sink in emergency situations. The molten salt fuel would then allow the attainment of substantially more powerful "passively safe" units than possible with solid fuels, i.e. able to handle the core heat-up in the event of a cooling system failure unattended and without permanent damage (see Fig.10).

Containment of radioactive materials

The types of fission products retained in the fuel of an ATW-like molten-salt system are different from those in solid fuel systems, and because of the extensive cleanup processes, they are present in much smaller amounts in the liquid fuel. Their release will, nevertheless, constitute a great hazard. Thus, containment of the radioactive materials at all times and for all conditions must be accomplished for the molten salt system.

In conventional reactors, many individual sealed units (fuel pins) are present with limited life requirement (approximately three years before replacement). Thus, if a few of the individual units of the first barrier were to fail for some reason, only a relatively small fraction of the fission product inventory would be released to the next barrier. In contrast, the molten salt system's first barrier (the primary system boundary) must last for the life of the plant, and if it were to fail, a large fraction of the fission product inventory in the salt might well be released to the next barrier.

High reliability in the performance of the primary system boundary must be assured to deal with this challenge, and a highly reliable and effective second barrier. Because the first barrier is not in the core and therefore does not affect the neutronic performance and is generally accessible, design flexibility exists to make the primary system boundary highly robust and to provide inspectability by remote means. Sealed vaults with appropriate heat removal systems, atmosphere control systems, and spill recovery systems could be used as the second barrier. Finally, a surrounding containment or confinement structure would complete the three-barrier strategy to radioactivity release envisioned in the defense-in-depth approach [14].

A special challenge for the accelerator-driven molten-salt system is the integration of the beam transport equipment with the containment barriers. The proton beam can not pass directly through heavy-walled structures without substantial losses and the generation of significant heat and radiation. Some type of redundant thin-window approach is necessary, in order to avoid making the whole accelerator cavity part of the containment enclosure.

Prevention of inadvertent criticality

Prevention of inadvertent criticality in conventional reactors is a relatively straightforward matter because it is a concern only for fuel handling, new fuel storage, and spent fuel storage. Solid fuel segregated into numerous individual assemblies can be handled in a very controlled manner, monitored for structural deterioration, and stored in well characterized and robust structures. In contrast, the molten salt system presents challenges in assuring that the location of all fissile material is known at all times. The potential for precipitation of fissile materials from the molten salt must be considered and taken into account in the design of the various subsystems. The potential for criticality in the fissile material feed system and in the salt cleanup and processing systems will have to be considered. Care will also be required to assure that potential spills and leaks would only accumulate in subcritical configurations. Overall, criticality concerns would be mitigated by the subcritical margin of the accelerator-driven system and by the lower inventories of fissile material.

Radiation exposures to personnel

Limitation of the integrated exposure to the operating staff of a conventional reactor power station involves equipment reliability and provisions in the design for ease of maintenance, inspections, testing and repairs. In this respect, the ATW system presents a challenge to designers and operators because of four reasons: 1) the fission products distributed throughout the primary system will produce high radiation fields in all areas adjacent to the primary system and potentially high exposures during incident recovery operations. 2) delayed neutrons produced throughout the primary system will activate all primary system equipment resulting in radiation exposure potential even after decontamination. 3) the target system will have limited life and require periodic replacement which could lead to additional personnel exposure. 4) the use of a lithium-based salt will result in the generation of substantial amounts of tritium, which could diffuse through metallic boundaries. All of these exposure potentials will have to be recognized in the design process, and special provisions will have to be included to protect workers during inspections, testing, maintenance, and repairs [14]. A considerable premium will be placed on highly reliable equipment and remote maintenance.

Safety features of the Oak Ridge MSBR design

A conceptual design for the ATW does not exist yet, and therefore a valid safety analysis of this concept cannot be performed. For reference, however, it is useful to look at the safety features of the Molten Salt Breeder Reactor (MSBR), for which a substantially developed conceptual design does exist [5], or the Molten Salt Reactor Experiment (MSRE) [19]. While the safety philosophy of the 1960's and 1970's was different from today's, many of the safety concerns and features would be directly applicable to the ATW system [35].

Passive safety features. The MSRE operation was found to be stable and self-regulating. Responsible for this behavior were the strong negative temperature coefficients of the fuel salt and the large heat sink offered by the graphite moderator. The MSRE was simple to control. In over 14,000 hours of critical operation, nuclear parameters never exceeded the operational limits and a reactor scram was never initiated. The criticality of the reactor was controlled by regulating the feed of fissile material. Similar operational characteristics could have been expected for the MSBR. Both reactors, however, were equipped with control rods, even though the MSRE never used them during operation.

Control rods. The control rods for the MSBR design were movable graphite cylinders, whose withdrawal from the core left an under-moderated region causing a reduction in reactivity. Should any of these control rods have broken, they would have floated out of the core, automatically resulting in decreased reactivity.

Decay-heat removal. Major passive decay-heat removal features were missing from the MSBR design. However, this was not a general limitation, but only specific to the MSBR design. It is reasonable in fact to expect that a properly designed molten salt system will exceed the level of passive decay-heat related safety of all conventional and advanced reactor designs.

