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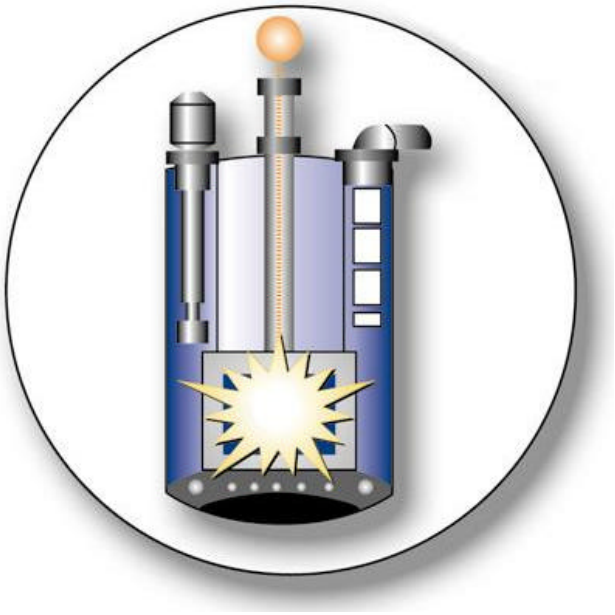

A Roadmap for Developing ATW Technology:

Target-Blanket Technology

September 1999

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ROADMAP FOR THE DEVELOPMENT OF ACCELERATOR TRANSMUTATION OF WASTE: TARGET AND BLANKET SYSTEM

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ROADMAP FOR THE DEVELOPMENT OF ACCELERATOR TRANSMUTATION OF WASTE: TARGET AND BLANKET SYSTEM

Report from the ATW Roadmap Target and Blanket Technical Working Group

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PREFACE

This report describes a roadmap to develop Accelerator Transmutation of Waste (ATW) target and blanket technology for an ATW system that would be deployable in about 25 years. The roadmap was developed for a deployment-driven scenario that was selected as part of the larger ATW roadmapping effort. After the results of the larger roadmapping effort became available, the ATW Roadmap Steering Committee determined that a "science-based" approach, rather than the deployment-driven approach, would best allow the technology to be developed to support a decision regarding ATW deployment and preferred technologies. The science-based approach would allow key feasibility, design and deployment questions to be addressed before large amounts of funds would be committed to a particular ATW development path.

Due to constraints in schedule for finishing the roadmap activity, the science-based approach was not incorporated into this report. However, the R&D tasks described herein, are those deemed necessary (in the judgment of those that prepared this report) to develop ATW target and blanket technology, regardless of the schedule in which they are pursued. Furthermore, the integral amounts of funds estimated for developing each of the constituent technologies are not expected to change much for variations in the R&D schedule.

At the recommendation of the Steering Committee, a five-year period of R&D was developed to provide information to support ATW technology and deployment decisions. Therefore, only those activities necessary for determining concept feasibility, design features, and deployment parameters were included. Pursuing the science-based path would delay the overall schedule reflected in this roadmap report by three to four years, and would change the required year-by-year allocation of funds. Specific activities delayed are those that would require large funding commitments, such as fabrication of prototype-scale heat removal components, and design and construction of large test facilities, such as the Target Test Facility, the Component Test Facility, and the ATW Demonstration Plant. However, it is believed that such an approach would allow a more prudent selection of technology options for ATW development.

ROADMAP FOR THE DEVELOPMENT OF ACCELERATOR TRANSMUTATION OF WASTE: TARGET AND BLANKET SYSTEM

EXECUTIVE SUMMARY

ES.1 Description of ATW Concept

The Accelerator Transmutation of Waste (ATW) concept has been proposed to eliminate long-lived actinides and fission products from spent light water reactor fuel, which would otherwise require disposal and isolation in a deep geologic repository. Although the need for disposal of radioactive materials, using a means such as a geologic repository, is not eliminated by implementing an ATW system, the requirements for such disposal are relaxed through elimination of a great proportion of the waste constituents that would require isolation for tens of thousands of years or longer and through the production of particularly impervious residual wasteforms.

As currently envisioned, an ATW facility consists of three major elements: (1) a high-power proton accelerator; (2) a spent fuel treatment / waste cleanup system; and (3) a subcritical target and blanket system that produces and utilizes an intense source-driven neutron flux for fission of transuranics and transmutation of fission products. The ATW Target and Blanket (T&B) system is to provide rapid destruction rates of actinides and efficient transmutation of key long-lived fission products, with minimal production of new waste. The objective of the T&B development program described in this roadmap report is to produce a fully integrated ATW T&B Prototype module, operating at full which could be used as a model for ATW development, should the Nation choose such a path.

The ATW T&B system consists of: 1) a spallation target producing the high intensity neutron source, and 2) a subcritical blanket (analogous to a reactor core) surrounding the target and containing the transuranic waste to be fissioned and long-lived fission products to be transmuted by neutron absorption. The source neutrons are generated by direct impingement of the accelerator proton beam onto a target material in a process called spallation.
The spallation neutrons are multiplied in the surrounding subcritical blanket, which contains the actinides and fission product transmutation assemblies. Because significant heat production occurs from fission in the surrounding blanket assemblies, adequate means for heat removal must be present, analogous to the case in critical fission reactors of similar power level. Leakage neutrons from the subcritical core are captured in the fission product assemblies, composed of rods containing long-lived fission products, such as technetium and iodine.

Subcriticality facilitates or accelerates tasks that might be more difficult or inefficient in critical systems. Subcritical systems do not rely on delayed neutrons for control and power change; they are driven only by the externally generated neutron source (*i.e.* by the proton beam coming from the accelerator), reducing the significance of reactivity feedbacks. Elimination of the requirements to maintain system criticality inherent with reactor systems allows flexible incorporation of fertile-free compositions, thereby maximizing the TRU destruction rates and relaxing the required separation efficiencies in the waste treatment steps. This makes possible, in principle, the destruction of any isotopes (actinides or fission products or mixture of both) with little concern for their neutronic behavior. Fertile materials are not needed to compensate for the neutronic uncertainties or provide negative doppler feedback to the fuel.

ES.1.1 Fuel and Fuel Processing

The ATW concept employs repeated cycles of exposure of waste materials to neutrons (fission and transmutation) and processing (cleanup and re-fabrication) of waste constituents. To avoid the large accumulation of waste in the recycle stage, procedures that allow the processing of materials with high radioactivity have been selected, namely pyrochemical processes. The selected ATW form for waste irradiation, further referred to as the ATW fuel, should conform to the choice of pyrochemical reprocessing.

Solid metallic fuels are the primary fuel choice for ATW because of their compatibility with pyrochemical fuel reclamation at the back end of the ATW cycle. An added bonus of metallic fuels is their higher thermal conductivity and greater resistance to thermal shocks (compared to ceramic fuels). Rapid temperature transients due to unplanned shutoffs of the accelerator driver can in fact occur in accelerator driven systems and result in thermal shocks that might not be acceptable with fuels of lower thermal conductivity. Although use of liquid fuels allows avoidance of burnup reactivity changes through simple, on-line addition of fissile material and removal of poisons, it was felt that, in comparison to solid fuels, liquid fuels present much larger unknowns associated with materials, operation and general engineering approach. Therefore, liquid fuel options were not addressed in detail in the roadmap, but may be considered in trade studies early in a research and development program.

ES.1.2 Fast Spectrum Operation

Selection of solid fuels (as opposed to liquid ones) leads to a selection of fast neutron spectrum operation over thermal spectrum operation, because transuranic elements have a considerably better ratio of fission to parasitic capture for fast neutrons than for thermal neutrons, the effects of fission product poisoning on reactivity are less severe, and radial power peaking is less pronounced. This roadmap effort does not address thermal-spectrum systems in detail, although, as in the case of liquid fuels, such systems should be considered in an R&D phase

Fast spectrum operation will require the use of nuclear coolants that do not significantly thermalize the neutrons before they have a chance to fission the transuranics. Sodium, lead-bismuth eutectic (LBE), and helium are suitable coolants for fast spectrum operations. In varying degree, considerable reactor experience exists for all three coolants, and they are compatible with the selection of metallic fuels. Each of them offers distinct advantages and drawbacks, so that careful assessment will be required to select the optimal choice.

ES.1.3 Preferred, Back-up and Alternative Options

The basic choices and range of alternatives in the ATW T&B system described in the previous section are captured in three basic T&B concepts which use a eutectic alloy of lead and bismuth (LBE), sodium, or helium as coolants. These concepts are predicated on previous reactor experience and expertise, and at this stage all the proposed ATW T&B systems are adaptations of existing reactor designs to subcritical operation.

In general, these adaptations replace the centermost part of the cores with a spallation neutron target and source diffuser. A vacuum beam tube is used to deliver a proton beam to the center of the core. A window separates the beam line from the spallation target. The spallation target can be integral with the core and use the same coolant fluid (as for the LBE-cooled ATW option or one

sodium-cooled ATW option) or be separated from the core, with its own cooling circuit. The core is divided in multiple fuel regions for shuffling (necessary to attain a more uniform power distribution), and burnable poisons are likely to be used to mitigate the reactivity decrease during burnup. Fission product transmutation regions are placed in the reflector volume. Because of the small cross-sections of fission products in the fast neutron spectrum, effective transmutation of the fission products might require some local moderation of neutrons.

Based on favorable neutronic and thermal-hydraulic properties, and on the potential simplicity of engineering design, the T&B system preferred for this road-map uses LBE arranged in a single loop configuration as target, coolant and reflector, with solid metallic fuel elements and moderated assemblies for fission product transmutation. The LBE technology, although developed and implemented in Russia, remains to be transferred and mastered in the US. Therefore, a backup T&B system is also proposed, consisting of a sodium-cooled target and blanket system with a solid tungsten target. The maturity of the sodium technology in the US allows the use of reasonably valid estimates for the implementation of these systems on a large scale, and therefore the sodium cooled T&B system is considered the reference concept for estimating the cost of large-scale implementation of ATW. Alternative concepts based on helium gas cooling (operating both in the fast and in the thermal spectrum regions) are also proposed and will be analyzed in the initial trade study phase, should further ATW research and development be authorized, but are not featured in the present roadmapping activity.

ES.2 ATW Target and Blanket (T&B) Development Program

The deployment of a series of ATW plants by the mid-21st century will require a series of carefully planned steps whose aim is to address and resolve the unknowns and uncertainties existing between today's technology and that required to operate successfully the nth-of-a-kind plant. The roadmap described in this document was developed to support the deployment scenario assumed for the larger ATW roadmapping effort. A science-based approach was also developed, which delayed commitments to any particular technology options until certain technical questions are resoled through a focused R&D program; that science-based program is described elsewhere

ES.2.1 Trade Studies and Base Technology Development (2001-15)

A two-year period of trade studies, where several options and technologies will be carefully assessed, is expected at the start of the Program, with initial down-selections at the end of the second year. The adoption of proven core and thermal-hydraulics technologies and the use of solid fuels, which are readily testable extensions of known technology, provide an initial basis for the required research and development for the ATW T&B system.The successful realization of the ATW prototype will require addressing specific issues in the following areas:

Fuel
Coolant Chemistry and Materials Compatibility
Heat Removal Systems
Nuclear Design and Safety
Spallation Targets

The base technology development phase will address technology transfer, conceptual component design, development of materials database, and feasibility issues specifically related to implementation in subcritical systems and will finalize important design choices. It is envisioned

that this phase will be particularly emphasized in the first eight years, with final decisions on the fuel, target and coolant technology in 2008, but it will continue until 2015 when the ATW Demo becomes operational. The specific R&D activities proposed for this phase will address issues of feasibility such that a decision whether to initiate the Demonstration and Large-Scale Integration phase can be justified. Final decisions on the fuel, target and coolant technology are expected in 2008. Confirmation work on ATW fuel will continue until 2015.

ES.2.2 Demonstration and Large-Scale Integration (2006-25)

Starting in 2006, activities will commence towards the realization of an integrated ATW Demonstration Facility (ATW Demo). As currently envisioned, the facility will start operation at 5% power (30 MW) in 2015. Power will be slowly increased, through increases in beam power and core reactivity, up to full power in ten years (2025). At low power levels, the facility, in addition to verifying system integration, will provide an important test bed for fuel and materials studies, studies of system dynamic behavior, abnormal conditions, and thermal-hydraulics of forced and natural convection. The ATW Demo will initially utilize proven fuel concepts, gradually converting to the final ATW T&B fuel system.

At full power in 2025,the ATW Demo becomes the ATW Prototype, and in addition to demonstrating the operability of ATW it will be used to quantify ATW economic performance, establish the scaling to the nth of a kind and determine ATW infrastructure requirements.

ES.3 The ATW T&B R&D Roadmap: Issues and Tasks

Consistent with the grouping of technical issues identified in the previous section we have divided the ATW T&B roadmap into five separate categories:

Nuclear Design and Safety
Coolant Chemistry and Materials Compatibility
Heat Removal and Ancillary Systems
Target Technology
Fuel Technology

For each of these categories, the specific issues associated with the reference technologies were identified and a plan for addressing those issues and developing the technologies to be part of a licensable ATW system was developed. Where applicable and possible, alternatives to the reference technologies were identified, and plans for development of those alternatives were incorporated into the roadmaps. For each category, the roadmaps are described as a combination of tasks designed to provide information for key decision points or for design and construction activities, as outlined in the roadmap report by the System Scenarios and Integration Working Group. Estimates of the personnel and financial resources required for the completion of each task were developed, as well as the envisioned schedule by which the tasks can be completed. The issues and tasks proposed to address those issues are summarized below for each category.

ES.3.1 Nuclear Design and Safety

The Nuclear Design and Safety task determines core design features, quantifies performance characteristics, confirms system nuclear safety, and supports licensing. Specific issues to be addressed/resolved by the Nuclear Design and Safety R&D include the following.

- Design for degree of subcriticality
- Accommodation of burnup reactivity loss
- Accommodation of feed composition variation with recycle
- Long-lived fission product incineration strategy
- Determination of dynamic behavior
- Safety analysis and confirmation
- Validation of simulation tools

Efforts to address these issues will begin with system trade studies, blanket design and fuel management studies for ATW concept options, and establishment of validated simulation tools. Initially, general safety issues will be addressed, but as blanket design efforts begin for specific ATW Demonstration Plant and Prototype Plant configurations, then analyses of accident scenarios will be performed.

ES.3.2 Coolant Chemistry and Materials Compatibility

The technical issues addressed by the Coolant Chemistry and Materials Compatibility portion of the Target and Blanket roadmap report include: LBE conditioning, cover gas control, fuel and structural materials corrosion in LBE, structural materials performance, heat removal, and waste management.

Effective technology transfer from Russia is essential to successful implementation of LBE technology within the aggressive schedule assumed for this roadmap. Specific activities are configured to allow U.S. personnel to access, understand and evaluate knowledge obtained from Russia, and then to establish domestic experience with the technology through laboratory testing and design efforts. R&D efforts proposed to address these issues include the design and operation of a LBE loop to be used for study of chemistry control techniques, materials compatibility studies, and heat transfer and flow characteristics (an effort which would be coordinated with those addressing LBE heat transfer and component design issues). It is believed that Russian stainless steel alloys specifically developed for LBE applications may have the best promise for ATW application. However, considerable effort will be expended to identify materials that will perform acceptably in a high-temperature lead-bismuth environment. The effect of spallation products on LBE coolant chemistry and materials compatibility will also be determined.

Because the technology for maintaining sodium coolant chemistry and for ensuring compatibility of structural materials with sodium is well developed and mature, only LBE was considered in this effort. It was determined that, with the exception of the effects of spallation products on the coolant chemistry, those issues for a sodium-cooled ATW reactor need not be the subject of a focused R&D plan.

ES.3.3 Heat Removal and Ancillary Systems

The Heat Removal and Ancillary Systems portion of the roadmap is modeled on U.S experience in the development of heat removal and ancillary systems for sodium-cooled systems. Because the U.S. has essentially no experience with lead or LBE as a nuclear coolant, considerable effort will be required to design and develop components for heat removal from a LBE-cooled target and blanket system. For the purposes of proposing a research and development program for LBE heat removal and ancillary systems, it is assumed that the relevant technology can be transferred from

Russia to the U.S. Conversely, given the extent of U.S. experience, efforts for a sodium-cooled ATW system can be directed toward design of components specifically for the ATW Demonstration Plant without up-front development effort.

Consistent with the approach used for development of relatively large reactor components during the U.S. Liquid Metal Fast Breeder Reactor (LMFBR) and Advanced Liquid Metal Reactor (ALMR) programs, this roadmap calls for base technology assessment and development (including technology transfer) to be performed by the U.S. National Laboratories, with development, design, and fabrication of large-scale components to be performed by industry. The proposed program includes development for the following components in an LBE system:

- primary and secondary pumps
- intermediate heat exchanger
- steam generator
- vessel features, such as head, support structures, and fuel handling equipment
- ancillary systems such as coolant cover gas system and cold trap and filter

ES.3.4 Target Technology

Two prospective target geometries are proposed for development for an LBE-cooled ATW system: an integral target, and a separate target. The integral LBE target is comprised of a thimble tube that is inserted into the LBE coolant in the middle of the core. The tube maintains the pressure boundary between the accelerator vacuum and the core coolant. The end of the tube is directly in the path of the proton beam, and is cooled in an integral sense with the reactor coolant. The separate LBE target uses a "windowless" design in that no structure is directly in the path of the beam other than LBE. The coolant for this target is separate from the core coolant, and requires a separate cleanup and heat removal system. This structure would experience less irradiation damage than the integral target.

The sodium-cooled target currently envisioned would be solid tungsten, clad in stainless steel (HT-9). A thimble tube, which provides the pressure boundary between the accelerator vacuum and the sodium coolant, would be located above the target assembly, and would be cooled by the sodium. The thimble tube material and the stainless steel cladding for the tungsten target would experience a combined proton and neutron environment in the presence of the sodium coolant.

Regardless of the technology chosen, the major research and development activities that are required for the successful demonstration of an ATW target include:

- selection and qualification of metals and alloys for service under anticipated target conditions, particularly for the target window,
- acquisition of spallation physics data and development of methods and codes for predicting spallation neutron and product yield,
- testing of target designs, including window designs, under proton irradiation in a new Target Test Facility,

ES.3.5 Fuel Development

Two conceptual fuel forms have been proposed as best suited for use in an ATW system. The reference fuel form is a dispersion fuel comprised of TRU-Zr metallic alloy fuel particles imbedded

in a Zr matrix and clad in stainless steel. This form is expected to have good high-burnup irradiation performance, excellent shock resistance in the event of accelerator beam interruption, and offers a potential fabrication route that may not require high-temperature processing, thus enhancing the retention of volatile TRU elements during fabrication. The alternate fuel form is a TRU-Zr metallic alloy slug, with a liquid-metal thermal bond, clad in stainless steel, much like Experimental Breeder Reactor II (EBR-II) fuel or fuel developed for the Integral Fast Reactor (IFR) concept. This fuel form offers a simplified fuel fabrication route that lends itself well to remote hot cell techniques, but requires fabrication at elevated temperatures where retention of volatile TRU elements may be difficult. No material property or irradiation performance experimental data exists for either fuel form.

The currently identified issues associated with the proposed fuel forms are related to fabrication and irradiation performance. Fuel development activities will include development and demonstration of remote fabrication techniques and establishment of a properties and irradiation performance database that will support licensing of the fuel form for ATW operation. Irradiation tests are proposed for insertion into the Advanced Test Reactor (ATR) in the U.S. and (if arrangements can be made) into the BOR-60 in Russia. Alternatively, an irradiation test program for the Fast flux Test Facility (FFTF) is also proposed, should the FFTF become available to support an ATW development program.

1. GENERAL APPROACH

1.1 Objective

The objective of the Target and Blanket (T&B) development program is to produce a fully integrated ATW T&B Prototype module, operating at full power and ready to be used as the model for a widely deployable burner of transuranic actinides and selected long lived fission products by 2025.

As explained previously, an ATW facility consists of three major elements: (1) a high-power proton accelerator; (2) a spent fuel treatment / waste cleanup system; and (3) a subcritical burner (target and blanket system) that produces and utilizes an intense source-driven neutron flux for fission of transuranics and transmutation of fission products. The ATW T&B system is to provide rapid destruction rates of actinides and efficient transmutation of key long-lived fission products, with minimal production of new waste. The ATW concept and technologies presented here are the result of many years of development at LANL as well as other major international research centers.