Loss of primary coolant. Whereas a primary-system pipe break accident in a solid fuel reactor can lead to loss of coolant with resultant fuel overheating, fuel melting, and eventually fission product release; such a failure in the MSBR would lead directly to the spillage of molten salt fuel. This is equivalent to a partial fuel meltdown with breach of the primary system boundary in a solid fuel reactor. In the MSBR design it was argued that in such an accident situation the fuel would immediately freeze and most likely retain the fission products (especially since the volatiles would have been virtually absent from the fuel) so that cleanup of the contamination, repair and restart of the reactor could follow quickly. This procedure however might not meet modern standards of plant safety.

Steam pressure. The MSBR was designed to operate with a steam pressure of 3800 psig, by far the highest steam pressure found in any nuclear steam generator. The steam generator and power conversion plant were based upon the coal-fired Bull Run Power Plant that produced a net efficiency greater than 44%. Obviously the intent was to capitalize on the large thermal efficiency possible at the high molten salt temperature using the supercritical steam conditions. Pressure relief systems had to be provided in the intermediate loop to prevent over-pressurization in the event of a steam generator tube rupture. Such high pressures in the steam generator would probably not be acceptable today in a molten salt system even in the presence of pressure relief systems, because of the very low pressure present in both the primary and secondary systems. It may therefore be desirable to consider lower pressure steam systems, i.e. 1000 psi, analogous to those employed in current low-pressure reactor designs such as the Integral Fast Reactor (IFR). While lowering the steam pressure will tend to lower the energy conversion efficiency, the decreased severity of steam line break accidents will facilitate the design of highly reliable heat exchangers and relax the requirements for the design pressure in the containment structure.

Natural convection capability. The MSBR core and heat exchanger, and also the steam generator, were designed to be all at about the same height, with little difference in the elevation of the respective thermal centers. Such a configuration did not encourage natural convection as a means of heat transport in case of a loss of forced flow. A configuration promoting natural convection would be required in the ATW design.

Shutdown capability. Because the MSBR was to operate with only a very small amount of excess reactivity to be compensated by external control, withdrawal of the control rods placed in the core was sufficient to ensure shutdown under all circumstances. Fuel drainage was not relied upon to shut down the reactor.

D.2.2.7. COST ESTIMATES FOR ATW SYSTEMS

Before a new and potentially expensive concept like ATW can be funded for feasibility studies and a demonstration research and development program, one has to be convinced that it does not have excessive intrinsic economical penalties. The extra costs incurred by an accelerator and a chemical processing plant in the ATW systems immediately raise the cost issue to a prominent level.

Lacking a complete conceptual design, costing of ATW systems is obtained by comparison to similar plants. The following analysis is based on the model developed in [39]. It is assumed that the target/blanket system would not cost more than a critical reactor of equivalent thermal power using similar technology (in terms of operating conditions, the IFR/ALMR reactor is the most similar to the molten salt blanket). The accelerator costs were estimated from the LANL-APT concept and the processing system cost was based on existing facilities, such as the LANL plutonium processing facility and on facilities evaluated for the IFR project.

Further details used in our cost estimates are as follows: The accelerator produces 1-GeV protons. 50 mA to drive the ATW burner. Using a liquid lead target, the yield from 1-GeV protons is 30 n/p. Assuming operating $k_{eff} = 0.96$, thermal to electric conversion of 40%, the engineered efficiency is 88% (12% of the electricity generated goes to the accelerator and the auxiliary operation). This ATW system generate 3000 MWt, or 1200 MWe power. Of this amount, 1060 MWe is available to the grid. It can support four 1000-MWe LWRs.

The cost for the chemical plant ($500M) should be lower than previously estimated for aqueous systems [8] and it is based on a comparison with the IFR processing costs [40]. The ATW fuel cycle includes: spent-fuel hydrofluorination, fuel and waste storage at the front end; reduction extraction of lanthanides, off-gas systems and in-blanket electrorefining at the back-end; replacement operation and storage space; waste treatment and end-of-life decommissioning. The ATW system does not require fuel pin (re)fabrication, and the separation requirements are in general lower than in the IFR/ALMR.

The blanket cost is estimated at $300M, which is a conservative estimate based on IFR core costs, augmented to include the target. The other component costs are adapted from [39], with proportional adjustments in turbine machines, miscellaneous equipment and electricity equipment. The result is $1700M. Adding the contingency and interest cost during construction, it amounts to $2590M.

The size of the ATW plant was taken at 3000 MWt, to match it with the standard LWRs to be serviced. One ATW system would process and transmute the spent fuel produced by four identical LWR plants of equal thermal power.

Using the total cost of $2590M derived for the actinide burner system and standard accounting practices [39], the surcharge on the nominal cost of nuclear-generated electricity (50 mils/kWhr) to destroy plutonium and the higher actinides is calculated to be 3 mils/kWhr. Further surcharges will apply to pay for the destruction of selected long-lived fission products.