Subcriticality does not render ATW "safer" than critical reactors. Rather, subcriticality facilitates tasks that might be more difficult or inefficient in critical systems. Subcritical systems do not rely on delayed neutrons for control and power change; they are driven only by the externally generated neutron source (*i.e.* by the proton beam coming from the accelerator). The significance of reactivity feedbacks is reduced.

Elimination of the requirements to maintain system criticality inherent with reactor systems allows flexible incorporation of fertile-free compositions, thereby maximizing the TRU destruction rates and relaxing the required separation efficiencies in the waste treatment steps. This makes possible, in principle, the destruction of any isotopes (actinides or fission products or mixture of both) with little concern for their neutronic behavior. Fertile materials are not needed to compensate for the neutronic uncertainties or undesirable reactivity responses of the fuel. Compensation for the reactivity decrease associated with extended burnup is motivated primarily by the desire to maintain plant power output rather than sustaining a critical reaction, as is the case with reactors.

Because of its subcritical mode of operation, the ATW is well suited to fission actinide mixtures that: (1) are not well characterized, (2) fission or transmute very poorly or not at all in some reactors; and (3) have potentially unstable reactivity responses. This includes higher actinides such as neptunium (the worst contributor to an oxidizing repository long-term performance uncertainties), americium and curium, all isotopes of plutonium and the long-lived fission products Technetium and Iodine.

1.2 Key Features of the ATW T&B

The ATW waste burner or T&B system consists of: 1) a spallation target producing the high intensity neutron source, 2) a subcritical blanket (analogous to a reactor core) surrounding the target and containing the transuranic waste to be fissioned and long-lived fission products to be transmuted by neutron absorption.

The source neutrons are generated by direct impingement of the accelerator proton beam onto a target material. The process, called spallation, has been successfully employed for neutron science research to produce intense neutron fluxes.

The spallation neutrons are multiplied in the surrounding subcritical blanket, which contains the actinides and fission product targets. Because significant heat production occurs from fission in the surrounding blanket assemblies, adequate means for heat removal must be present, analogous to the case in critical fission reactors of similar power level. Leakage neutrons from the subcritical core are captured in the transmutation blanket, composed of transmutation elements with long-lived fission products, such as technetium and iodine.

2. DESIGN OPTIONS FOR ATW T&B

Based on previous work at LANL as well as other major international research centers, we have developed basic choices for the fuel and neutronics design of the ATW T&B system.

2.1 Solid Metallic Fuel

The ATW concept includes repeated cycles of exposure to neutrons (fission and transmutation) and processing (cleanup and re-fabrication) of waste constituents. To avoid the large accumulation of waste in the recycle stage, procedures that allow the processing of materials with high radioactivity have been selected, namely pyrochemical processes. The selected ATW form for waste irradiation, hereafter referred to as the ATW "fuel", should be compatible with the choice of pyrochemical reprocessing.

Solid metallic fuels are the primary fuel choice because of their compatibility with the pyrochemical fuel reclamation at the back end of the ATW cycle. An added bonus of metallic fuels is their higher thermal conductivity and good resistance to thermal shocks. Rapid temperature transients due to unplanned shutoffs of the accelerator driver can occur in accelerator driven systems and result in thermal shocks that might not be acceptable with fuels having low thermal conductivity and poor shock resistance.

Liquid fuel systems have been considered and examined at great length, both water-based (oxide slurries and nitrate solutions) and molten salts. Liquid fuels provide the obvious advantage of being able to avoid large burnup reactivity losses (typical of ATW systems which do not contain fertile materials) by straightforward addition of fissile material. They also present much larger unknowns associated with materials selection, operation and general engineering approach to design, construction and licensing.

2.2 Fast Spectrum Operation

The ATW T&B system is essentially a nuclear facility dedicated to the fissioning of transuranics and to the utilization of surplus neutrons to transmute long-lived fission products. After the choice of solid fuels (as opposed to liquid fuels) is made, fast neutron spectrum operation is preferred over thermal spectrum operation because transuranic elements have a considerably better ratio of fission to parasitic capture for fast neutrons than for thermal neutrons, the effects of fission product poisoning on reactivity are less severe, and radial power peaking is less pronounced.

The higher probability of fission associated with a fast neutron spectrum lessens the buildup of transuranics with higher atomic number and additionally provides a larger number of excess neutrons per fission for use in the transmutation of the long-lived fission products. Cross sections are smaller overall in the fast-spectrum operating regime, however, which leads to overall larger inventories than for thermal systems. However, burn-down strategies can ultimately be implemented in ATW systems to limit the final inventory of transuranics at the end of operations.

Fast-spectrum ATW systems will require the use of coolants that do not significantly thermalize the neutrons before they have a chance to fission the transuranics. Sodium and Lead-Bismuth Eutectic (LBE) are the primary coolants considered for fast spectrum operations in the main text of this document. Sodium and LBE coolants each offer distinct advantages and drawbacks, so a careful assessment will be required to select the optimal choice.

Helium is also a potentially suitable coolant for ATW application, well compatible with fast-spectrum operation and the choice of metallic fuel. Because the roadmapping effort focused mainly on liquid metal coolants and was not able to develop an assessment of the helium option, helium –system development is not addressed in the bulk of this document; however, a brief overview of helium use in ATW systems is presented in the appendix.

Table 2.1 Advantages and Disadvantages of LBE and Sodium Coolants for an ATW System

Coolant	Reactor experience (reactor-years)	Advantages	Disadvantages
Lead-Bismuth Eutectic (LBE)	80	Optimal cross sections Excellent reflector Excellent target Very high boiling point Low void coefficient No reaction with air, water Excellent natural convection	Polonium production Optically opaque Not implemented outside Russia
Sodium	260	Highly efficient coolant High power density possible Non corrosive Good natural convection Mature technology in US, France and Japan	Fire hazard Needs separate reflector Needs separate target Optically opaque Positive void coefficient
Helium	100	Neutronically transparent Optically transparent High conversion efficiency No coolant mixed waste	High pressure Needs separate reflector Needs separate target Large size vessel

2.3 Thermal Spectrum Alternatives

Because fission product poisoning has a large impact in the thermal spectrum, the use of a thermal spectrum for fertile-free ATW systems is possible only if the fuel fissile content can be regularly

replenished during operations. Such systems would implement liquid fuels, like molten salts, or "semi-fluid fuels, as does the pebble bed gas-cooled reactor concept. While not specifically examined in this roadmap effort, due consideration of these systems may be appropriate for the trade study phase, which will lead to technology and system selection.

3. ATW TARGET AND BLANKET SYSTEM

3.1 T&B Options: Neutronics and Thermal-hydraulics

The basic choices and range of alternatives in the ATW T&B system described in the previous section is captured in each of the T&B concepts which use sodium and LBE as coolants. These concepts are predicated on previous reactor experience and expertise, and at this stage the proposed ATW T&B systems are adaptations of existing reactor designs to subcritical operation.

In general, these adaptations replace the centermost part of the cores with a spallation neutron target and source diffuser. A vacuum beam tube directs a proton beam to the center of the core. A window separates the beam line from the spallation target. The spallation target can be integral with the core and use the same coolant fluid (as for one LBE-cooled ATW option or the sodium-cooled ATW option) or separated from the core, with its own cooling circuit. The core is divided in multiple fuel regions for shuffling (necessary to attain a more uniform power distribution), and burnable poisons are likely to be used to mitigate the reactivity decrease during burnup. Fission product transmutation regions are placed in the reflector volume. Because of the small cross-sections of fission products in the fast neutron spectrum, effective transmutation of the fission products will require local moderation of neutrons. The two liquid metal options are described in the following sections.

3.1.1 Sodium-cooled, with a solid tungsten target, metallic fuel, in a configuration essentially identical to the ALMR reactor concept proposed in the US.

In this concept a proton beam is sent into a subcritical core cooled by liquid sodium. The source neutrons are produced by spallation on a composite tungsten target cooled by the primary sodium coolant. A window cooled by the same sodium primary coolant provides the separation between the accelerator (beam guide) and the subcritical core. The core is constituted of a hexagonal array of metallic fuel element bundles. Fission heat generated in the subcritical core is removed by the sodium forced circulation through intermediate heat exchangers into a secondary sodium loop and from there into a conventional steam generator. Due to the fast spectrum operation, there is a substantial excess of neutrons generated in the process of transuranic fissioning in the ATW core. These neutrons are available for moderation and capture in transmutation assemblies placed in the surrounding blanket, containing the long-lived fission products of concern, such as technetium-99 and iodine-129.

The US has a long history in the development of sodium cooled fast reactor systems. Several reactors were built and operated successfully including EBR-I, EBR-II and FFTF. The EBR-I holds the distinguished achievement of being the first nuclear reactor in the world to produce electricity. The EBR-II and FFTF tested fuel and structural materials for the Liquid Metal Fast Breeder Reactor (LMFBR) program and the more recent integral Fast Reactor (IFR) and Advanced Liquid Metal Reactor (ALMR) programs. Although the IFR and ALMR programs were recently canceled in the U.S. and the test reactors are shut down, a wealth of data and experience

Were accumulated. For example, special alloys such as HT-9 were developed and proven in the FFTF to very high dose

To apply sodium reactor technology to an accelerator driven system, development activities should include the design and testing of a sodium-cooled solid target, the materials issues with operating in the proton beam, and the effect of the proton beam on sodium chemistry. It is expected that the basic technology required will be substantially similar to that developed for the ALMR program, and can be adapted to the ATW concept with minimal modifications. Figures showing the major features of the ALMR and the layout of an ALMR core are presented in Figures 3.1 and 3.2.

The current concept of a sodium-cooled target is solid tungsten, clad in stainless steel and housed in an array of hexagonal assemblies similar to those used in FFTF. The target assemblies would be sufficiently long to completely attenuate the beam. A thimble tube, which provides the pressure boundary between the accelerator vacuum and the sodium coolant, would be located above the target assembly, and would be cooled by the sodium. The thimble tube material and the tungsten cladding would experience a combined proton and neutron environment in the presence of the sodium coolant.

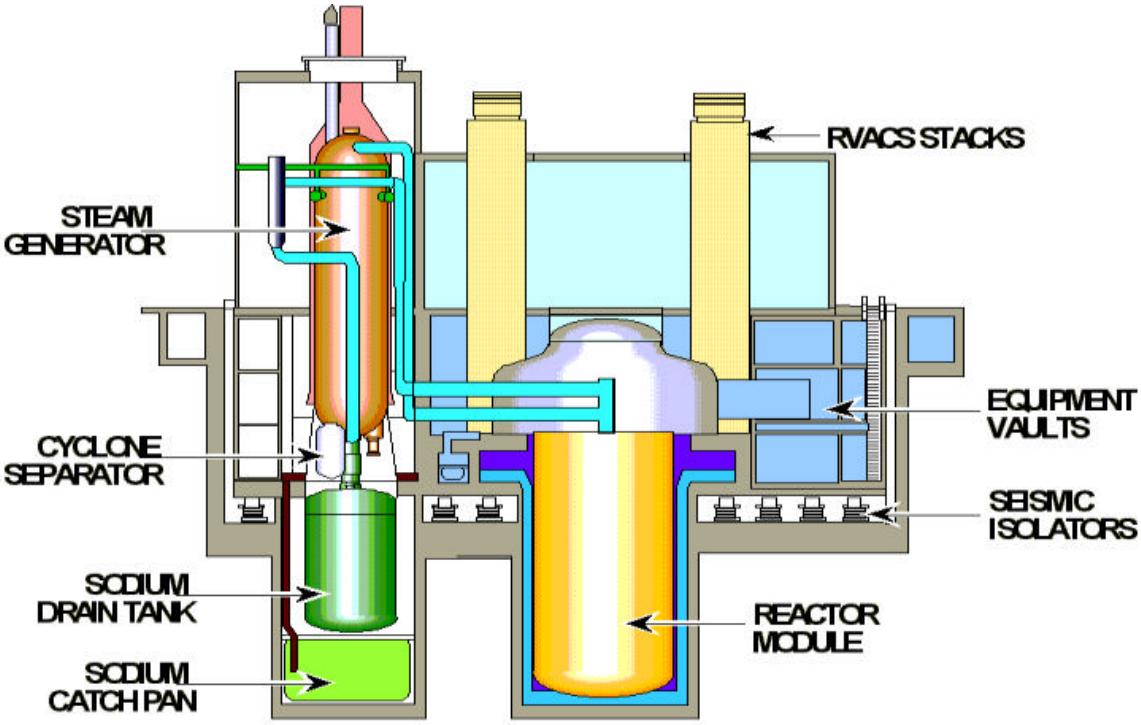

Figure 3.1. Major features of the advanced Liquid Metal Reactor Design

3.1.2 Lead-bismuth eutectic-cooled, with integrated LBE target, metallic fuel, in a configuration similar to that of the Russian BRUS-300 reactor concept

In this concept a proton beam is sent into a subcritical core cooled by liquid lead bismuth eutectic. The source neutrons are produced by spallation directly on the LBE primary coolant, although incorporation of a separate LBE target is an option. A window cooled by the same LBE primary coolant provides the separation between the accelerator (beam guide) and the subcritical core. The

source neutrons multiply in the surrounding subcritical core, constituted of hexagonal array of metallic fuel element bundles. Fission heat generated in the subcritical core is removed by the LBE forced circulation through intermediate heat exchangers into a secondary LBE loop and from there into a conventional steam generator. Because of its higher atomic weight, the LBE system operates in a faster neutron spectrum regime than the sodium-cooled design. There is a substantial excess of neutrons generated in the ATW blanket, which are available for moderation and capture in the transmutation assemblies placed in the blanket.

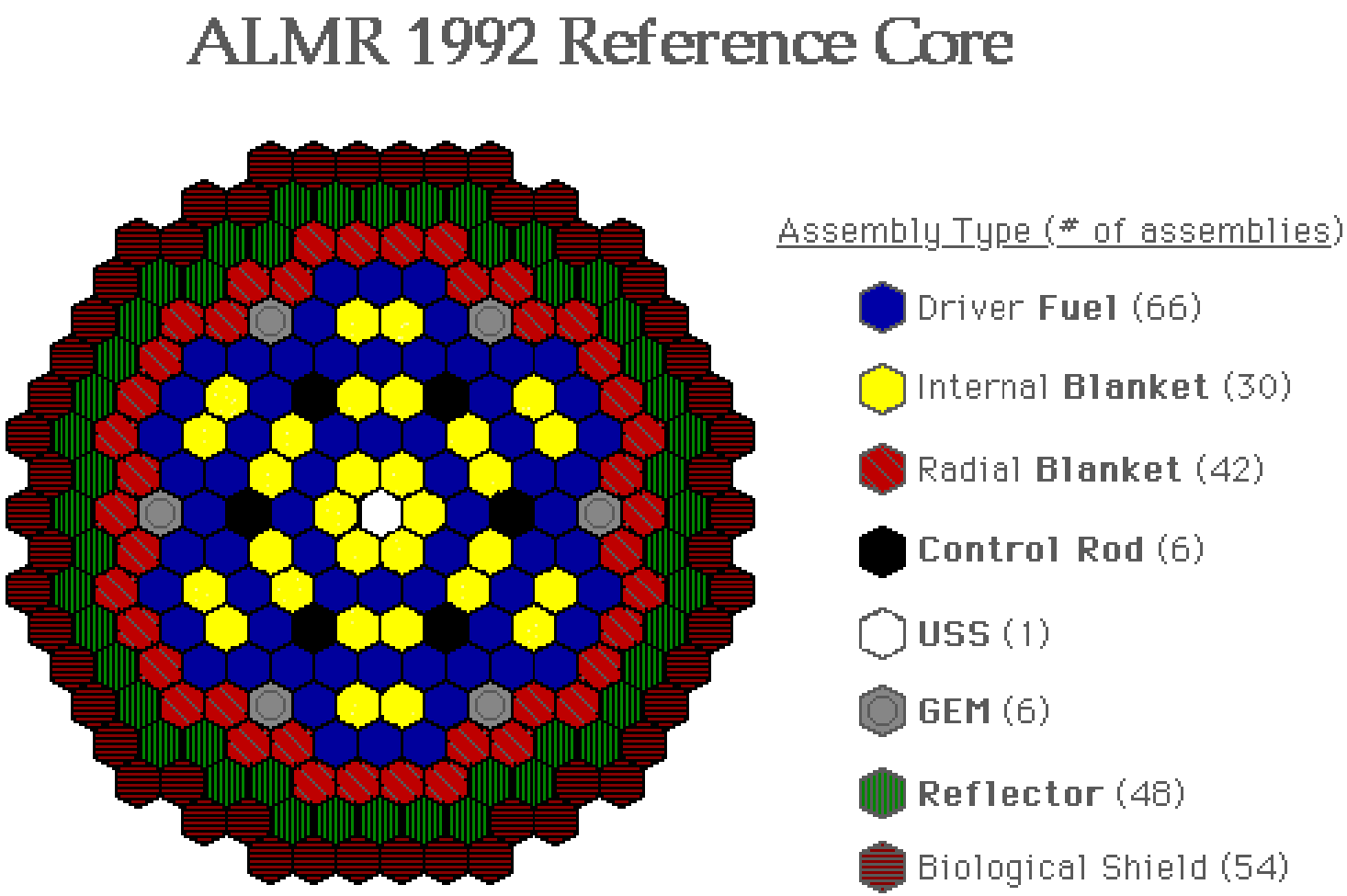

Figure 3.2. Core layout for the reference core of the Advanced Liquid Metal Reactor.

Table 3.1 Comparison of Natural Circulation Characteristics of Major Coolants

Coolant	Reactor Reference	T_{in}/T_{out}	Driving Head		Heat Capacity (C_p)		Margin to Boiling (T_{boil}-T_{out})	
		°C	g/cm^3	Norm'd (A)	J/m^3°Kx10^6	Norm'd (B)	°C	Norm'd (C)
LBE	BRUS-300	340/510	0.201	1	1.53	1	1160	1
Lead	BREST	420/540	0.146	0.71	1.50	0.98	1185	1.02
Sodium	ALMR	388/535	0.036	0.18	1.08	0.71	350	0.30
Water	4-loop PWR	283/325	0.073	0.36	2.93	1.91	18	0.02

LBE (the eutectic composition is 44.5wt% Pb - 55.5wt% Bi) possesses some unique physico-chemical properties, making it an excellent nuclear coolant and spallation neutron source. Its low melting point (123.5°C), high boiling point (1670°C) and very low vapor pressure allow for a wide operating temperature range, eliminates coolant boiling and enhances circuit safety. The high density of LBE combined with wide permissible temperature range offers extraordinary natural convection cooling capability for enhanced passive safety (see Table 3.1). LBE's low chemical activity inhibits violent reactions (fire and explosion) with air and water. The sealed vessels and circuits readily prevent air-borne lead contamination from exceeding established industrial standards.

Although LBE can be rather corrosive and can be contaminated by solid admixtures due to interaction with construction materials and oxygen, Russian-developed "heavy metal technology" has been successful in neutralizing these adverse effects through proper materials selection and active control of oxygen activity in the coolant. In essence, this technology involves maintaining the oxygen level in LBE coolant so as to stabilize a self-healing protective oxide film on the surface of the structural materials to prevent corrosion, while leaving no excess oxygen in the lead to form lead oxide. The structural material alloy compositions were developed to enhance the protective nature of the oxide films.

LBE has a very high useful neutron production from spallation (30 neutrons per proton per GeV) and extremely low neutron capture cross section. The neutron transparency of LBE allows for a widely spaced core with relatively lower pressure drops and pumping power requirements. The coolant is also self-shielding against gamma radiation. Because of the unique nature of Pb-208 (the most abundant lead isotope) and the high atomic weight and neutron transparency of both lead and bismuth, neutrons in LBE have a small average lethargy. LBE makes an ideal reflector and "storage medium" for neutrons and thus is ideally suited for transmutation applications. The integration of nuclear coolant, spallation target and reflector using the same fluid in the ATW LBE target concept drastically simplifies the subcritical burner design by streamlining flow configuration and by removing target and reflector structures. Early problems with LBE nuclear systems (corrosion of structural materials, oxygen balance, and handling of the Polonium generated through neutron irradiation) have reportedly

been solved in the course of developing LBE systems for submarine propulsion reactors in Russia. The Russians deployed this technology

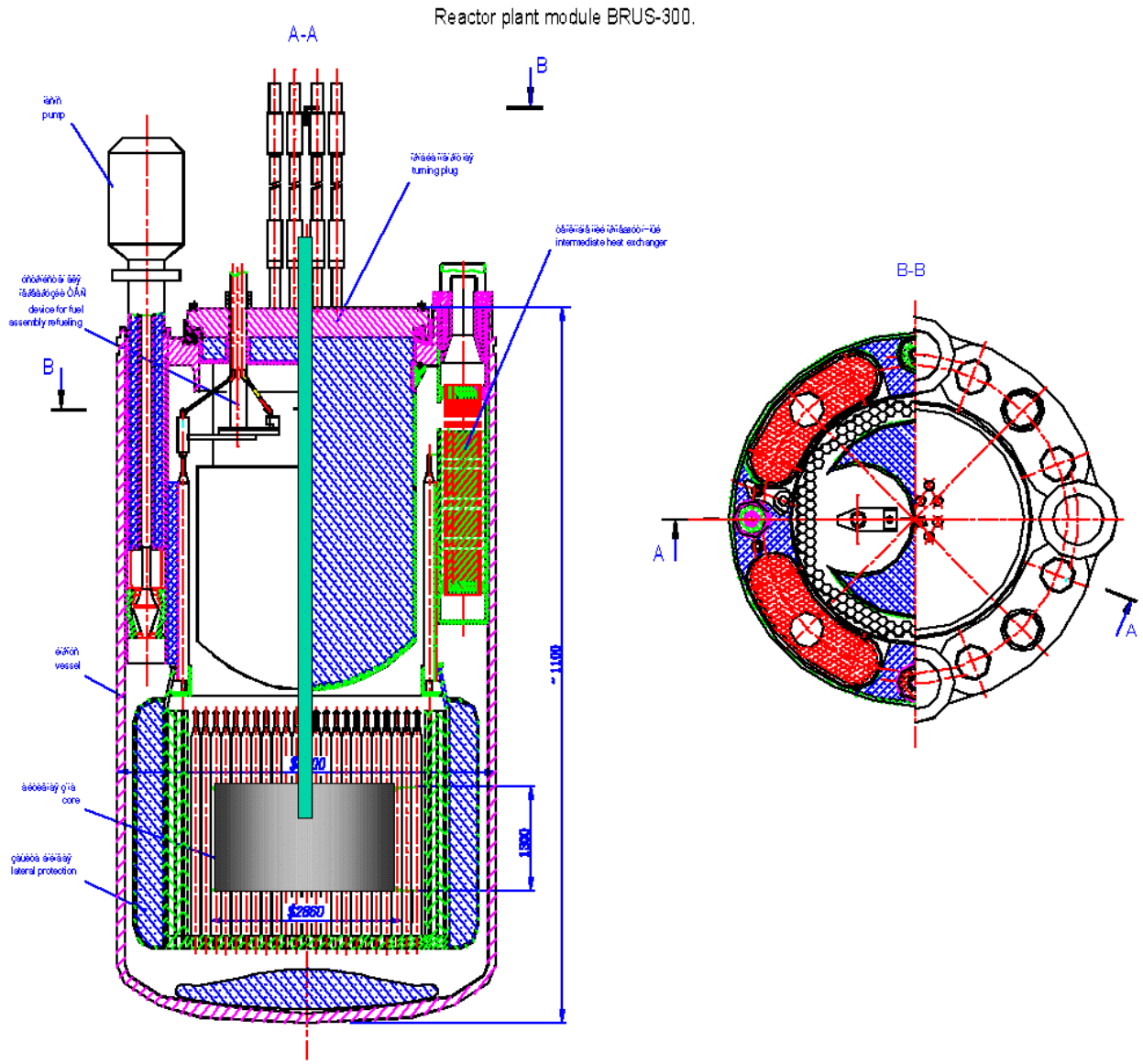

Figure 3.3 Conceptual LBE-Cooled ATW target and blanket based on the Russian BRUS-300 concept.

Table 3.2 Design Parameters for the ALMR and BRUS-300

	ALMR	BRUS-300
Rated thermal power, MW	840	1000
Primary circuit: coolant core outlet temperature, °C core inlet temperature, °C core pressure drop, MPa average flow, m^3/s pumps, number x power, kW heat exchangers, number x type	 Na 499 360 0.49 5.4636 4 x 1871 2x Shell/Tube	 Pb-Bi 510 340 4 4
Intermediate circuit: coolant temperature at heat exchanger outlet, °C temperature at heat exchanger inlet, °C flow rate, m^3/s pumps, number x power, kW	 Na 477 327 5.110 2 x 3402	 Pb-Bi 460 290
Steam generating circuit: steam generators working medium steam capacity, t/h superheated steam pressure, MPa superheated steam temperature, °C feedwater temperature, °C	 1 x helical coil water-steam 1889 15.17 455 216	 water-steam 1200 4,0 435 150
Power conversion efficiency	36%	
Fuel composition Core loading, kg	U-23 Pu-10Zr (BOEC loading) --3255	Zr-U / UO_2 ~3750
Refueling	3 x 23 mo.	
Vessel Diameter, cm Height	316 SS 917 1935	 600 1100
core height	107 cm	
core layout	Radial Heterogeneous	
coolant flow rate, m^3/s	5.4636	
fuel elements	126 Fuel S/A 60 Int. Blkt. S/A 48 Rad. Blkt. S/A	217 pins/ S/A 127 pins/ S/A 127 pins/ S/A
structure and clad	HT-9/HT-9	
external side length	Pin: 3.84 m S/A 4.7752 m	
outside distance across flats	0.142875 m	
box wall thickness	4.445 mm	
fuel rod OD	7.4422	
fuel rod pitch	8.92 mm	
number of rods per element	217 (fuel)	
number of elements	126 fuel S/A 60 intl. Blkt. 48 rad. Blkt.	
Fuel assembly run, years	5.75	3

for nuclear submarine reactors and have accumulated over 80 reactor-years of experience (mostly in 150-MWt units) and have recently proposed extending the technology to pure lead systems. Recently, considerable insight was gained in the Russian implementation of the LBE technology, which is also favored in Europe and Japan for ATW-like applications and generated interest in the U.S. for use in future "proliferation-resistant" reactors. In view of the large potential benefit to the implementation of the ATW concept, we are of the opinion that a strong technology transfer effort should be pursued in the ATW R&D phase to fully master this technology.

A schematic of the ATW LBE-based T&B is shown in Figure 3.3. The design is an adaptation the Russian BRUS-300 LBE reactor concept, which was developed to be the LBE counterpart to the ALMR sodium reactor, with similar engineering features.

3.2 Description of ATW T&B Design Options

At this stage of development, the ATW T&B designs are straightforward adaptations of reactor concepts to subcritical operation. At present, the T&B concepts envisioned for the sodium-cooled and LBE-cooled variants of an ATW system will have parameters very similar to those for the ALMR and BRUS-300. Various design parameters for these two reactor designs are presented in Table 3.2.

3.3 ATW Fuel Options

As the ATW system is currently envisioned, its fuel must meet four functional requirements: 1) the fuel form must reliably contain fission products during normal operation to design exposure and during certain off-normal events; 2) the fuel form must maintain a coolable geometry and maintain fissile material in a predictable configuration; 3) The fuel form must be amenable to recycle/refabrication with minimal losses; 4) the fuel form must interface efficiently with the ATW pyrochemical back-end reprocessing.

A metal-matrix metallic dispersion fuel, specifically, a TRU-10Zr fuel particle dispersed in a zirconium matrix is proposed as the reference blanket fuel form, with the TRU-Zr fuel particle volume fraction being adjusted to provide a bulk fuel composition of Zr-25TRU. A similar fuel form, consisting of U-Mo alloy fuel particles dispersed in an aluminum matrix, is being developed for research and test reactors.

Metallic fuel has typically been fabricated using vacuum injection casting at temperatures considerably above the liquidus temperature of the fuel alloy. Injection casting of a Zr-10TRU alloy is expected to require prolonged heating to above 1800 C, which is likely to volatize a significant fraction of the volatile TRU elements from the melt. Therefore, a low-temperature fabrication process appears to be desirable to enable the retention of transuranics into the fuel form. It is believed that processes currently in development for fabrication of low-enrichment research and test reactor fuel can be successfully employed.

3.3.1 Primary candidate: Dispersion Fuel

The primary fuel choice for ATW assemblies is a dispersion fuel form consisting of 20–40 μm particles composed of ~25% Zr and ~75% TRU (mostly Pu) embedded within a matrix of Zr yielding an overall composition of ~75% Zr and 25% TRU. Several process steps must be

developed, and they depend to a large extent on the specific composition and morphology of the feed material, i.e., the product of the fuel recycle processes. These steps are: (1) the alloy for the fuel particles is formed; (2) it is then "powderized" to form the desired small particles; (3) it is blended with the zirconium matrix powder (which is assumed to be another output stream from the electrorefining step and, hence, must also be powderized) and possibly some sort of binder for subsequent steps; (4) pellet compacts are formed from the blended powders, either hot-pressed or sintered after pelletization; (5) they are inspected (those that pass proceed and those that don't are crushed and fed back into the blending step); (6) they are inserted (perhaps along with a bond material) into cladding tubes; (7) the elements are sealed; (8) the fuel elements are inspected (with recycle of those that fail); and finally (9) the elements are combined into assemblies for return to the reactor.

3.3.2 Backup candidate: Cast Fuel

A backup means of fuel fabrication should be carried along in the ATW program should some characteristic of the primary fuel choice or its fabrication prevent its deployment. Currently this backup method consists of injection casting of long metallic slugs in a manner similar to that employed at ANL for EBR-II fuel since the 1960's and was intended, more recently, for IFR fuel. This fuel would have the disadvantage of not being as thermal-shock resistant as dispersion fuel and will require processing at temperatures that are expected to volatilize significant amounts of Am and perhaps some other actinides. In many other respects, it is a simpler process, involving the following steps: (1) metallic products from the ATW fuel feed or recycle processes are combined, melted, and cast into molds; (2) the molds are removed from the metal slugs; (3) the slugs are then cleaned and inspected (with rejects going back into the alloying/casting step); (4) the slugs are inserted into cladding tubes (with metallic bond material, if appropriate); (5) the elements are sealed (with back-fill gas if appropriate); (6) the elements are inspected; and finally (7) the elements are combined into assemblies.

3.4 ATW T&B Performance and Representative Parameters

3.4.1 The Blanket

The consumption of transuranic actinides is accomplished in the reference concept using a subcritical blanket (nominal k-eff = 0.97) composed of hexagonal fuel assemblies similar to conventional liquid metal reactor (LMR) assemblies. The blanket, which has an annular shape, surrounds the central LBE target and source buffer/diffuser region. The accelerator beam impinges vertically on the LBE target from above to produce a source of energetic neutrons that initiate the truncated fission chains in the subcritical blanket. Transmutation of long lived fission products (LLFP) is achieved in transmutation assemblies placed laterally outside the blanket region. Materials that moderate the neutron spectrum can be incorporated in the LLFP transmutation assemblies to maximize the LLFP consumption rate.

Previous analyses of the ATW burner have assumed a unit power level of 2000MWt. Recent deliberations by the Scenarios and Systems Integration Working Group (SSIWG) have resulted in a new reference power level of 840MWt; the smaller size should facilitate adoption of modular construction techniques and may be more economical.

3.4.2 The Operational Cycle

The fertile-free fuel composition proposed for use in ATW maximizes the net destruction rate of transuranics per unit of thermal power, but leads to a high rate of reactivity loss with burnup. Unmitigated, this reactivity loss would result in a decline of the power output and a loss of electricity production revenue. The loss can be compensated through active reactivity control (e.g., control rods) or active source control (requiring an accelerator sized for the minimum system reactivity). Adoption of a relatively short duration for the irradiation interval (e.g., 100 d) reduces the reactivity loss and minimizes the cost penalties (and potential safety issues) associated with active control. To help flatten the power distribution, which tends to be peaked toward the center of the ATW blanket, the fresh fuel assemblies can be loaded into the outermost blanket positions, and shuffled inward as fuel is depleted with burnup accumulation. Discharge burnup (for fixed cycle duration and specific power) can be adjusted by varying the number of irradiation cycles. Analyses to date have assumed three-to-six 100-day irradiation intervals (and up to six shuffling zones in the blanket); with the six-cycle residence time, the equilibrium cycle discharge burnup is approximately 35%.

3.4.3 Mass Flows and Inventories

Preliminary estimates of TRU mass flows and inventories have been made for the 840 MWt LBE-cooled concept. For these analyses, the discharge burnup level is 45% for LWR TRU composition and 33% for the equilibrium-recycle composition. Basic characteristics and estimated mass flows for that system are as follows:

Power (MWt)	840
Irradiation Interval (d)	100
Cycle Duration (d)	122
Fuel residence time (cycles)	6
Average power density (W/cc)	350
Mass flows with LWR discharge feed (kg/y):	
Transuranic actinides charged	1158
Transuranic actinides discharged	642
Transuranic actinide consumption	516
Minor actinides charged	126
Minor actinides discharged	84
Minor actinide destruction	42
Discharge burnup with LWR feed (atom%)	44.6
Beginning of cycle transuranic inventory with LWR feed (kg)	1072
Equilibrium cycle mass flows (kg/y):	
Transuranic actinides from recycle	1311
Transuranic actinides from LWR	258
Total transuranic actinides charged	1569
Transuranic actinides discharged	1053
Transuranic actinide destruction	516
Minor actinides from recycle	231
Minor actinides from LWR	27
Total minor actinides charged	258
Minor actinides discharged	210
Minor actinide destruction	48

Discharge burnup with equilibrium feed (atom%)	33
Beginning of cycle transuranic inventory with equilibrium feed (kg)	1483

The option of simplifying fuel management by increasing cycle duration (and reducing the number of batches/irradiation cycles) does not appear to be an attractive option because it would increase the burnup reactivity loss proportionally. Current estimates of this reactivity loss based on the 100-d irradiation interval are 8%Δk for the LWR transuranic feed composition and 6%Δk for equilibrium-recycle feed composition.

Current estimates of the Tc-99 destruction rates are 15% per year in the fast spectrum of the ATW blanket and roughly 50% or greater in a moderated spectrum adjacent to the blanket. Specific target design concepts have not yet been developed to the extent required to estimate actual Tc-99 (or I-129) mass flows.

Table 3.3 provides a listing of key performance parameters for the conceptual ATW system considered thus far.

Table 3.3 Generic ATW Performance Parameters (indicative only)

Parameter	Assumed or input	Computed	Units
k-effective	0.97		
nu (neutrons/fission)	2.95		
Neutron importance factor	1.2		
Spallation threshold	0.2		GeV
n/p multiplier-lead	33.8		
Beam Energy Deposit fraction	0.7		
Burner Inlet Temperature	350		°C
Burner Outlet Temperature	500		°C
Thermodynamic efficiency	37.0		%
Operating fraction (capacity factor)	70.0		%
Atoms per kg of TRU	2.51×10^{24}		
Atoms per kg of Tc-99	6.08×10^{24}		
Atoms per kg of Iodine	4.71×10^{24}		
Tc conversion efficiency	0.7		
Iodine conversion efficiency	0.7		
Neutrons per proton		27.0	
Neutron multiplier		33.3	
Neutrons per second/target		7.60e19	
Fissions/second/target		2.50e19	
Fissions/year/target		5.52e26	
Neutron/fission for Tc		0.534	
Neutron/fission for Iodine		0.115	

Total neutrons/fission for Tc and I		0.649	
kg TRU fissioned/year/target		220.0	kg TRU/yr
kg TRU fissioned/year total		1760.0	
Fission heat		832.1	MWt
Target/blanket total heat		840.00	MWt
Electricity per target/blanket		310.8	MWe

4. ATW TARGET AND BLANKET (T&B) DEVELOPMENT PROGRAM

The deployment of a series of ATW plants by the mid-21st century will require a series of carefully planned steps whose aim is to address and resolve the unknowns and uncertainties existing between today's technology and that required to operate successfully the nth-of-a-kind plant. These steps include selection, development and demonstration of the base technology, and the design, construction and operation of systems which will become progressively closer to the first of a kind plant.

4.1 Base Technology Development (2001-15)

The adoption of proven core and thermal-hydraulics technologies (BRUS-300, ALMR) and the use of solid fuels, which are readily testable extensions of known technology, contribute to limit the extent of the required research and development for the ATW T&B system. While largely based on existing technologies and materials, the successful realization of the ATW prototype will require addressing specific issues in the following areas:

Fuel
Coolant Chemistry and Materials Compatibility
Heat Removal Systems
Nuclear Design and Safety
Spallation Targets

The base technology development phase will address technology transfer, critical components design, development of materials database, feasibility issues specifically related to implementation in subcritical systems and finalize important design choices. It is envisioned that this phase will be particularly emphasized in the first eight years, with final decisions on the fuel, target and coolant technology in 2008, but it will continue until 2015 when the ATW Demo becomes operational. Extensive collaboration is envisioned with other International groups working on similar technologies for ATW. The main T&B test programs in this base technology phase include:

- Test of spallation target concepts.
  The Target Test Facility (TTF) is proposed to test several target concepts for use in the ATW T&B system. Built at the site of a large power proton accelerator, the facility will perform materials, thermal-hydraulics and neutronics tests on the two basic target concepts (liquid LBE and sodium-cooled solid tungsten). At appropriate times in the testing of various target prototypes, fissile material will be introduced to provide data on the neutronic coupling of a spallation target and a subcritical assembly of fissile material.

- Irradiation tests of materials for window applications.
  Prototypical tests of exposure to high-energy protons will be performed using the Los Alamos LANSCE beam and the Paul Scherrer Institute (PSI) beam in Switzerland.

- Transfer of LBE technology from Russia.
  Successful development of an LBE-cooled ATW system within a 25-year period requires that the U.S. acquire Russian technology for LBE coolant. LBE appears to be at this point the best potential candidate for coolant and spallation target for ATW systems, but the least understood and developed in the Western World. A strong effort in technology

transfer is recommended. Many of the proposed tasks in this Target and Blanket roadmap include effort for technology transfer and U.S. laboratory experimentation, for the purpose of developing U.S. expertise with LBE coolant.

- Thermal-Hydraulic Test Loop (LT-1)
  This non-isothermal loop will be used to study Pb-Bi heat transfer characteristics and compatibility with structural materials and to assist in technology transfer from Russia. Large-scale hydrodynamic and thermal-hydraulic tests (including development and design of heat exchangers, steam generators, valves, pumps and other pieces of equipment for sodium systems) will be conducted in Russia (Podolsk and Obninsk) and in cooperation with the Laboratories of ENEA (Italy) and Karlsrhue (Germany) using their recently constructed large scale LBE facilities.

- Nuclear Data Tests (NT-1)
  Tests of validity of basic nuclear data specific to ATW (Pb, Bi, TRU data), conducted at existing facilities in cooperation with international partners.

- Development of fuel concepts
  Because there is essentially no data available on the properties or performance of the proposed fuel composition, key feasibility tests will be performed to establish the suitability of the reference fuel type and the selected alternatives.

Final decisions on the fuel, target and coolant technology is expected in 2008. Further development work on the selected technologies will continue until 2015

4.2 Demonstration and Large-Scale Integration (2006-25)

Starting in 2006, activities will commence towards the realization of an integrated ATW Demonstration Facility (ATW Demo). The facility will use full-size components but will start operation at 5% power (30 MW) in 2015. Power will be slowly increased, through increases in beam power and core reactivity, up to full power in ten years (2025). At low power levels, the facility, in addition to verifying system integration, will provide an important test bed for fuel experiments, materials studies, studies of system dynamic behaviour, abnormal conditions, and thermal-hydraulics of forced and natural convection. The ATW Demo will initially utilize proven fuel concepts, gradually converting to the final ATW T&B fuel system.

- Successful design and construction of the ATW Demo will require the development of large-scale heat removal components (e.g., pumps, heat exchangers, and steam generators). Integrated testing will be required for first-of-a-kind, large components prior to installation in the plant. Because facilities for integrated testing of large sodium components are no longer available in the U.S., a new Component Test Facility (CTF) is proposed as part of the Demonstration and Large-Scale Integration effort. The necessity for this facility may be alleviated if foreign facilities are both suitable and available to meet the need.