This exercise in ATW system costing is not an attempt to put a hard number on costs. It does, however, provide a general indication on the cost issue. It seems reasonable to expect that the ATW systems will cost more than other types of nuclear reactors, and that their introduction in the nuclear energy production infrastructure will increase the cost of nuclear-produced electricity. These extra charges, however, will enable the nuclear industry to deal with the waste problem without depending on geologic storage. It is our belief that such disposal costs are not accurately known for the geologic repository solution at this point, given the uncertainty concerning the feasibility and public acceptance of this option.

It is possible that a centralized geological repository will not be realized, or that permanent nuclear waste storage will become acceptable only in conjunction with a transmutation process that destroys the actinides and the hazardous, long-lived fission products contained in the spent fuel. The increase in electricity rates introduced by transmutation systems might then be properly viewed as the adjustment that needs to be paid in order to take into account the real costs of waste disposal.

D.2.2.8. SUMMARY

In the past several years, the Los Alamos ADTT program has conducted studies of an innovative technology for solving the nuclear waste problem and building a new generation of safer and non-proliferant nuclear power plants.

Based on the LANSCE accelerator experience and the advent of high power accelerator technologies, drawing from the extensive development and operation knowledge bases of the Molten Salt Reactor Experiment at the Oak Ridge National Laboratory, the ALMR/IFR program at the Argonne National Laboratory and plutonium processing experience at Los Alamos, we are developing ATW as a nuclear concept with unique features and capabilities, and we deem it to be technically feasible with a reasonable amount of research and development.

The ATW concept destroys higher actinides, plutonium and selected fission products in a liquid-fuel subcritical assembly. The accelerator provides the extra neutrons and the level of subcriticality needed to perform these operation safely and effectively.

Among the special features of ATW, the following four are especially important:

1. ATW flexibly accepts nuclear spent fuel from any existing type of nuclear reactor using practical and efficient processes
2. Inhibits plutonium accumulation, proliferation and diversion at all levels
3. Effectively destroys plutonium, higher actinides (utilizing them for energy and neutron production), and selected fission products, removing them from the nuclear waste stream
4. Is fully compatible within the existing or foreseeable future nuclear infrastructure of base-load reactors

Admittedly, the ATW system uses technology which is not as developed as present-day commercial nuclear technologies, and actual realization of its objectives will take 15-20 years of hard work and development (see timeline in Fig. 11). No pure actinide burner, subcritical or otherwise, has ever been operated and the experience base for pyroprocessing (molten salt and liquid metal chemistry) is smaller than for more established techniques (i.e. aqueous processing). In the paper we have only hinted at the process chemistry required by ATW. While these processes have been identified and look promising, their full characterization is still far from complete, including the determination of the all-important decontamination factors in the waste streams.

We are clearly just beginning to explore the possibilities of these liquid-fueled accelerator-assisted nuclear systems. However, a molten salt fuel reactor was operated satisfactorily in the past and the liquid lead technology proposed for the neutron target is still well established in Russia. High-power linacs are currently on the drawing board in the US and, through the innovations introduced by the MSR and IFR program, pyroprocessing is now considered a viable alternative to aqueous processing. The ATW system will capitalize on these assets and take advantage of the experience.

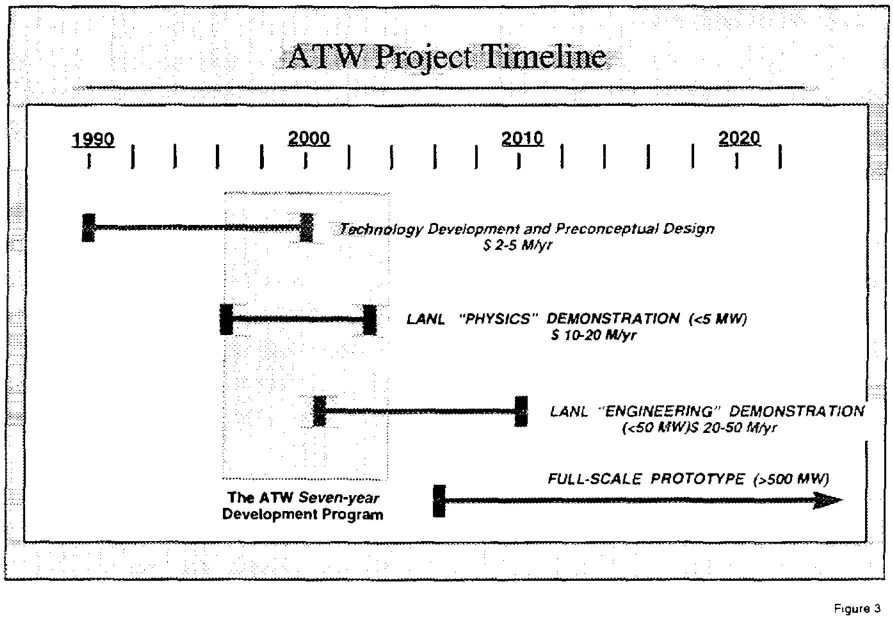


FIG. 11. Anticipated ATW project timeline at LANL.

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