- Successful operation of theATW Demo at low power levels will have proved the feasibility of the ATW concept. Subsequent high power operation will: (a) demonstrate the integration of all key processes, technologies, and components of an operable ATW system, (b) confirm licensability of the integrated system, and (c) enable thorough testing of operational limits on materials and systems performance. The expected ATW

performance, regarding both TRU fission and fission product transmutation, will be quantified duringits operation, while licenseability and process integration issues will be addressed and resolved during the design and construction phase. The Demo is scheduled to become operational in 2015 and is expected to continue to operate through the mid-century. Its main role in later life will be to provide a test bed for advanced concepts.

- At full power in 2025,the ATW Demo becomes the ATW Prototype, and in addition to demonstrating the operability of ATW, it will be used to quantify ATW economic performance, establish the scaling to the nth of a kind and determine ATW infrastructure requirements. The prototype may become uneconomic to operate after more ATW T&Bs are constructed using improved technology and the nth-of-a-kind stage is reached, so it may be decommissioned by mid-century.

5. THE ATW T&B R&D ROADMAP: ISSUES AND TASKS

Consistent with the grouping of issues identified in Section 3, the development of roadmaps to address specific issues was organized into five separate categories:

Nuclear Design and Safety
LBE Chemistry and Materials Performance
Heat Removal and Ancillary Systems
Target Technology
Fuel Technology

For each of these categories, the specific issues associated with the reference technologies were identified by a selected group of individuals, primarily drawn from U.S National Laboratories, and a plan for addressing those issues and developing the technologies to be part of a licensable ATW system was developed. Where applicable and possible, alternatives to the reference technologies were identified, and plans for development of those alternatives were incorporated into the roadmaps. To provide support for the roadmap and concepts presented in Sections 2 through 4 of this document, descriptions of each of these Target and Blanket roadmaps are included here. For each category, the roadmaps are described as a combination of tasks designed to provide information for key decision points or for design and construction activities, as outlined in the roadmap report by the System Scenarios and Integration Working Group. Estimates of the personnel and financial resources required for the completion of each task, in accordance with the deployment-driven scenario assumed for this effort, are presented, as well as the envisioned schedule by which the tasks can be completed.

5.1 Nuclear Design and Safety

5.1.1 Background/Introduction

This portion of the ATW T&B roadmap describes ATW research and development needs in the areas of nuclear design and safety and outlines technical activities to address these needs. The task of developing this T&B roadmap has been divided among several working groups, each responsible for a particular technology area (e.g., technologies for accelerator, beam delivery, spallation target, coolant, fuel/cladding, etc.). In most of these technology areas, a preferred “reference” technology

option has been selected from among several alternatives as the primary focus of the development effort. A decision has not yet been made, however, regarding the coolant system (with lead-bismuth eutectic (LBE) and sodium under active consideration) and the spallation target (liquid LBE and solid tungsten cooled by sodium are prominent options). The current choices result from balancing system performance objectives (described elsewhere) and the objective of keeping development costs/risks acceptably low. Most of these choices are consistent with international directions in development of accelerator driven systems, although the ATW focus on rapid destruction of all transuranic elements (vs. more selective incineration of minor actinides) leads to important differences in fuel form and partition/recycle technology. These reference technology choices (in italics) and principal alternatives under consideration are summarized as follows:

Technology Area	Options
Accelerator:	*proton LINAC*, cyclotron
Spallation Target:	*liquid Pb-Bi Eutectic (LBE), tungsten*, mercury
Subcritical Core:	*Fast-spectrum system with solid fuel*, thermal-spectrum system fueled with molten salt
Coolant:	*LBE, sodium*, helium, molten salt
Fuel:	*Zr-TRU metallic dispersion*, Zr-TRU metal, nitride, molten salt, graphite coated pebbles
Cladding	*Russian alloy EP823*, domestic alloys (TBD) and core structure:
LLFP transmutation:	*Moderated ex-core Tc and I transmutation assemblies*, unmoderated transmutation assemblies (in-core or ex-core), Tc admixed with fuel
Primary System:	*Pool layout with intermediate loop…*

It should be apparent that these options are strongly inter-dependent owing to material properties (e.g., the fast neutron spectrum is enabled by use of a liquid metal coolant) and material compatibility considerations (e.g., compatibility of coolant and cladding).

The nuclear design and safety R&D activities have been planned under the assumptions that the R&D required to demonstrate performance capabilities and limits of the various technologies under consideration (e.g., temperature capability of the fuel/cladding system) will be addressed by the working groups responsible for these technology areas. The nuclear design and safety research recommended here plays a central, integrating role in defining appropriate goals for these limits, and in exploiting the demonstrated capabilities and accommodating the demonstrated limits in design of the ATW core and confirming system safety.

5.1.2 Nuclear Design and Safety R&D Issues

The Nuclear Design and Safety task addresses the need to design the ATW core such that TRU and LLFP incineration is accomplished as economically as possible consistent with technology constraints and safety requirements. This task determines core design features, quantifies performance characteristics, confirms system nuclear safety, and supports licensing. Specific issues to be addressed/resolved by the Nuclear Design and Safety R&D include:

- Degree of Subcriticality: The choice of the ATW system subcriticality margin ($1 - k_{eff}$) has to be carefully optimized by balancing the objectives of minimizing accelerator power requirement (favors high source multiplication through small subcriticality

margin) and precluding criticality or excessive power increases in response to operational or accidental reactivity additions of uncertain magnitude (favors large subcriticality margin).

- Burnup Reactivity Loss: The fertile free fuel composition proposed for use in ATW maximizes the net destruction rate of transuranics per unit of thermal power, but leads to a high rate of reactivity loss with burnup. Unmitigated, this reactivity loss results in a decline of the power output and a loss of electricity production revenue. The loss can be compensated, at a cost, through active reactivity control (e.g., control rods) or active source control (requiring an accelerator sized for the minimum system reactivity). This motivates the search for core design and fuel management options which minimize the burnup reactivity loss and the development of optimized control strategies which compensate for the reactivity loss safely and cost-effectively.

- Feed composition variation with recycle: The mix of transuranic isotopes in the ATW fuel evolves gradually with repeated recycle from the initial mix of LWR spent fuel (which is itself variable) to an equilibrium composition which additionally depends on the ATW neutron energy spectrum, the timing of fuel cycle operations, and recovery fractions in recycle and refabrication. Fuel assembly and core design options, fuel management strategies, and control schemes that effectively accommodate such variations in fuel composition and associated variations in reactivity worth must be devised.

- LLFP incineration: The fast spectrum of the ATW blanket is ideal for incineration of transuranics because it avoids the buildup of the higher minor actinides to levels that complicate fuel recycle and waste form production/disposal. Moreover, the fast spectrum provides for an excess of neutrons that can be used to transmute long-lived fission products (namely, Tc-99 and I-129) whose geologic disposal is of particular concern based on assessments of repository long-term radiological risk. It is important to design the ATW core and the incineration targets such that the LLFP transmutation efficiency is maximized (e.g., through moderation of the spectrum in which LLFP are irradiated) and the associated demands on the chemical processing steps are minimized.

- Dynamic behavior: The neutron kinetic behavior of source driven systems differs in fundamental ways from that of critical systems. Source transients can be initiated by changes in accelerator beam characteristics or by changes in the geometry, temperature, or composition of the spallation target. System neutronic response to both source transients and reactivity transients is governed by prompt neutrons, and -- depending on the subcriticality level and the magnitude of the perturbation -- both the magnitude and the spatial shape of the fission power may be very sensitive to such changes. Moreover, the fertile free fuel composition in ATW leads to a low delayed neutron fraction and Doppler coefficient. A detailed understanding of the space-time dynamic behavior of ATW during operational transients and potential accident sequences is needed to confirm operability and safety and to formulate requirements on system monitoring, control and safety protection systems.

- Safety confirmation: The subcriticality of accelerator driven systems is believed to offer an important safety advantage over critical systems in that the former are able to accommodate unprotected reactivity insertion accidents when fertile-free fuels (with

low Doppler feedback and small delayed neutron fraction) are employed. On the other hand, accidental increases of the source are possible in accelerator driven systems, and thus the ability of ATW system to safely accommodate various source transients must be demonstrated. In addition, the weaker sensitivity of accelerator driven systems to reactivity feedback effects is a disadvantage in loss of flow or loss of heat sink accidents, since reactivity feedback mechanisms may be ineffective in reducing power, making the shutdown of the neutron source essential to preventing system damage under such accident conditions. The potential for fuel melting due to under-cooling and for subsequent accumulation of fuel into a critical mass is particularly serious in view of the low Doppler coefficient, small delayed neutron fraction, and (for LBE coolant) the high inertial resistance of the heavy liquid metal to fuel dispersal. A similar safety concern arises from the potential of attaining supercriticality through seismically induced compaction of the core. These safety issues, as well as safety issues unique to the use of LBE coolant and to the containment aspects of coupling of an accelerator and a subcritical reactor must be addressed and resolved.

- Validity of simulation tools: The design of the ATW core and confirmation of system safety will rely extensively on simulations of neutronic, thermal-hydraulic and thermo-structural phenomena in normal and off-normal system states. Many of the required simulation tools (computer codes, modeling approaches, and databases) have been developed in the past for analysis of other types of nuclear systems. Additional work is needed, however, to extend and validate these simulation tools for application to ATW. The tools must accommodate specific ATW features such as the subcritical (source driven) system state, the coupling of the neutron spallation process and the core neutronic behavior, and the specific choices of materials (e.g., LBE coolant and spallation target, fertile free fuel, Russian structural alloys, etc.).

5.1.3 Nuclear Design and Safety Research and Development

The R&D program developed by the Nuclear Design and Safety subgroup is described in this section. This program (tasks, schedule, budget) was developed to address key issues related to ATW core design, safety confirmation and licensing. The program was structured to (a) exploit developments planned by other working groups and opportunities for collaboration with international groups working on ADS technology, and (b) ensure that key core design and safety parameters impacting other ATW tasks are provided to the responsible groups at the appropriate times.

5.1.3.1 Assumptions

The following assumptions were made in developing the nuclear design and safety R&D plan:

- Adopt reference ATW technologies and implementation scenario -- but not specific parameter choices (except as initial estimates); both sodium and LBE coolant will be maintained as options prior to down-selection. Because of greater performance and safety uncertainties associated with the use of LBE, this option will be the focus of much of the early development.

- Develop R&D plan assuming successful development and demonstration of the reference technologies; backup options are identified and cognizance of their development status internationally and their potential application to ATW is maintained.

- A phased development approach is adopted in which distinct, but overlapping, R&D stages are defined. The first stage emphasizes trade-off studies and base technology development. This stage is followed by nuclear design and safety R&D to support construction and operation of a demonstration facility "DEMO" and, subsequently, of a prototype ATW plant "PROTO" producing 310 MW of electric energy.

- Operation of the DEMO plant will evolve from initial operation at low power (30 MWt) using proven LMR fuel (e.g., U-10%Zr) to higher power (up to 420 MWt) and fertile-free ATW fuel; this ATW fuel would initially be produced using LWR discharge transuranics and later would also incorporate transuranics derived from recycle of the fuel discharged from DEMO.

- Critical experiments (in cooperation w/France, Japan and/or Russia) may be required to validate nuclear data for selected important nuclides.

- Experimental data needs required to confirm LBE, fuel, target and window, FP transmutation assemblies, plant components, etc., are addressed by other working groups and technology development tasks; this includes severe-transient fuel behavior tests (e.g.; transient tests of fuel melting and relocation) which may be required.

- Capabilities needed to model the spallation process and products are addressed primarily in the Target Technology portion of this Target and Blanket roadmap.

5.1.3.2 Overall Schedule and Estimated Budgets

Phase 1 (years 1 to 3, ~$3.5-4.0M per year)

-- Perform scoping calculations to characterize basic performance and safety characteristics of full-size (840 MWt) ATW system using reference technologies and initial assumptions. Identify impacts associated with adoption of LBE vs. sodium as coolant and LBE vs. tungsten as spallation target.

-- Perform trade studies and assessments to quantify safety and performance trade-offs associated with variations in core geometry, TRU and LLFP loading, fuel management scheme, control strategy, etc. Studies build on previous work as much as possible. Key results include minimum subcriticality level, fuel assembly design, core and reflector/shield concepts, control and shutdown strategy, LLFP transmutation assembly concepts and irradiation strategy.

-- Assess the adequacy of data (nuclear data, thermophysical properties, etc.), analysis models/methods, and computer codes for neutronic, thermal-hydraulic, and thermal-structural analysis; perform benchmark studies and sensitivity analyses

-- Initiate enhancements of simulation tools; emphasize those enhancements required for DEMO design and safety confirmation.

-- Initiate critical experiments program to validate nuclear data and analysis methods.

-- Initiate DEMO design and safety assessments; develop system requirements; identify safety issues and define design-basis and beyond-design-basis accident sequences.

Phase 2 (years 4 to 10, ~$5.0 per year)

- -- Update concept studies, fuel cycle studies, and control/safety studies for ATW and DEMO (focus on DEMO); utilize latest information/constraints developed in other tasks (e.g., fuel and target development tasks).

- -- Execute critical experiments program and apply results to reduce uncertainties in design predictions

- -- Perform selected cross section measurements at existing facilities based on needs demonstrated in phase 1.

- -- Continue enhancements of methods and codes for application to DEMO design and safety analyses; validate codes and data to support DEMO design, licensing and initial operation; perform code quality assurance.

- -- Perform DEMO conceptual design emphasizing initial 30 MWt configuration and use of conventional LMR fuel (U-10%Zr). Investigate and select concepts for system control, transition from conventional to fertile free ATW fuel, LLFP incineration, etc.

- -- Initiate DEMO safety studies to support preparation and review of the PSAR; identify safety/licensing issues; define design basis and beyond-design basis accidents (accident delineation and sequence identification); analyze accident sequences to characterize system safety behavior; prepare PSAR.

Phase 3 (years 11 to 20, ~$6.0M per year)

- -- Perform detailed nuclear design for DEMO; specify assembly and core design (including source diffuser), fuel management scheme, reactivity control scheme, reflector/shield design etc. Develop design and operating strategies to accommodate (a) variable power output (30 MWt to 420 MWt); (b) variations of fuel loading ranging from standard LMR fuel (e.g., U-10Zr) to fertile-free ATW fuel (using LWR discharge transuranics and, once available from the pyrochemical separation facility, self generated transuranics); (c) “heterogeneous” configurations in which the ATW fuel is irradiated as experimental assemblies; and (d) moderated Tc and I transmutation assemblies.

- -- Perform DEMO safety studies to support safety reviews and licensing; identify and resolve safety/licensing issues through analyses and use of experimental data (maximize use of past experience in resolving reactor safety issues for liquid-metal cooled reactors); complete definition of design basis and beyond-design basis accidents; analyze accident sequences to confirm system safety and support preparation of the DEMO FSAR.

- -- Analyze DEMO startup tests and measured operational behavior.

- -- Perform DEMO core-follow analysis; assess codes and data against DEMO facility operations (for “homogeneous” and “heterogeneous” configurations); use DEMO data

to help validate codes for application to design and safety assessment of the prototype facility "PROTO".

-- Validate predictions of nuclide transmutations against analytical chemistry and NDA measurements of irradiated-fuel isotopics (performed in the pyrochemical pilot plant)

-- Initiate PROTO nuclear design; specify assembly and core design, fuel management scheme, reactivity control scheme, reflector/shield design, etc.

-- Initiate PROTO safety assessments; identify safety/licensing issues, resolve safety issues through analyses and use of experimental data and DEMO operational information; define design basis and beyond-design basis accidents (accident delineation and sequence identification); analyze accident sequences to confirm system safety.

-- Support PROTO safety reviews and licensing interactions

5.1.3.3 Task Descriptions

The nuclear design and safety R&D program is divided into the following four elements:

A. Establish ATW Capabilities and Design Goals
B. Develop and Validate Simulation Tools
C. DEMO Design and Safety Confirmation
D. DEMO Operational Analysis
E. PROTO Design and Analysis

Descriptions of each R&D element and associated tasks are provided below. The performance schedule and effort allocation for each task are shown in the accompanying spreadsheet.

A. Establish ATW Capabilities and Design Goals

This program element provides for execution of analyses to guide the selection of ATW (840 MWt system) design features and to characterize the neutronic, thermal-hydraulic, thermo-structural, and safety performance of the system. The analyses will concentrate on the reference ATW technologies [e.g., liquid metal (Na or LBE) coolant, metallic fuel, and a subcritical arrangement of assemblies around the target and source-diffuser region. Trade studies will be performed to address the performance and safety implications of variations in choice of coolant, target concept, core and assembly geometry, TRU and LLFP loading, fuel management scheme, control strategy, etc. Safety considerations will be incorporated early on as an integral part of the design process. To the extent possible, these studies build on previous work on ADS at LANL and elsewhere, as well as on the international experience with the design and operation of liquid metal reactors (LMRs). Studies of principal backup technology options (e.g., gas coolant) may also be conducted on a limited basis.

The tasks associated with this program element are:

-- *Neutronics and fuel cycle evaluations*: Execute nuclear design concept trade studies and fuel cycle calculations accounting for the evolution of the TRU feed isotopics with multiple recycle of ATW discharge fuel. Compute depletion dependent performance

characteristics (e.g., reactivity loss rate, power distribution, TRU and LLFP incineration rates, reactivity feedback coefficients, material worths, etc.) as a function of the major design and operational variables.

-- *Core thermal-hydraulic analysis*: Compute detailed temperature and flow rate distributions within the core, accounting for irradiation effects, cladding oxidation, coolant chemistry effects, etc. Investigate intra- and inter-assembly flow re-distribution during transients. Evaluate design options (e.g., pin lattice and assembly geometry, pin spacer concepts, coolant flow allocation strategies) with the goal of optimizing thermal performance for a given power deposition profile.

-- *Core thermo-structural analysis*: Investigate the thermo-structural response of the core assemblies to thermal transients -- as a function of assembly irradiation history. This response gives rise to reactivity feedback which are important in operational transients and accident sequences. The thermo-structural feedback associated with system cool-down is a component of the power defect and needs to be properly characterized in setting the subcriticality margin.

-- *Dynamic behavior and control strategies*: The neutron kinetic behavior of source driven systems differs in fundamental ways from that of critical systems. Moreover, the fertile free fuel composition in ATW leads to a low delayed neutron fraction and Doppler coefficient and a high reactivity loss rate (and associated reduction of source multiplication) with burnup. Different strategies for coping with this reactivity loss (e.g., increase of accelerator beam current, insertion of reactivity via control rod withdrawal, ...) have been proposed. This task is concerned with characterizing system dynamic behavior and developing optimum control strategies for accomplishing power maneuvers and burnup reactivity loss compensation.

-- *Safety assessments*: Evaluate integrated system response during operational transients and postulated accident sequences. Verify safe shut-down and decay heat removal for a range of generic sequences that potentially challenge system integrity and public safety. Emphasize safety issues unique to accelerator driven systems (e.g., source transients) and to the reference fuel and coolant technology options.

Key results for Task 1 include target values or ranges for minimum subcriticality level, fuel assembly design parameters, core and reflector/shield concepts, fuel management scheme, control and shutdown strategy, LLFP transmutation assembly concepts and irradiation strategy. Key trade-offs involved in adoption of different choices (including the choice of coolant material) will be clarified. This information will help establish the capabilities of ATW the requirements for its design and is needed to guide efforts on design of the DEMO and PROTO facilities.

B. Develop and Validate Simulation Tools

This R&D program element provides for assessment, adaptation, development, and validation of the simulation capabilities (data, methods/models and computer codes) required for ATW core design, and for evaluation of core performance and system safety characteristics. These simulation tools are also required to support the development and operation of facilities needed to develop and demonstrate ATW technologies. The effort related to computer codes is

maintained over the entire duration of the ATW development program, although the schedule concentrates most of the effort within the periods preceding the startup of the DEMO facility. A lower level of effort during the subsequent period is devoted to maintaining the codes and associated databases and to implementing upgrades as required.

The tasks associated with this program element are:

-- *Nuclear Data*: The goal of this task is to ensure the availability and quality of the basic nuclear data required for analysis of ATW systems; the scope of this task does not include spallation data, which are assumed to be addressed by the target technology development task. The focus initially is on assessment of priorities, including sensitivity and uncertainty analyses, and on evaluations of key data which may not have received sufficient attention in the past (e.g., cross sections for lead and bismuth). Participation in an international critical experiments program (details to be defined) is planned to benchmark the available data and improve their validation status. Provisions are also made for new data measurements (at existing facilities) should the need for such measurements be demonstrated in the assessment activity; candidate data for new measurements include minor actinide fission cross sections and neutron yields, and capture cross sections for Bi, Tc-99, and minor actinides. Finally, the use of integral measurements from critical experiments to reduce uncertainties in predicted ATW performance parameters, through use of formal data adjustment procedures, is planned subsequent to completion of the experimental measurements program.

-- *Neutronics and fuel cycle codes:* This task addresses the adaptation, development, testing, and validation of computer codes used to analyze the neutronic behavior and the fuel cycle performance of ATW systems. The need to develop new codes is minimized through extension and use of codes previously developed, e.g., the LANL MCNP code and the ANL code system for LMR design analysis. The planned effort encompasses both the MCNPX Monte Carlo code and deterministic codes for static and dynamic applications. Extensions required to model accelerator driven systems and potential ATW features (e.g., moderated LLFP transmutation assemblies) will be implemented and the resulting codes will be validated for use in ATW design and safety confirmation.

-- *Core thermal-hydraulics (subchannel T-H analysis):* This task provides for development/adaptation and testing of a code for use in thermal-hydraulic design analysis of the ATW core. This code will be used to predict coolant, structure and fuel temperatures in each core subchannel, consistent with the coolant flow rates and power deposition distribution, and to optimize coolant flow allocations and design of assembly flow orifices. Thermo-physical properties and heat transfer correlations appropriate to the LBE coolant option will be implemented.

-- *Assembly Thermal-Hydraulic Tests:* Compile and validate thermal-hydraulic data (e.g., pin-bundle heat transfer coefficient correlations for LBE) for use in core thermal-hydraulic design and analysis. This task would be carried out in collaboration with Russian organizations (IPPE, Gidropress) and would make use of existing measurements and experimental facilities to the greatest extent possible.

-- *Structural analysis (thermal and radiation effects):* Computing capabilities will be developed for predicting displacements and associated stresses of core components consistent with thermal profiles and irradiation history, and for estimating the thermo-structural response of the core to variations in power and flow. The ability to predict this thermo-structural response is needed to characterize reactivity feedbacks during operational transients and accident sequences.

-- *Systems/safety code:* This task provides for adaptation of existing capabilities (previously developed for LMR analyses) for integrated ATW system-level modeling of neutron kinetics, heat transfer, fluid dynamics, and structural mechanics of fuel elements, coolant channels and core structures, and for thermal-hydraulics in ex-core coolant piping, plena, and components. An key element of this task will be to implement features required to model heavy liquid metal (LBE) coolant, including a structure corrosion model, LBE systems phenomenological models, multidimensional fluid dynamics effects in the reactor and the primary coolant system, including corrosion product transport and deposition. Models describing the effect of accelerator beam and target transients on the neutron source will also be implemented. Balance of plant thermal hydraulic and component models will be provided to enable evaluation of system response to transients initiated in the balance of plant. Control systems will be modeled for their impacts on transient behavior and safety performance.

C. DEMO Design and Safety Confirmation

This R&D program element addresses the design of the DEMO core and the resolution of nuclear safety and licensing issues. The planned effort encompasses nuclear, thermal-hydraulic, and thermo-structural design of the DEMO core; safety confirmation and licensing support; analysis of DEMO operation and experiments subsequent to facility start-up; and analysis support of DEMO conversion from conventional LMR fuel to fertile-free ATW fuel.

The tasks associated with this program element are:

-- *Design requirements*: Participate, with other working groups, in defining DEMO performance requirements and specifications. Carry out design and safety trade studies as required to help guide the selection of such key DEMO parameters as degree of subcriticality, power level, power density, core inlet and outlet temperatures, choice of materials for core assemblies and components, core geometry flexibility to accommodate ATW experiments, control and shutdown system concepts, operating cycle length, etc.

-- *Nuclear design*: Perform nuclear and fuel cycle analyses with the objective of optimizing DEMO core layout and assembly design characteristics. Key design variables include core size and shape, fuel enrichment, arrangement of fuel and experimental assemblies, lattice geometry and material loadings for the different types of assemblies, source diffuser geometry and position relative to the core, etc. Optimization goals include: low burnup reactivity loss rate, uniform power distribution, maximum operating flexibility, and minimal likelihood and consequences of accident initiators. The required design analyses will be executed using both Monte Carlo (MCNP-X) and deterministic codes originally developed for LMR analysis. In addition to developing the core design and predicting nuclear performance parameters, this task will provide information essential to other R&D tasks, for example reactivity coefficients and kinetics parameters for use in dynamic studies and safety assessments. The nuclear design effort will progress from conceptual to final design stages and will support the development of both “homogeneous” and “heterogeneous” variants of the DEMO, and the transition of DEMO to ATW fuel.

-- *Dynamics and control*: Analyze the dynamic behavior of DEMO, particularly its response to accelerator beam transients and reactivity perturbations. Assess and formulate

monitoring, control and safety protection requirements. Develop optimum control strategies for accomplishing power maneuvers and burnup reactivity loss compensation.

-- *Subchannel thermal-hydraulics and flow orifice design*: Carry out thermal-hydraulic design of DEMO. Evaluate and select key thermal-hydraulic design parameters (e.g., pin lattice and assembly geometry, pin spacer concepts, coolant flow allocation strategies) with the goal of optimizing thermal performance for a given power deposition profile. Compute detailed temperature and flow rate distributions within the DEMO core, accounting for irradiation effects, cladding oxidation, coolant chemistry effects, etc. Characterize intra- and inter-assembly flow re-distribution during transients.

-- *Assembly mechanical design and thermo-structural analysis*: Develop detailed assembly designs for the different types of assemblies used in DEMO, accommodating irradiation goals, thermal performance requirements, and coolant-structure compatibility constraints. Characterize the thermo-structural response (displacements of core components and associated stresses) of DEMO core components to thermal and irradiation effects during normal operations and off-normal sequences.

-- *Safety studies and licensing support*: Identify nuclear safety issues impacting the design and operation of DEMO. Specify design-basis and beyond-design-basis accidents. Perform integrated safety analyses to characterize DEMO behavior during operational transients and accident sequences, accounting for all important neutronic, thermal-hydraulic and structural phenomena and for the mutual coupling among these phenomena. Evaluate accident consequences and confirm the effectiveness of measures to mitigate these consequences. Utilize the results of these assessment to enhance safety through adjustment of design features. Prepare safety documentation, interact with licensing authorities, and resolve safety issues raised in the licensing process.

D. DEMO Operational Analysis

This program element supports the operation of the DEMO facility and its transition to ATW fuel (including self-recycled fuel); it also provides for improvement and extension of analysis methods and data based on experience gained in the operation of DEMO. The tasks associated with this program element are:

-- *Analysis of start-up tests*: Participate in the specification of start-up physics tests and perform simulations of these tests and of initial DEMO operations. Compare simulation results with measured test data. Assess the sources and causes of disagreements between measurements and calculations, and modify simulation tools as warranted. Use measured results to improve the validation status of the computer codes.

-- *Operations and core follow analysis*: Perform nuclear/fuel cycle, thermal-hydraulic, and safety evaluations as required to support DEMO operations and technology demonstration experiments.

-- *Evolution of discharged/recycled fuel and LLFP transmutation assembly isotopics*: Qualify predictions of nuclide transmutations against analytical chemistry and NDA measurements of irradiated-fuel and transmutation assembly isotopics (performed in the

pyrochemical pilot plant). Quantify uncertainties in discharge assembly compositions. Assess impacts of uncertainties in recycled fuel composition on performance and safety.

E. PROTO Design and Analysis

This program element addresses the nuclear design and safety analysis needs associated with the development of a prototype ATW plant (PROTO), building on the ATW R&D efforts and the experience gained through DEMO design, operation and experiments. The tasks related to design of the PROTO facility closely parallel the foregoing tasks for DEMO, although the addition of the balance of plant systems for electricity generation must be accommodated in the design analysis and particularly in the safety appraisals. Because the tasks specifically related to this program element are scheduled more than twenty years into the future, detailed task descriptions are not provided at this time.
The tasks associated with this program element are:

- *Pre-conceptual design and design requirements*
- *Nuclear design*
- *Dynamics and control*
- *Thermal hydraulics and structural design*
- *Subchannel T-H analysis and coolant orifice design*
- *Fuel assembly mechanical design and thermo-structural analysis*
- *Safety studies and licensing support*
- *Analysis of start-up tests*
- *Operations and core follow analysis*

5.1.4 Estimated Cost and Schedule

Schedules and resources required for the Nuclear Safety and Design tasks described above were estimated for the purposes of estimating the sequenced cost of developing a sodium-cooled and LBE-cooled ATW target and blanket system. The projected cost and schedule information developed in this effort, for the deployment-driven scenario assumed, are summarized in the tables contained in Attachments 1 and 2. Although year-by-year allocation is not shown in the tables, a linear distribution of the total effort and costs reflected in the tables was assumed (for simplicity); those assumptions provide the basis for the year-by-year summary of costs presented in Section 6. An average, fully-burdened effort rate of $220k per full-time equivalent year was assumed.

5.2 Coolant Chemistry and Materials Compatibility

5.2.1 Introduction

The technical issues addressed by this portion of the Target and Blanket roadmap report include: LBE conditioning, cover gas control, structural materials corrosion in LBE, structural materials performance, heat removal, and waste management. Because the technology for maintaining sodium coolant chemistry and for ensuring compatibility of structural materials with sodium is well developed and mature, only LBE was considered in this effort; it was determined that those issues for a sodium-cooled ATW reactor need not be the subject of a focused R&D plan.

Because LBE coolant technology is critical to the success of the proposed LBE-cooled ATW system, it was determined that many of the LBE chemistry and materials compatibility R&D tasks are required in the first phase (5-8 years) of the ATW program. The focus of this plan is on the

initial R&D needs with the specific subsystem design, system integration and performance optimization of later phases defined in less detail.

For this roadmap, the research topics were partitioned into four major tasks:

-LBE Coolant System
-Structural Materials
-Spallation Products
-Coolant Technology Safety System

The first major task is to develop the ability to operate a LBE coolant system by applying and verifying the existing Russian LBE coolant technology. The emphasis is on adapting the application of the coolant technology to configurations and conditions that are prototypical for the proposed ATW system, including the interaction with the T/B system design effort to ensure the successful maintenance of the coolant quality. The decommissioning and decontamination of LBE coolant will be considered as well. The important tasks to be accomplished include: replicating the ability to run LBE loops under steady-state conditions, establishing an acceptable operating condition range, establishing the ability to recover from abnormal conditions, developing flow regimes for coolant quality control, and developing an operational LBE coolant waste management system.

The second major task is to select adequate structural materials, which are an integral part of the coolant technology. Structural materials must be found that maintain structural performance in neutron and neutron plus proton radiation fields and are corrosion resistant to LBE coolant. A choice must be made between Russian or US design structural materials. Russian ferritic/martensitic steels are known to be corrosion resistant and able to maintain adequate mechanical performance in LBE cooled reactors. Russian austenitic steels are known to perform in LBE cooled reactors, but radiation damage data is limited to low dose so additional irradiation data will be required. Russian steels may provide cost and time savings, but the quality assurance (QA) efforts to validate design and performance data for Russian steels might be extensive. Additionally, production capacity for producing some Russian steels is not currently available. If the existing Russian data is not sufficient, tests may be needed to establish a database suitable for design needs. Some promising US steels have extensive radiation experience but have not been proven corrosion resistant in a LBE system. If modifications are made to replicate Russian steels, substantial corrosion and radiation tests will be required. For any choice of structural materials, there is scant data on materials' properties under the combined effect of protons and spallation neutrons, so some irradiation tests, within a coolant-quality controlled environment are definitely needed. The important tasks to be accomplished include: determining the corrosion resistance of U.S. steels to choose the steels to be used, establishing adequate mechanical property and corrosion resistance data, and ensuring the steels chosen are adequate in the unique environment of proton plus spallation neutron damage.

The third major task deals with the effects of spallation products, both on the chemical effects on coolant quality and corrosion, and the effects on system safety. This aspect is new to the Russian coolant technology. However, due to the existence of a self-healing protective oxide film on the structural materials in an actively controlled coolant environment for corrosion resistance, the very dilute concentrations of spallation products likely will not alter the coolant technology significantly. . The important subtasks to be accomplished include: calculating the expected amount of spallation products, determining the expected products from interactions between the spallation

products and the LBE coolant, determining the problems associated with the introduction of these spallation products by conducting surrogate materials experiments, and determining safety and cleanup needs.

The fourth major task singles out coolant system-wide safety issues for investigation. Ultimately, it is the convergence of the technical determination of safety and the regulation compliance that is needed for design and operation of larger test and demonstration facilities (e.g., the Target Test Facility and the ATW Demonstration Plant). The important subtasks to be accomplished include: determining potential hazards from the system from coolant leaks, cover gas leaks, and oxygen ingress, establishing design basis accidents, developing emergency response and cleanup procedures, and containment design.

After the R&D phase, the expected work on LBE coolant technology that continues through operation of test and demonstration facilities will be in the areas of integrated system tests, continued improvement of system performance and operability, material surveillance and certification, and design data needs-driven activities.

The following sections list the tasks, approach and approximate duration and costs. (Some parts of the program can be undoubtedly enhanced substantially by inputs from the IPPE and Gidropress specialists who know better the status and the required development work).

5.2.2 LBE Coolant System

Russian success with LBE provides assurance that LBE technology can be successful for ATW application in the U. S., therefore only reference technologies are identified for LBE.1.

LBE Coolant System tasks should finish before the coolant selection point in 2008 and to provide input for design of the ATW Demo Facility.

Duration: 12 years
Total Cost: $ 13.5M

1) Establish Normal Operating Conditions. This task is designed to establish the capability to operate a LBE system under normal (non-accident) conditions. LBE.1.1 should finish before the start of the Demo facility design. The baseline tasks should finish early to provide data needs for Demo facility licensing. Small portions could go on until the design completion.

a) Evaluate existing Russian coolant technology by establishing a QA program to certify the validity and suitability of the Russian data. Items to be evaluated include: oxygen control for corrosion prevention, coolant quality maintenance, structural material compatibility with LBE, polonium hazards and mitigation, cover gas system, and LBE safety criteria.

b) Corrosion and coolant quality control experiments. This experimental testing task will include: corrosion tests of select Russian and US steels, kinetics studies of oxygen replenishment for healing protective oxide films, and tests to understand the kinetics of slag cleanup. Initial corrosion tests, oxygen replenishment studies, and slag cleanup

studies can be performed in existing or modified LBE loops at IPPE or Gidropress. As part of this activity new US test facilities will be built to duplicate and verify key coolant technology features. Two LBE loops are anticipated, one for materials and thermal hydraulics studies and a second, smaller loop for spallation products testing. The second loop can be delayed if spallation product effects are found not significant from calculations.

c) Monitoring and surveillance programs. After the base technology is transferred and certified for use in the test facility, there will be continued efforts to monitor the LBE facility's structure integrity, corrosion and material degradation. This task is to design the procedures and tests that will be used in the test facility for monitoring and surveillance.

2) Establish Acceptable Operating Condition Range. After a baseline operating condition is established, a series of tests that determine the operating range is necessary. This will require study of the effects of variation of operating conditions of the test loops, and the postulation of design basis transients for testing.

3) Recovery from Abnormal Conditions. Once the safe operating window is established, procedures for recovering from abnormal conditions need to be established. Recovery procedures are needed to recover normal coolant chemistry and recover protective oxide films.

a) Determine consequences of deviation from normal conditions. Due to the relatively slow kinetics of corrosion and coolant contamination under normal operating conditions, short periods of abnormal conditions will not necessarily lead to facility shutdown or failure. This task will determine those deviations that will lead to recoverable consequences.

b) Determine response time and recovery and restoration procedures. LBE technology has the ability to recover from abnormal conditions and restore previously fouled (not corroded but had slag formation) components. Experimental tests are required to establish the operation parameters requires to re-establish desired conditions.

c) Determine limits for coolant system failure. when the deviation from normal conditions persists and coolant condition cannot be restored, the system has to be shut down for repair.

4) LBE Loop Test Facility. All the activities required to design, operate, and experiment with the LBE loop test facility are included in this task. Specific activities include guiding the actual implementation of coolant system in test facility construction, system operation, monitoring and evaluation of system performance in the test facility, and formulation of design input for subsequent test and demonstration facilities.

5) LBE Coolant System Operational Waste Management. Chemistry control systems will be needed for the cover gas and the LBE coolant system. Three waste streams are expected from the system: volatile species from the cover gas system, solid products removed from filtering systems, and activated products removed as liquids. Systems will need to be designed to remove these wastes from the system and to package and dispose of these wastes in an

appropriate manner. Studies will also be performed that determine the form and stability of the waste forms in changing from reactor operating temperature to storage temperature.

6) Thermal-hydraulic Requirements for Coolant Chemistry Control. Efforts to develop and design heat removal equipment are described in the Heat Removal and Ancillary Systems portion of this roadmap report. However, effort is included here to ensure a proper understanding of the relationship between coolant flow and corrosion resistance is attained.

a) Developing flow regimes for coolant quality control. The Russians have indicated that the ability to maintain corrosion resistant films and cleaning up coolant system contamination depends on maintaining minimum flow throughout the entire system. This task is to determine minimum flows and to work on system designs that ensure minimal flows.

b) Monitoring and testing of coolant quality and control. This task is designed to set up long term monitoring programs for the test facilities to improve coolant flow distribution parameters.

7) LBE Coolant Decommission and Decontamination. When test loops, the test facility, and the DEMO plant are eventually decommissioned, procedures for D & D of the LBE will be required. This task is designed to perform initial studies on proper D & D.

5.2.3 Structural Materials

The research and development needs for the target window, fuel cladding, and core structural materials are driven by two factors: compatibility with lead bismuth coolant and mechanical stability in an irradiation environment. Russian experience and success in lead bismuth reactor coolant systems will be heavily relied upon in designing the research and development program. Acceptance of structural materials for use in the ATW system will depend on acceptable corrosion resistance, acceptable mechanical integrity in an irradiation field, and the ability to verify and document the tests that support the materials properties.

Core Materials

Because the ATW system is expected to operate at temperatures and pressures comparable to those experienced in the Russian LBE-cooled reactors, the reference materials will be the proven Russian materials. For these steels, the compatibility with lead bismuth coolant will be verified, irradiation performance will be verified where significant Russian data exists, and radiation performance will be tested where Russian data does not exist. Initially, U.S. steels with extensive irradiation data from U.S. fast reactor programs will be subjected to scoping-type LBE corrosion test to determine whether those materials warrant further consideration.

Target Window Materials

The target window will be susceptible to intense radiation damage from both the proton beam and the spallation neutrons. This radiation environment is unique to the ATW system. Radiation damage, along with large amounts of hydrogen and helium produced, are expected to embrittle the

target window material. The large flux of high-energy neutrons and protons may also have detrimental effects on the corrosion resistance of the target window material. The corrosion and mechanical integrity under radiation performance will be determined through efforts desribed in the Target Technology portion of this roadmap. The tasks listed below regarding the spallation target are designed to ensure acceptable performance in lead bismuth coolant. Acceptable materials will then be passed to the spallation target working subgroup for radiation testing.

Materials Selection

ATW structural materials must be compatible with lead-bismuth cooling and maintain acceptable mechanical performance at temperatures up to 600°C. Based on Russian experience, it is proposed that structural materials used at temperatures greater than 450°C (including fuel cladding) will be fabricated out of ferritic/martensitic steels while those used at temperatures below 450°C will be fabricated from austenitic steels. To reduce the initial cost and time requirement, it is proposed that LBE-compatible Russian steels be used for test and demonstration facilities. The Russian ferritic/martensitic steel EP-823 will be the reference material for cladding and other high temperature (greater than 450°C) components. A high-silicon content Russian austenitic steel (EI-211) will be the reference material for low temperature (less than 450°C) components. The U.S. ferritic/martensitic steel HT-9, extensively tested for radiation stability in the U.S. fast reactor programs, will be the alternative material for cladding and other high temperature (greater than 450°C) components. The U.S. austenitic steel 316L (or possibly D-9), also extensively tested for radiation stability in the U.S. fast reactor performance, will be the alternative material for low temperature (less than 450°C) components. Other U.S. ferritic/martensitic steels will also be considered as alternative window materials.

The majority of the materials selection and testing activities will be completed before the freeze of the LBE loop test facility design, but irradiation testing of structural materials in an integrated (LBE and proton/spallation neutron) environment and continued material surveillance and certification efforts will continue through test facility operation and operation of the Target Test Facility. If the initial LBE corrosion tests show that certain US steels are compatible with LBE, much effort can be saved.

Duration: 11.5 years
Cost: $ 5.9M

1) Determine Expected Operating Conditions (temperature and core configuration). Determine design acceptance criteria for structural materials. This task consists of interacting with the nuclear design group to determine expected core operating parameters.

2) Develop Corrosion and Mechanical Property Acceptance Criteria. Many of the mechanical property acceptance criteria will be similar to those of LMFBR material.

3) Modify U.S. Steels for Corrosion Resistance. This step is an alternative if standard U.S. steels fail to perform in LBE coolant and a determination that manufacture of Russian steels will not be appropriate. The significant material costs are for radiation and testing of new materials. This task can be delayed into later stages of R&D.

4) QA of Russian Steels Database. This task consists of two steps intended to provide data for Russian materials that will support licensing of test and demonstration facilities in the U.S.

a) QA of Russian ferritic/martensitic steels -the Russians have already proven compatibility of EP 823 with LBE. This task is designed to provide an audit trail and acceptability criteria for using Russian data in a U.S. licensed design. If this audit trail cannot be successfully completed, irradiation testing may be necessary.

b) QA of Russian austenitic steels - the Russians have proven corrosion compatibility with their austenitic steels. Part of this task is designed to provide an audit trail and acceptability criteria for using Russian data in a U.S. licensed design. Russian Austenitic steels have not been tested to high fluence. This task assumes the target blanket has been designed to not allow non-core components to see higher dose than experienced by Russian steels.

5) Spallation Target Window Material Development. This task is designed to ensure that the target window material will maintain integrity (corrosion and mechanical properties) during ATW operation. The majority of this task is addessed in the Target Technology portion of this roadmap. However, the efforts described in this portion of the roadmap are intended to identify alloys with acceptable corrosion resistance in LBE for consideration as target window materials.

This activity should coincide with spallation target testing and Target Test Facility operation. Specific activities include selection of LBE-compatible material candidates, coordination with target development and testing efforts (such as proton and spallation neutron irradiation tests, and irradiation in LBE loops), and evaluation of alternative materials.

7) Materials surveillance and development program. -After the various test facilities (i.e., LBE test loops and the Target Test Facility) become operational, and integrated testing of window and other materials is finished, a surveillance program using small test samples inserted into the test facilities will be used to help determine the long term performance of the materials in the test facility system. Additionally, a new alloy development program should be initiated in which new concepts are tested in the test facility on test samples.

5.2.4 Spallation Products Effects

Spallation products are unique to the ATW system and were not encountered in Russian reactor experience. Spallation products may lead to corrosive effects on the passive oxide film protecting structural materials and to the buildup of oxides with the potential to limit coolant flow. Additionally, high vapor pressure species will evaporate out of the coolant and accumulate in the cover gas. The potential for corrosion and for radiological exposure increases with increasing concentrations of spallation products in the cover gas. Research and design tasks are aimed at understanding the effect of spallation products on the ATW cooling system and to designing

methods to maintain coolant and cover gas cleanliness. The concentration of spallation products is relatively small and is not expected to create significant coolant control problems, but a research program is necessary to confirm the absence of deleterious effects. If the initial calculation shows no significant changes to the coolant chemistry and oxygen control technique, the experiments can be delayed and performed at reduced scales. This task finishes the preparatory tasks before the end of the test facility design and carries out experimental studies in the test facility during its operation.

Duration: 8.6 years
Cost: $ 3.1M

1) Thermodynamic Assessment of Spallation Product Effects on LBE coolant. Using the calculated spallation product determined by efforts described in the Target Technology portion of this roadmap, calculations will be performed to determine the products that will be formed when the spallation products interact with the LBE coolant. Products are expected to fall into three distinct classes, volatile species that evaporate into the cover gas, species that form stable oxides in the coolant, and species that remain dissolved in the coolant. From these thermodynamic calculations, experiments will be designed to test the effects on corrosion and radioactivity buildup.

2) Surrogate Materials Experiments. Based on the thermodynamic calculations described above, surrogate materials will be introduced into coolant loop and cover gas simulations. Specific phenomena to be investigated include the effect of volatile species on cover gas chemistry, oxide formation and the effect on fluid flow and corrosion resistance, and the effect of dissolved species on corrosion resistance.

3) Determine Safety and Cleanup Needs. Systems for coolant and cover gas cleanup in both routine and abnormal conditions will be developed.

4) Test Spallation Product Effects in Systems - These tasks are designed to verify coolant system operation in prototypical systems (test loops and the Target Test Facility). The significant materials costs are for a pool-type LBE loop used to test the effects of complicated geometry on materials corrosion and coolant contamination and cleanup in LBE.

5.2.5 LBE Technology System Safety

This task is designed to provide an understanding of the limits of the oxygen control system on maintaining the passive oxide film. The task is also designed to provide the expected consequences of abnormal conditions along with any emergency response guidelines. As scheduled, this task finishes before the freeze of the test facility design.

Duration: 12 years
Cost: $ 3.5 M1) Hazard Assessment. The hazard assessment task is designed to determine at which oxygen levels the coolant control system fails and to determine the consequences of coolant or cover gas leaks from the system.

a) Oxygen limits of the coolant chemistry control system. This task is designed to determine the oxygen levels at which protective films fail, to determine the consequences of such a failure, and to develop recovery procedures.

b) Effect of oxygen ingress on heat transfer. This task is designed to understand the consequences of leaks of oxygen into the system on coolant quality and heat transfer properties.
c) Active coolant leaks. This task is designed to understand the consequences of leaks of radioactive LBE from the system.

d) Cover gas leaks. This task is designed to understand the consequences of leaks in the cover gas system.

2) Design basis accidents. This is designed to determine the design basis accident for the ATW system and will be done in conjunction with many of the other design groups.

3) Emergency response procedures and cleanup. This task is to design emergency response measures and cleanup means, given the understanding developed in the Hazard Assessment task described above.

4) Containment system design. Given the understanding of accident consequences developed in LBE.5.1, input will be provided to design of the STF and ATW containment systems.

5.2.6 Project administration and management

Overall management of the LBE subgroup project is needed for the duration of the ATW project. The project management needs are estimated at 1.5 FTE per year of the project. The project will also require $200K per year for general materials, supplies, and travel in the first 5 years and $100K per year afterwards.

Duration: 12 years
Cost: $ 6.3M

5.2.7 Estimated Cost and Schedule

Schedules and resources required for the Coolant Chemistry and Materials Compatibility tasks described above were estimated for the purposes of estimating the sequenced cost of developing a LBE-cooled ATW target and blanket system. The projected cost and schedule information developed in this effort, for the deployment-driven scenario assumed, are summarized in the tables contained in Attachment 3. Although year-by-year allocation is not shown in the tables, a linear distribution of the total effort and costs reflected in the tables was assumed (for simplicity); those assumptions provide the basis for the year-by-year summary of costs presented in Section 6. An average, fully-burdened effort rate of $220k per full-time equivalent year was assumed.

5.3 Heat Removal and Ancillary Systems

Safe and efficient operation of the subcritical facility must be assured to remain within design limits during both normal and off-normal operating conditions. A coolable core configuration must be guaranteed under any condition.

Major R&D areas necessary to achieve these objectives are: analytical and experimental investigations of heat removal characteristics of sodium and LBE under a variety of operational conditions and geometrical configurations; characterization of flow distribution through both analytical and experimental (simulation mock-ups) investigations; engineering of primary and secondary system components; design and evaluation of systems for active and residual heat removal in the core, transport loops, vessel and containment. Using the established ALMR and BRUS-300 designs as the basis for the ATW T&B system will allow a head start for use of both the sodium and the LBE coolant solutions.

Issues and tasks specific to the coolant medium are addressed in the following sections.

5.3.1 Heat Removal and Ancillary Systems for a LBE-Based ATW System

5.3.1.1 Introduction

Many of the issues associated with development of the target and blanket of an ATW system are related to removal of heat from the target and blanket and to the other systems that are required for reliable and safe operation of the target or blanket components. These issues are very similar to those associated with a critical (i.e., k-eff = 1) reactor in previous reactor development programs. Therefore, the experience gained from those programs is drawn upon to identify the issues and obstacles associated with successful development of an ATW system and to propose approaches to their resolution.

The portion of the ATW roadmap described in this section of the Target and Blanket Roadmap report describes an envisioned program for developing heat removal and ancillary systems for the target and blanket of a LBE-cooled ATW system. Included in the scope are the development activities for large components of the target and blanket that have unique application due to the utilization of LBE coolant. Other activities considered are those for development of systems to support or maintain the coolant material itself in a condition that allows safe and reliable operation.

5.3.1.2 GENERAL APPROACH

This portion of the roadmap was developed based on U.S experience in the development of heat removal and ancillary systems for sodium-cooled reactors. However, it should be noted that the U.S. has essentially no experience with lead or LBE as a nuclear coolant. Therefore, LBE technology in the U.S. is in an infancy stage, unlike sodium technology, which is considerably more mature. At the moment, it is difficult to gauge the relative maturity of Russian LBE technology, because Russian information has only just recently been made available to U.S. personnel. However, for this effort, Russian LBE technology is assumed to be substantially beyond the infancy stage (based on evidential Russian experience and expertise with LBE systems), but below the maturity level of sodium coolant (based on the limited size and number of installations or units that utilized LBE coolant).

For the purposes of proposing a research and development program for LBE systems, the assumption is made that technology will be transferred from Russia to the U.S., and that collaborative U.S.-Russian programs will be able to develop the technology for implementation in an ATW system. Again, U.S. experience with sodium systems is considered applicable, such that time for development is significantly decreased below that required if no such experience existed.

Consistent with the approach used for development of relatively large reactor components during the U.S. Liquid Metal Fast Breeder Reactor (LMFBR) and Advanced Liquid Metal Reactor (ALMR) programs, this roadmap calls for base technology assessment and development (including technology transfer) to be performed by the U.S. National Laboratories, with development, design, and fabrication of large-scale components to be performed by industry. It is envisioned that industrial participation would be incorporated into the program from the outset, allowing a smooth transition of development activities. For all components, a period of technology transfer and laboratory experimentation and testing is incorporated for the purposes of enhancing U.S. capability with LBE coolant technology prior to large-scale component development. This roadmap estimates resources required to develop, design, fabricate and test the component protoype units, with subsequent production of full-size components assumed to be part of separate design and construction contracts for specific ATW plants (e.g., the ATW Demonstration Plant). The technology for some ancillary systems (e.g., a Cover Gas Cleanup System) is developed to the point of its transfer to industry for design of large-scale units.

The development of sodium-related components historically included (and still includes, in countries such as France and Japan) a considerable amount of integral testing of large-scale, first-of-a-kind units. Such testing is employed to measure and verify performance under prototypic conditions and to validate models used in design and safety analysis. Therefore this roadmap incorporates integral testing of component prototypes, the results of which will be available to influence designs of full-scale components. It is envisioned that during the early years of component development, the selected industrial vendors will develop early designs of full-scale components based on transferred technology on the results of lab-scale testing in the National Laboratories. Important parameters and features of those designs will then be incorporated into designs for component prototypes, which will then be fabricated for integral testing. These component prototypes will likely be scaled to optimize testing and assessment in the integral test; a component development program management and integration function is proposed to accommodate that need and to ensure that component design and development efforts are consistent with requirments eveloving from parallel efforts to design ATW plant systems. The proposed

scheduling of the integral testing is such that modifications can be made to the full-scale designs based on test results prior to full-scale component fabrication for the ATW Demonstration Plant.

Finally, the U.S. no longer has a facility for performing such testing, and it is not clear that a facility suitable for testing using LBE is available abroad. Therefore, for completeness, this roadmap calls for establishment of a new Component Test Facility (CTF) in the U.S. The CTF is envisioned to be a facility established in an existing building at a U.S. National Laboratory or site and would be used for testing of components using LBE at high temperatures. The cost and schedule for establishing such a facility are roughly estimated and incorporated into the present roadmap.

5.3.1.3 LBE Technology Status and Technology Transfer

Successful use of LBE technology will be hinging on the transfer and implementation of Russian developed techniques, as no industrial expertise with LBE exists outside of Russia. Building upon Russian experience, a US-compliant technology base must be developed for heat removal components such as pumps, heat exchangers, steam generators, and ancillary equipment (e.g., refueling machines, cover gas equipment, and general flow diagnostics). Addressing these issues will require construction and operation of large-scale thermal-hydraulics test beds (LBE loops) but could be facilitated by the extensive use of existing Russian facilities and collaboration with European efforts in the same direction (Germany and Italy).

5.3.1.4 General Materials Selection Issues

For all components in the primary or secondary systems of a LBE ATW, compatibility of materials with LBE will be an important consideration. Because the environment of application will vary (i.e., some components will experience core outlet temperatures, while others will operate at cold-leg temperatures; some components will accommodate water in addition to LBE), a single materials solution is not likely. The technology development required for ensuring that materials can be used in a LBE environment is being considered elsewhere in the ATW Target and Blanket roadmap, so those specific issues are not addressed here. However, it is realized that successful materials performance will require appropriate materials selection and establishment and control of desired LBE chemistry. Russian experience also indicates that the presence of an engineered oxide film on the surface of LBE components will be essential to corrosion protection; therefore, the means to form those oxide films, either during component fabrication, during testing, or during plant startup must be devised. Although not detailed below, it is envisioned that materials issues will be a primary consideration in the development and design of each component and system.

5.3.1.5 Components and Systems

Specific components and systems envisioned to be part of a LBE-cooled ATW system are addressed individually in the following subsections. For each, the issues and obstacles associated with successful development that were identified as part of this effort and the proposed development path are described.

LBE Pumps

Issues

The primary objective for a pump development effort is to develop pumps for primary and secondary LBE flow that will achieve the required flow rates and operate reliably. Pump development for the ALMR program has focused on electromagnetic pumps, due primarily to size restrictions in the primary vessel, but also due to their demonstrated reliability and minimal required maintenance. However, initial conversations with Russian technologists indicate that electromagnetic coupling with LBE is sufficiently low as to render electromagnetic pumping of LBE impractical for large systems. Therefore, this roadmap considers only the development of mechanical pumps.

The density of LBE is relatively higher than that of other liquid coolants, such as water or sodium. Therefore, it is envisioned that attention will be given to development of a pump concept that is capable of a relatively high mass flow rate, with expected high torque on the pump shaft. Consideration will be given to suitable impeller and shaft designs. Also of concern are the materials to be in contact with LBE. However, because the pumps will be placed in the cold legs of the primary and secondary systems, the materials compatibility problems should not be as difficult as those encountered in the core or hot legs of the primary and secondary. Seal materials for pump shafts that are compatible with LBE must also be identified.

Approach

Discussions with Russian technologists indicate that Russia has considerable applicable experience with pumping of LBE. Therefore, the proposed pump development effort begins with efforts to transfer pump technology from Russia. After a short period of laboratory testing for development of potential concepts, development of large-scale pumps will begin. It is envisioned that National Laboratory personnel will prepare specifications for the prototype pump design, and will then manage the industrial development contract. Both primary and secondary pump designs will be considered, as the two are likely to have different sets of requirements. (The secondary pump is expected to be required to operate at higher system pressure and to provide a higher pump head.) Initial efforts will focus on the primary pump, with the secondary pump development building upon what is learned from the primary pump. The prototype pumps are to be delivered for testing by National Laboratory personnel (in collaboration with the industrial contractor) after six years. Integrated testing will be performed using heated LBE loops at the CTF, or at some other appropriate facility. The results from the successful testing of the prototype will be used to modify and validate the design of the primary and secondary pumps for the ATW Demonstration Plant..

LBE Steam Generators

Issues

Experience with steam generators for sodium-cooled reactors has shown that large-unit fabrication techniques, materials selection, and accommodation of maintenance and inspection are all important considerations for steam generator design. Such issues are anticipated for the ATW LBE system as well. An understanding of LBE flow and heat transfer characteristics will be essential to design of efficient steam generators. As with other components, selection of materials for compatibility with LBE is important, including consideration of joining techniques and the resulting material conditions, such as weld and heat-affected zones produced by welding tubes to sheets. Because the steam generator will see relatively high LBE temperatures, materials compatibility will be of more concern than for cold-leg components, such the pumps. The ALMR program considered single-pass designs that used helical coils and double-wall designs but selected

the helical coil design. Such designs may also be considered for the ATW LBE system.. Double-wall tubes may be attractive for accommodating compatibility with high temperature water on one side of the tubes and high-temperature LBE on the other.

Approach

Because many of the issues addressed with steam generators are similar to those encountered in the development of intermediate heat exchangers, the technology transfer and laboratory concept development phases for these two sets of components are proposed to be conducted jointly. Development, design and fabrication of the large-scale components will be pursued separately, however. As with the pump, a six-year period for delivery of a prototype steam generator is proposed, followed by integrated testing at the CTF. However, because the concepts for the ATW Demonstration Plant do not incorporate a power plant (i.e., heat is to be exchanged to the atmosphere), steam generator development is not required early in the ATW program. Therefore, the proposed plan calls for prototype steam generator to begin later in the program, after a decision regarding coolant type.

LBE Intermediate Heat Exchangers

Issues

It is envisioned that an intermediate heat exchanger (designed to transfer heat from a LBE primary system to a LBE secondary system) can be designed in a straightforward manner. Issues anticipated include design for efficient heat transfer in LBE and joining of tubes to tube sheets in manner that will not produce affected material zones that would be susceptible to LBE corrosion.

Approach

Because much of the technology to be employed for intermediate heat exchangers is similar to that employed for steam generators, the technology transfer and laboratory concept development portions of this development effort will be combined with those for the steam generators, as described above. Because of the relatively simpler design of the intermediate heat exchanger (i.e., relative to pumps or steam generators), a four-year period is believed sufficient for development and delivery of a prototype unit. The results from the successful testing of the prototype will be used to modify or validate the design of the intermediate heat exchanger for the ATW Demonstration Plant..

Vessel Features: Support Structures

Issues

Although many aspects of core support will be detail that is addressed during the design of a specific unit, it is envisioned that accommodation of LBE primary coolant may influence some aspects. Support of blanket assemblies and the spallation target must be accomplished in a manner that will not obstruct LBE flow, by either forced or natural convection. Support members must be designed to avoid excessive loading from the inertia of LBE flow. Support schemes will be further complicated by the relatively high density of the coolant. Materials selection for these components must consider the temperature and flow conditions of the coolant.

Approach

A relatively small amount of effort is proposed for technology transfer, although it is anticipated that Russian viewpoints on these matters will be valuable. Effort is also proposed for laboratory

testing and concept development and for analysis. Developed concepts will be incorporated into conceptual designs for the ATW Demonstration Plant, to be ultimately manifested in design and construction of that plant.

Vessel Features: Head

Issues

It is anticipated that utilization of an LBE primary will have implications for the design of over-the-core features. As with sodium-cooled reactors, it will be important to seal air out of the cover space over the LBE primary tank (although the concern is over control of oxygen potential in the primary system for corrosion control, rather than to avoid a reaction as with sodium and air). If a freezable seal for a rotating head cover is to be employed, as with EBR-II, then a suitable seal alloy that is compatible with LBE vapors and plenum cover gases must be identified. Depending on the control scheme to be implemented, control hardware might need to be accommodated through the vessel head, so any vessel head concepts must be compatible.

Approach

Russian experience and design concepts will be transferred, and concepts will be developed in the laboratory. Successful concepts will be incorporated into conceptual designs for the ATW Demonstration Plant, to be ultimately manifested in design and construction of that plant

Vessel Features: In-Core Fuel Handling

Issues

Although in-core fuel handling has been successfully deployed at several operating sodium-cooled reactors, fuel handling in the LBE primary tank of an ATW system will be more problematic – primarily due to the high density of the LBE coolant, which will likely be higher than that of the fuel material. Techniques for handling fuel assemblies in molten LBE must be developed, and conceptual designs of fuel handling apparatus must be compatible with concepts developed and proposed for the vessel head and for internal support structures.

Approach

Russian experience with fuel handling in LBE will be sought during a period of technology transfer, followed by laboratory testing and experimentation. This activity will culminate in the design, fabrication, and testing of a prototype fuel-handling device that will provide experience for the eventual design of the fuel handling system for the ATW Demonstration Plant. The testing of the prototype will be performed at the proposed CTF.

Flow Design

Issues

Because the U.S. has no substantial experience with LBE as a heat transfer medium, it is anticipated that pointed effort must be applied to understanding the heat transfer and flow behavior of LBE. This information must then be applied to conceptual design of the primary system. Some

specific issues anticipated include the optimal design of fuel assemblies to ensure adequate heat transfer and design of the primary tank and internal structures to avoid LBE stagnation (which is shown by Russian experience to have derogatory effects on LBE corrosion resistance). Finally, models for LBE heat transfer and flow through the conceptual blanket and primary vessel designs must be validated through measurements made with a scaled model.

Approach

As Russian experience with LBE flow in a reactor core is understood, laboratory measurements of LBE heat transfer characteristics will be made and flow models will be developed. As the conceptual design for the blanket and primary vessel are formed, a scaled model will be assembled for measurement of key flow parameters, which will then be used to validate and or modify the flow models. Models of fuel assemblies will also be fabricated for experimental evaluation of fuel assembly design features and for validation of flow models.

Window Cooling

Issues

Whether the LBE spallation target is integral to the primary system, or whether it is comprised of a separate LBE loop, it will employ a window between the proton beam channel and the LBE environment. Although the design of such a window is still undetermined, it is anticipated that substantial amounts of heat will be deposited in the window by the incident proton beam. Heat transfer models for the prospective window geometries must be developed, and conceptual designs for enhancing heat removal from the window must be developed and models validated.

Approach

Russian experience and ideas will be sought, however, it appears that the Russians have not yet operated an LBE spallation source. Laboratory tests to measure heat transfer characteristics and to validate heat transfer models for selected window geometries will be performed. Irradiation testing of the TC-1 target test will provide a prototype test of a window cooling design currently under consideration as part of another program. Specific window designs and operation of the TC-1 irradiation test are considered in the Target Development portion of the Target and Blanket roadmap.

Cover Gas Cleanup and Oxygen Control System

Issues

Although the technology development (including technology transfer from Russia) for establishing and maintaining coolant chemistry is addressed in the LBE Chemistry and Materials Compatibility portion of the Target and Blanket Roadmap report, the development of system components that implement that technology is addressed here The technology to maintain cover gas chemistry above a sodium-cooled, pool-type reactor is well established; however further development will be required for a LBE ATW cover gas system. Such a system will be required to remove not only volatile fission products but also volatile spallation products. Oxygen potential in the LBE coolant will be managed partially by control of cover gas chemistry, which places additional requirements

on the cover gas system. It is expected that the chemistry of the cover gas, and the complications of volatile condensate will introduce unique challenges.

Approach

The technology transfer and development associated with cover gas chemistry will be addressed by the activities described in the LBE Chemistry and Materials Compatibility portion of the Target and Blanket roadmap. The effort described here will focus on design and testing of a cover gas cleanup system prototype. The system will be assembled and operated to demonstrate that oxygen content can be controlled to desired levels using the principles learned through technology transfer.

Cold Trap and Filter Design

Issues

Based on information learned thus far from Russian experience, it is clear that filtering of solids from the LBE coolant will be required – this will be particularly true for oxides that form in the partially oxidizing environment of the LBE coolant. As with sodium, it is expected that some contaminants will precipitate out from the coolant as solids, therefore a cold trap is also warranted. Because the chemistry technique applied to control these contaminants in the LBE will be new for the U.S., a development effort leading to a prototype system is proposed.

Approach

As with the cover gas clean-up system, the technology transfer and development associated with cover gas chemistry will be addressed by the activities described in the LBE Chemistry and Materials Compatibility portion of the Target and Blanket roadmap. The effort described here will focus on design and testing of a cold trap and filter prototype.

5.3.1.6 Component Test Facility (CTF)

As is consistent with U.S. and international programs to develop components for large nuclear systems, this roadmap calls for integral testing of many large, first-of-a-kind components. It is envisioned that such testing will entail placement of these components into a large LBE loop system (likely to be comprised of primary and secondary loops) for sequential and simultaneous testing to demonstrate performance in high-temperature LBE and to validate performance models. Most recently, this type of testing in the U.S. was performed. using facilities at ETEC, at Hanford, and elsewhere in the world. Because no U.S. facilities exist for testing of large-scale components in high-temperature LBE, establishment of a new Component Test Facility (CTF) in the U.S. is proposed as part of this roadmap. Other facilities for testing of large-scale LBE components may be available, however, in Russia. If such facilities prove to be suitable, and if arrangements for their use can be established, then substantial cost savings over those estimated in this roadmap might be realized.

Although there is no stated preference for a government-owned CTF over a privately-owned and operated CTF, the facility is proposed here to be placed into an existing building at a U.S. National Laboratory or DOE site. It is believed that such a placement could best accommodate the proposed schedule at a minimal cost. At the time of this roadmap activity, only a conceptual image of the CTF was envisaged, with the details of the testing to be accommodated within it

undetermined. Therefore, the facility construction cost is roughly estimated here at $50M. Annual operating costs are estimated at about $10M, with about 40% of that cost for Materials and Supplies (M&S). Although a two-year test period is believed sufficient for completion of the test program, funding for facility operations is proposed for a longer duration to accommodate further development work or unforeseen difficulties.

5.3.1.7 Continued Component Evaluation and Improvement

After the ATW Demonstration Plant has been completed and is operational, there will be a continued need to evaluate the performance of the operating components. In addition to monitoring the existing components to troubleshoot and to ensure reliability, the objectives of the program will be to gather data for improvement upon the component designs. This effort is envisioned to be relatively small, but sufficient to ensure adequate surveillance of the equipment and archiving and assessment of its performance.

5.3.2 Heat Removal and Ancillary Systems for a Sodium-Based ATW System

Development of sodium technology was largely abandoned in the US upon the termination of the ALMR program, before many of the components for the ALMR concept could be designed and constructed. No large-scale capability for sodium component engineering and manufacturing expertise exists today in the US, although significant and active expertise exists in Japan and France. Extension of the sodium heat removal technology to the level required for implementation in the ATW T&B system will require significant effort, and will likely benefit from the transfer of new technology from France and Japan, and the construction of a new sodium thermal-hydraulic large-scale test facility. Successful construction and operation of the ATW Demonstration Facility will require the fabrication of large-scale components.

Some major differences exist between the issues to be addressed for LBE systems and sodium systems. 1) Materials compatibility is not an issue with sodium and the stainless steels typically considered for sodium component usage. 2) Coolant chemistry control is well established with existing and previously operated sodium-cooled facilities, although the potential presence of spallation products in the sodium coolant has not been addressed for operating systems. 3) The thermal hydraulic characteristics for sodium are well known, whereas those for LBE are not well known in the U.S.

It was not possible to provide a thorough assessment of the status of sodium component technology as part of this roadmapping effort. However, it is assumed that because sodium components have been successfully built and operated at varying scale at several different sodium-cooled reactors, the technology for secondary pumps, heat exchangers, and steam generators is sufficiently mature for implementation in the proposed ATW Demonstration Plant. An exception is the electromagnetic primary pump, the development of which is included in this roadmap. The ALMR program called for implementation of electromagnetic pumps for the primary system, partly because such pumps could be better accommodated than mechanical pumps within the limited space available. However, the development of those pumps was not completed before termination of the ALMR program. It is proposed here that such a pump design be developed and tested as a prototype in a new Component Test Facility (CTF).

Development of other ancillary components is provided for in this roadmap, including development of new support structure, vessel head, and fuel handling concepts concepts. Other work will include flow mock-up of proposed blanket configurations, including flow coupling with the tungsten spallation target.

The CTF required to support development of components for a Na-cooled system is envisioned to be smaller than that required for development of LBE components – primarily because several LBE components would be tested, whereas only a newly-developed electromagnetic pump would

be tested in high-temperature sodium. It seems likely that only a single high-temperature sodium loop would be required. For this reason, the CTF for Na component testing is proposed to cost 40% less to build and operate than that for LBE component testing.

5.3.3 Estimated Cost and Schedule

Schedules and resources required for the Heat Removal and Ancillary System tasks described above were estimated for the purposes of estimating the sequenced cost of developing a sodium-cooled and LBE-cooled ATW target and blanket system. The projected cost and schedule information developed in this effort, for the deployment-driven scenario assumed, are summarized in the tables contained in Attachments 4 and 5. Although year-by-year allocation is not shown in the tables, a linear distribution of the total effort and costs reflected in the tables was assumed (for simplicity); those assumptions provide the basis for the year-by-year summary of costs presented in Section 6. An average, fully-burdened effort rate of $220k per full-time equivalent year was assumed.

5.4 Development Issues Associated with Target Design

5.4.1 General Issues

The ATW system consists of a proton accelerator, a target for the production of spallation neutrons, a sub-critical blanket and a reprocessing/fuel production system. Three separate coolant technologies are to be investigated in the early stages of the ATW development program. These include sodium, lead-bismuth eutectic (LBE), and perhaps helium.

Within the LBE technology, there are two target geometries that have been proposed, an integral target, and a separate target. As described in section 3, the integral target is comprised of a thimble tube that is inserted into the LBE coolant in the middle of the core. The tube maintains the pressure boundary between the accelerator vacuum and the core coolant. The end of the tube is directly in the path of the proton beam, and is cooled in an integral sense with the reactor coolant. The separate LBE target uses a "windowless" design in that no structure is directly in the path of the beam other than LBE. The coolant for this target is separate from the core coolant, and requires a separate cleanup and heat removal system. This structure would experience less irradiation damage than the integral target.

At the appropriate time, a primary coolant technology will be selected. The decision on the best option for the path forward on ATW should reflect serious evaluation of the sodium and LBE options (recognizing that helium may also be considered).

Regardless of the technology chosen, the research and development activities that are required for the successful demonstration of an ATW target include:

a) selection and qualification of metals and alloys for service under anticipated target conditions,
b) determination of effect of service conditions (radiation damage and material/coolant interactions) on materials properties and on component behavior,
c) documentation of materials properties as a function of anticipated service conditions and development of these properties if not available,

d) determination of corrosion processes, corrosion rates and corrosion control and mitigation practices for materials exposed to ATW target coolants and interfaces
e) development of structural design criteria that are consistent with the anticipated, service-induced degradation of mechanical properties and also assure the safe and efficient utilization of window, and structural materials in the target
f) prototyping of target components and confirmation of the compatibility of the design materials with the selected component fabrication and joining technologies,
g) determination of realistic design lifetimes for systems and components,
h) validation of thermal-hydraulic and safety codes,
i) devolpment of necessary physics methods and data, and
j) in the case of liquid metals, determination of the effects of spallation reactions on coolant chemistry.

The materials issues associated with all three technologies are very challenging, and will require irradiation tests during the development phase. The operating and irradiation conditions for the targets are shown in Table 5.4.1. A description of the issues associated with the development of each target system is given in the sections below.

5.4.2 Integral Lead-Bismuth Cooled System Target Development

The issues associated with the development of the integral LBE target system include the selection of alloys, material properties, changes to materials due to irradiation, changes to the LBE chemistry, nuclear data, design criteria, thermal hydraulics and validation of codes and methods. These issues and the development tasks proposed to address those issues are discussed separately below. A summary of the tasks discussed here is presented in Table 5.4.2, which appears at the end of this section.

5.4.2.1 Selection of Alloys

The use of the LBE system will require the development of special alloys not used in the U.S. A major issue is the selection and validation of these alloys. The process to achieve this involves:

1. establishing design data needs (material properties such as strength, ductility and fracture toughness as a function of displacement damage, in-beam and out-of-beam corrosion resistance, swelling tendencies, etc. that are required by designers),

2. providing irradiation and corrosion test programs to satisfy the design data needs,
3. conducting irradiation tests on near prototypical structures and components that were designed, fabricated and irradiated to simulate anticipated service situations,
4. post irradiation examination and analysis of test samples, structures and components, and
5. documenting the test results and related laboratory/industrial information in a Materials Handbook that provides the designers with a single source for properties.

To manage and guide this program a multi-laboratory, ATW materials steering committee is needed. This will help assure complete integration of the materials related needs that emerge from the various sub-groups.

Estimated Cost: $1.5M over 4 years

Table 5.4.1 Summary of Operating Conditions

Target Type	Structure	Coolant	Temperature (C)	Proton Flux p/cm2 s	Neutron Flux n/cm2 s	Damage Rate dpa/y	Hydrogen generation appm/y	Helium generation appm/y
Integral LBE	Russian Alloy EP-823	LBE	500	2.2×10^{14}	7×10^{15} (0.7×10^{15} sp.n.)	p: 15 Sp.n: 23 f.n.: 7 total: 45	p: 13320 Sp.n.:1590 f.n. : 50 total: 14960	p: 2580 Sp.n. 300 f.n. 10 total: 2890
Separate LBE (no window)	Russian Alloy EP-823	LBE	500	4.4×10^{12}	7×10^{15}	30	1640	310
Sodium-Cooled Tungsten	HT-9	Sodium	400	2.2×10^{14}	7×10^{15} (0.7×10^{15} sp.n.)	p: 15 Sp.n: 23 f.n.: 7 total: 45	p: 13320 Sp.n.:1590 f.n. : 50 total: 14960	p: 2580 Sp.n. 300 f.n. 10 total: 2890

The assumptions and bases for these values are:
A 25-mA current, uniformly spread over a 30-cm dia. circle, 75% availability
dpa, He/dpa, H_2/dpa for protons and spallation neutrons are based on IPPE calculation for TC-1, corroborated roughly with APT Project calculations
dpa, fission neutron: 30% dpa due to sp.n = dpa due to f.n 10xsp.n., He/dpa, H2/dpa for f.n. conservatively estimated at 1/10 that of sp.n.

5.4.2.2 Properties of all materials in the unirradiated condition

The lack of qualified, US alloys for service in the ATW-LBE target will necessitate an alloy development program and/or a program to transfer the technologies associated with production and fabrication of the silicon-modified alloys from Russian to U.S steel producers. Prior to design, these alloys must be tested to determine their properties in the unirradiated condition.

The activities required to determine properties of unirradiated alloys are described in the Coolant Chemistry and Materials Compatibility portion of this roadmap and are not duplicated here.

5.4.2.3 Change in material properties due to proton and neutron irradiation, and exposure to the LBE coolant

Perhaps the most significant issue with regard to the target design, and certainly the most costly to address, is the change in materials properties due to irradiation and exposure to LBE coolant. Iron, chromium and nickel, components of many structural steels, are soluble in liquid lead bismuth and experience corrosion and erosion corrosion when exposed in flowing, liquid lead bismuth. Experience suggests that US structural alloys including the 300 and 400 series stainless steels and ferritic/martensitic steels such as HT–9, will demonstrate unsatisfactory performance in the lead bismuth target/coolant environments, even in the absence of proton/neutron irradiation. However, materials compatibility with the liquid lead bismuth can be significantly improved by silicon additions to steels and control of the oxygen content in the lead-bismuth alloy. Use of silicon modified steels and oxygen control measures apparently provide the basis for successful application of LBE coolant technologies to several Russian nuclear systems, including Alpha-class nuclear submarines. The Russian experience, which apparently includes the use of silicon-modified austenitic and ferritic/martensitic steels, establishes positive expectations for successful use of the LBE as a coolant for the ATW system. However, extrapolation of the Russian experience to the beam window and other ATW components exposed to high-energy proton/neutron spectra should not be attempted without demonstration of acceptable performance and an understanding of the irradiation behavior of the materials.

There is little or no data on the effects of high-energy proton beams and spallation neutrons on materials properties at the temperatures of interest to the ATW. Therefore, an irradiation test program is necessary to quantify the effects of displacement damage, hydrogen and helium build-up and the accumulation of spallation products on materials properties. Recent studies of candidate materials for the Accelerator Production of Tritium (APT), irradiated and tested at or near 130°C range have shown that fission reactor data is not an accurate predictor of irradiation effects by high-energy particle beams. These studies also confirmed the accumulation of high levels of hydrogen and helium in the irradiated materials. The temperatures anticipated for the LBE coolant in the ATW are in the range where helium embrittlement of steels is well documented. These observations illustrate the necessity for irradiation testing under anticipated ATW temperatures and proton/neutron spectra.

As shown in the operating conditions summary the yearly irradiation dose is very high (30 - 45 dpa per year), and the amount of hydrogen and helium generation substantial. To approach these types of damage rates, an irradiation at a high-power beam facility (such as LANSCE) is required. In addition the irradiation must be performed in a flowing LBE loop with chemistry control in order to simulate the ATW operating environment. As shown in the following table, the cost of such an

experiment is large. This cost and time estimate is based on recent APT experience, and is very reliable. It is expected that a minimum of 2 inserts will be required, therefore a task for construction of a second insert is added.

Task	Number of months	Cost ($M)
Design of insert	6	0.7
Construction of insert	4	0.3
Design, construction of second insert	10	1.0
Attach to shield, perform checks, install into A-6	5	2.1
Irradiate for 6 months	9	11.06
Cooldown	4	0.7
Remove Insert	1	0.2
Package and ship specimens	3	0.3
Post irradiation examination	6	1.0
Analyze Data and Report	3	0.4
Total	41	17.76

5.4.2.4 Approved Structural Design Criteria

Because the irradiation exposures in the APT tests frequently reduced the uniform elongation of metals to 1% or less, current AMSE codes and standards may not be appropriate for ATW application. Therefore, structural design criteria are required to supplement ASME codes and standards. This task can make use of the criteria developed for APT, and modify them for the ATW alloys.

Estimated Cost: $0.3M

5.4.2.5 Nuclear Data for Pb and Bi

New cross section databases that extend up to 150 MeV for Pb and Bi into MCNPX radiation transport simulations must to be implemented in codes that predict protonic and neutronic behavior of the LBE target. The impact on predictions of neutron production, heating, and spallation product formation need to be assessed. This is a small effort because these cross sections have already been produced under APT support; however, the implementation of these cross sections into MCNPX requires further work. The spallation product inventory needs to be determined experimentally. This will be a major constituent of the source term for safety calculations. Two subtasks are proposed.

Subtask 1:
Implement new cross section databases that extend up to 150 MeV for Pb and Bi into MCNPX radiation transport simulations.

Estimated Cost: $80k (0.3 FTE) for 1-year

Subtask 2: Three different experiments to determine spallation and fission product inventory are proposed: 1) investigate thin plus thick target spallation products in Pb/Bi; 2) investigate fission product yields versus Z and A in thin-targets; and 3) study Bi and Pb neutron-induced reactions using the GEANIE gamma-ray detector.

Estimated Cost: $ 2.4M over 5 years

5.4.2.6 Validated Physics Methods and Codes

Physics models need to be developed for Pb/Bi spallation/fission to enhance intranuclear cascade codes for simulations of spallation/fission in Pb/Bi. This will result in more accurate predictions of neutron spallation production, and the production of fission/spallation products in the target which impact target radioactivity and radiation damage. Time-dependence of the decays of the products can be calculated with the CINDER code

Estimated Cost: $1.2M over 5 years

5.4.2.7 Target fluid mechanics

The target fluid mechanical design is integral with that of the core coolant. The coolant flow distribution through the target must be validated to ensure proper cooling of all structures. Development tasks presented in the Heat Removal and Ancillary Systems portion of this roadmap will address this issue.

5.4.2.8 Release of spallation products from LBE

The spallation products in the LBE coolant are major constituents of the source term for safety calculations. Given the spallation product inventory, we must understand the amount of release to the environment during accident conditions.

Out-of-beam experiments will be performed to determine the release of spallation and activation products from LBE to the surrounding systems or environment. This activity is important for the understanding of the source term for assumed accident conditions. These experiments will determine the release rates from LBE at various temperatures.

Estimated Cost: $0.8M over 1 year

5.4.2.9 Validate natural circulation for residual heat removal

If natural circulation heat removal is to be used for decay heat, the heat transfer performance of the target design must be validated. Development tasks presented in the Heat Removal and Ancillary Systems portion of this roadmap will address this issue.

5.4.2.10 Validate LBE system thermal hydraulic code for use with target evaluation

The prediction of the thermal-hydraulic behavior during normal and abnormal operations of the proposed designs will need to be addressed in detail during design. Design codes are needed to provide an adequate and safe ATW design. The results of studies generated using such codes will

yield guidelines to designers and to the safety and operations staff, who may modify the design and must develop procedures for efficient and safe operation.

This proposed activity will provide the development of an adequate design and analysis tool for evaluating LBE target designs and performance. Analyses will yield guidelines to designers and to the safety and operations staff, who may modify the design and must develop procedures for efficient and safe operation. The single major task is the development of the computational tool, i.e., TRAC-LBE. TRAC can model a fixed-composition liquid eutectic system having the following modifications:

- The thermal and caloric equations of state are inserted for the LBE.
- The single-phase wall heat-transfer relations are modified for liquid metals.
- Internal heat generation is added to the LBE. This requires a marker that switches from the steady-state value when the material is in the proton beam and the beam is on to a greatly reduced value that is characteristic of the initial decay rate once the material leaves the beam volume or the beam is turned off.
- An initial input specification is created for the primary and intermediate loops for ATW.
- We will generalize the advection capability to include an arbitrary number of components, each with its own source and sink terms. This will accommodate tracking polonium and any number of oxides, along with oxygen, hydrogen, and water. Source and sink terms will be added to include generation, destruction, and plateout.
- Chemical reaction rates are implemented.
- Deposition and entrainment models of dross are developed and implemented.
- Models will be developed to modify the wall friction and wall heat transfer based on significant oxide formation and collection.
- Experimental data will be benchmarked.
- Scoping calculations will be run on the proposed ATW system, including several accident scenarios.
- All tasks will be documented in a final report.

Estimated Cost: $0.9M over 3 years

5.4.3 Separate Lead-Bismuth Cooled System Target Development

Most of the issues associated with the development of the separate LBE target system are identical to the integral system. These include the selection of alloys, material properties, changes to materials due to irradiation, changes to the LBE chemistry, nuclear data, design criteria, and validation of codes and methods. Because these issues and associated development tasks are identical, they will not be repeated here.

The major issue that is different from the integral system is related to the development of a windowless target system. The thermal-hydraulic design of such a system needs to be tested at near full scale to ensure stability of the vacuum to LBE interface and to understand that amount of vaporization of LBE into the vacuum and the condensation of that vapor on structures upstream.

Table 5.4.2 Summary of Development Activities for the Integral LBE Target

Data Needs Prior to Start of Final Design of Demo Plant	Development Strategy	Est. Cost ($M)	Time Required
Selection of Alloys	• Establish Materials Steering Committee to guide alloy selection, materials out-of-beam tests, materials in-beam tests, review of foreign data, and development of a Materials Handbook	1.5	4 years
Properties of all materials in the unirradiated condition	• Reduce available data. • Transfer data from Russia • Perform testing as required • Document in Materials Handbook (start with APT Handbook)	(1)	2 years
Change in material properties due to proton and neutron irradiation, and exposure to the LBE coolant	• Use any applicable data from APT program. • Collaborate with small scale experiment at PSI. • Perform irradiation experiment at a high power proton beam (e.g. LANSCE) facility. • Perform post irradiation examination of material samples. • Document in Materials Handbook	17.76	4 years
Change in LBE coolant due to proton and neutron irradiation, and the effect on LBE chemistry	• Perform out-of-beam tests with H injection to simulate proton beam. • Perform in-beam experiments (in conjunction with item above) to determine effects on LBE chemistry, and validation of mitigation scheme. • Document in Materials Handbook	(1)	3 years
Processes for fabrication (forming, welding, etc.)	• Perform manufacturing prototyping of critical components and joints.	2	2 years
Approved Structural Design Criteria	• Start with APT Structural Design Criteria and adapt to ATW needs	0.3	1 year

Continued on next page

Table 5.4.2 Continued

Data Needs Prior to Start of Final Design of Demo Plant	Development Strategy	Est. Cost ($M)	Time Required
Nuclear Data for Pb and Bi	• Reduce available data • Measure Bi capture cross section • Measure spallation product yields in Pb and Bi	2.4	5 years
Validated Physics Methods and Codes	• Develop accurate model of proton and neutron induced fission in Pb and Bi. • Maintain and enhance MCNPX and Cinder90 • Produce data library for Bi up to 150 MeV • Validate codes and models against data	1.2	5 years
Validate target fluid mechanical design	• Perform scaling analysis • Perform scaled non-LBE tests • Perform final LBE test	(2)	4 years
Release of spallation products from LBE (to support source term)	• Perform small scale experiments with doped LBE	0.4	1 year
Validate natural circulation for residual heat removal	• Perform scaled out of beam test. Can be done in conjunction with other out of beam tests.	(2)	2 years
Validate LBE system thermal hydraulic code	• Update TRAC with LBE properties • Validate against experimental data	0.9	3 years
Total		26.4	5 years

1. Issue addressed by Coolant Chemistry and Materials Compatibility portion of this roadmap.
2. Issue addressed by Heat Removal and Ancillary Systems portion of this roadmap.

5.4.4 Sodium Cooled System Target Development

As stated in Section 3, the sodium-cooled target currently envisioned would be solid tungsten, clad in stainless steel (HT-9). The target assemblies would be long enough to completely attenuate the proton beam. A thimble tube, which provides the pressure boundary between the accelerator vacuum and the sodium coolant, would be located above the target assembly, and would be cooled by the sodium. The thimble tube material and the stainless steel cladding for the tungsten target would experience a combined proton and neutron environment in the presence of the sodium coolant.

The issues associated with the development of the sodium-cooled target system include the changes to materials properties due to irradiation, changes to the sodium chemistry, nuclear data, design criteria, thermal hydraulics and validation of codes and methods. The issues associated with the sodium-cooled tungsten target are described here. A summary of the tasks proposed to address these issues is presented in Table 5.4.4, which appears at the end of this section. Because the proposed task descriptions are essentially the same as the similarly-titled tasks for development of an LBE target, the detailed descriptions are not presented.

As described in Section 5.4.2 relative to the integral LBE target, evolution of material properties under proton and neutron irradiation. However, previous experience has shown that, in general, sodium coolant systems (with properly maintained chemistry) do not degrade stainless steel alloys. The U.S. LMFBR, IFR and ALMR programs had collected significant amounts of data on the performance of various stainless steels in sodium-cooled, fast-spectrum environments. However, there is little or no data regarding the synergistic effects of proton and neutron irradiation on the properties of these materials. Therefore, an irradiation test program similar to that described for the integral LBE target is proposed for development of this target design.

5.4.4.1 Approved Structural Design Criteria

Because the irradiation exposures in the APT tests frequently reduced the uniform elongation of metals to 1% or less, current AMSE codes and standards may not be appropriate for ATW application. Therefore, structural design criteria are required to supplement ASME codes and standards. This task can make use of the criteria developed for APT, and modify them for the ATW alloys.

5.4.4.2 Nuclear Data for Sodium

New cross section databases that extend up to 150 MeV for sodium need to be implemented. The impact on predictions of neutron production, heating, and spallation product formation need to be assessed.

Because sodium coolant will exposed to the proton beam, sodium spallation will occur and many isotopes will be generated in the coolant that are not present in reactor systems. For example, Na-22 will be produced, which cannot be removed with cold traps. The yields of this isotope and others must be determined to enable evaluation of the source term for accident conditions, and the evaluation of various plant component designs for operational exposure implications.

5.4.4.3 Target fluid mechanics

The coolant flow distribution through the target must be validated to ensure proper cooling of all structures. Assuming upflow through the tungsten targets, thimble tube window will be cooled with hot sodium. This design needs careful attention because any leak will flood the vacuum tube with sodium.

5.4.4.4 Release of spallation products from sodium

Sodium spallation products will be generated directly in the coolant, while smaller quantities of tungsten spallation products will likely migrate through the target cladding into the coolant or may be released through a target cladding breach. The generation and release of spallation products into the coolant must, therefore, be evaluated.

5.4.4.5 Validate natural circulation for residual heat removal

If natural circulation heat removal is to be used for decay heat, the heat transfer performance of the target design must be validated. Development tasks presented in the Heat Removal and Ancillary Systems portion of this roadmap will address this issue.

5.4.4.6 Validate LBE system thermal hydraulic code

The prediction of the thermal-hydraulic behavior during normal and abnormal operations of the proposed designs will need to be addressed in detail during design. Design codes are needed to provide an adequate and safe ATW design. The results of studies generated using such codes will yield guidelines to designers and to the safety and operations staff, who may modify the design and must develop procedures for efficient and safe operation.

5.4.5 Integral Target Testing

For any of the types of targets being currently proposed for ATW development, testing of a target prototype with an incident proton beam will be necessary to confirm target performance and to identify performance problems or limitations. Therefore, the development plan for each target type includes testing in a Target Test Facility, proposed to be built at the Los Alamos Neutron Science Center (LANSCE). The envisioned facility will provide for testing of target prototypes that use loops of the specified coolant (i.e., Na or LBE), allowing for measurement of spallation neutron and product yields and spectra, and assessment of techniques to address concerns over spallation product accumulation. The incorporation of fissile material to test and validate the coupling of a spallation target with a fissile multiplication blanket is also proposed.

The initial LBE target proposed for testing is the Target Complex-1 experiment, which is currently begin developed under a joint program between LANL and the International Science and Technology Center (ISTC# 559). Following that test, a second target test is proposed, which will include a small amount of fissile material (a small value of k-eff is envisioned). For the Na-cooled W target, a single test unit is proposed, which will be used initially without and then with a fissile material.

5.4.6 Estimated Cost and Schedule

Schedules and resources required for the Target Development tasks described above were estimated for the purposes of estimating the sequenced cost of developing a sodium-cooled and

LBE-cooled ATW target and blanket system. The projected cost and schedule information developed in this effort, for the deployment-driven scenario assumed, are summarized in the tables contained in Attachments 6 and 7. Although year-by-year allocation is not shown in the tables, a linear distribution of the total effort and costs reflected in the tables was assumed (for simplicity); those assumptions provide the basis for the year-by-year summary of costs presented in Section 6. An average, fully-burdened effort rate of $220k per full-time equivalent year was assumed.

Table 5.4.4 Summary of Development Activities for the Sodium Cooled Target

Data Needs Prior to Start of Final Design of Demo Plant	Development Strategy	Est. Cost ($M)	Time Required
Change in material properties due to proton and neutron irradiation, and exposure to the sodium coolant	• Use any applicable data from APT program. • Collaborate with small scale experiment at PSI. • Perform irradiation experiment at a high power proton beam (e.g. LANSCE) facility. • Perform post irradiation examination of material samples. • Document in Materials Handbook	17.76	4 years
Change in sodium coolant due to proton and neutron irradiation, and the effect on sodium chemistry	• Perform out-of-beam tests with H injection to simulate proton beam. • Perform in-beam experiments (in conjuction with item above) to determine effects on LBE chemistry, and validation of mitigation scheme. • Document in Materials Handbook	2.5	3 years
Approved Structural Design Criteria	• Start with APT Structural Design Criteria and adapt to ATW needs	0.3	1 year
Nuclear Data for Sodium	• Reduce available data • Measure Bi capture cross section • Measure spallation product yields in Pb and Bi	2.4	5 years
Validated Physics Methods and Codes	• Maintain and enhance MCNPX and Cinder90 • Produce data library for Na up to 150 MeV • Validate codes and models against data	1.2	5 years
Validate target fluid mechanical design	• Perform scaling analysis • Perform scaled non-sodium tests • Perform final sodium test	2.5	4 years
Release of spallation products from LBE (to support source term)	• Perform small scale experiments with doped sodium	0.4	1 year
Validate natural circulation for residual heat removal	• Perform scaled out of beam test. Can be done in conjunction with other out of beam tests.	1.2	2 years
Total		28.26	5 years

5.5 ATW Fuel Technology

Functional criteria for a fuel form to be employed in an ATW system have been considered as follows: 1) the fuel form(s) must incorporate actinides and long-life fission products separated from LWR fuel and be fabricable in a manner that is consistent with desired separations technologies to be employed in the ATW concept, 2) the fuel form must reliably contain fission products during normal and certain off-normal operations; and 3) the fuel form must maintain a coolable geometry and must remain in a predictable location within the ATW blanket. A roadmap to develop a licensable fuel design has been developed along with an estimate of the resources required to complete the roadmap. In broad terms, the activities identified in the roadmap seek key accomplishments with respect to ATW fuel development: 1) identification of candidate fuels for use in all phases of the demonstration, 2) technical qualification of these fuels for their in-core use, and 3) fabrication of these fuels to support their actual use in experiments as well as the demonstration systems.

Two conceptual fuel forms have been proposed as best suited for use in an ATW system. Both concepts make use of cylindrical fuel rods in hexagonal assemblies in a manner similar to that employed in developed fast reactor technologies. However, there is little or no experience with the particular fuel composition selected for this application. The fuel forms proposed here for the ATW concept are believed to be feasible based on experience gained with metallic alloy and dispersion fuel forms; however, it should be realized that much of what is surmised about the expected performance of these fuel forms is speculative.

5.5.1 Candidate Fuel Forms

The primary fuel form is a dispersion fuel comprised of TRU-Zr metallic alloy fuel particles imbedded in a Zr matrix and clad in stainless steel. This form is expected to have good high-burnup irradiation performance, excellent shock resistance in the event of accelerator beam interruption, and offers a potential fabrication route that may not require high-temperature processing, thus enhancing the retention of volatile TRU elements during fabrication. This fuel form may or may not require a liquid metal thermal bond between fuel and cladding. The disadvantage of this fuel form is a fabrication process that may be relatively complex for remote implementation. The secondary fuel form is a TRU-Zr metallic alloy slug and liquid metal thermal bond clad in stainless steel, much like Experimental Breeder Reactor II (EBR-II) fuel or fuel developed for the Integral Fast Reactor (IFR) concept. This fuel form offers a greatly simplified fuel fabrication route that lends itself well to remote hot cell techniques, but requires fabrication at elevated temperatures where retention of volatile TRU elements may be difficult. No material property or irradiation performance experimental data exists for either fuel form.

An overview of the issues associated with these two fuel forms is given in the three next sections. The fabrication issues and envisioned processes are discussed in some detail, because fabrication poses the greatest challenge to the success of the proposed fuel forms. That section is followed by a description of the R&D plan proposed to select, qualify and implement a successful ATW fuel in a demonstration system.

5.5.1.1 Primary Candidate: Dispersion Fuel

The primary fuel choice for ATW assemblies is a dispersion fuel form consisting of 20–40 μm particles composed of ~25% Zr and ~75% TRU (mostly Pu) embedded within a matrix of Zr yielding an overall composition by weight of ~75% Zr and 25% TRU. Several process steps must be developed, and they depend to a large extent on the specific composition and morphology of the feed material, i.e., the product of the fuel recycle processes. These steps are: (1) the alloy for the fuel particles is formed; (2) it is then "powderized" to form the desired small particles; (3) it is blended with the zirconium matrix powder (which is assumed to be another output stream from the electrorefining step and, hence, must also be powderized) and possibly some sort of binder for subsequent steps; (4) pellet compacts are formed from the blended powders, either hot-pressed or sintered after pelletization; (5) they are inspected (those that pass proceed and those that don't are crushed and fed back into the blending step); (6) they are inserted (perhaps along with a bond material) into cladding tubes; (7) the elements are sealed; (8) the fuel elements are inspected (with recycle of those that fail); and finally (9) the elements are combined into assemblies for return to the reactor.

Alloy Formation

If the ATW fuel recycling process can be adjusted to provide a product of the correct composition of Zr and TRU, then further melting to form an alloy will not be necessary. Thus there is probably a considerable incentive to investigate this possibility and to go to some lengths to incorporate it into the electrorefining step. If this is not possible, then a furnace will need to be provided to melt the TRU (Pu melts at 913 K). After the Pu is molten, Zr powder is added to produce the alloy while raising the temperature enough to keep the mixture molten but not to the point at which the vapor pressure of Am, the most volatile of the higher actinides, becomes appreciable. Furnaces of the type needed for this step have been developed and used in the IFR and Spent Fuel Treatment Demonstration (SFTD) programs at Argonne National Laboratory at a scale somewhat smaller than necessary for ATW application. However, scale-up should pose no particular problems, and the engineering details of the mechanism for adding the Zr that will need to be developed should also be tractable.

Particle Formation

Metallic product from the ATW feed material and fuel recycle processes should be amenable to crushing, e.g., with a ball-mill, to form the appropriate particles. Otherwise some other means will be necessary. Grinding of the ingot from the alloying furnace is one possibility, and crushing with an ore-crusher-type machine may be possible if the ingot is sufficiently brittle. A more desirable means for forming particles is probably a hydriding and de-hydriding process. Some metals, e.g., Pu, form hydride in an H_2 atmosphere at relatively low temperatures and the product spalls off as the process proceeds, thus exposing fresh metal so that the reaction proceeds to completion and the product is very friable. The alloy discussed here may behave in that manner, although the presence of the Zr may hinder the spallation. In the latter case, however, the dissolution of H in Zr and the embrittlement of Zr that results should make the product much more easily crushed than the solid alloy ingot. In either event, the hydrided product would then be crushed to produce metal hydride particles of the desired size. This material would then be reheated in vacuum to drive off the hydrogen and leave a particulate metal mass, perhaps requiring some additional crushing to yield an unconsolidated powder.

Again, furnaces of the type necessary for these operations have been developed and operated as part of the IFR and SFTD programs, although not with a hydrogen atmosphere. The latter

adaptation should be straightforward, however, and present no serious problems. A similar process line would be used to prepare the Zr product stream from the ER to form the matrix part of the fuel mixture. The potentially large number of steps in this process would require considerable hot-cell space, however. Because this equipment will process particulate material of small size, special attention must be given to dust control. Moreover, the metallic powders would probably be pyrophoric, affecting the safety case for the facility.

Blending

Any number of commercially available blenders can be adapted for blending the two fuel components and, if necessary, a binder. A heated V-mixer has been developed for remote mixing of electrorefiner salt, zeolite, and glass frit for preparing a ceramic waste form in the SFTD Program at temperatures of 775 K. The unit for the present application would be simpler, since heating should not be necessary.

Pelletization

Assuming a 10-mm-dia × 15-mm-long pellet size, this operation will need to produce on the order of 10,000 pellets per day. This could well be the most challenging part of the fuel fabrication process, since there is very little experience with hot-cell operation of high rate automated processes. However, there is a great deal of experience in high-volume production of sintered oxide pellets for commercial nuclear fuel that may be applicable to the present situation. Formation of green pellets in one machine with subsequent sintering in a large furnace may be preferable to hot-pressing individual pellets. A detailed trade-off study of the various options is beyond the scope of the present report. It seems reasonable to conclude, however, even without a detailed study, that the engineering details required to implement a remotely-operated pelletizing operation can successfully be worked out.

Pellet Inspection

Qualification of the process steps up through pelletization should ensure a very uniform product so that statistical quality control procedures may be used, requiring only a relatively small number of pellets to be inspected to assure that the processes are operating within their anticipated limits. This inspection should require little, if any, automation, using standard weighing and measuring techniques. However, if inspection of each pellet is required, development of an automated system will be necessary. Small pick-and-place robots should easily be able to handle movement of the large number of pellets from optical measuring devices to electronic balances, etc. Although an automated line such as this will require considerable attention to engineering details, its development again should be straightforward.

Element Fabrication

This step is simply the placing of a certain number of pellets inside a cladding tube, and could, indeed, be the last action performed by the robot used for the inspection step. If a metallic bond (e.g., Na) is used, a slug of the bond metal could also be placed into the cladding tube before the first pellet. No insurmountable problems are apparent in this step. Swaging of the cladding tube to reduce the fuel-cladding gap size, in lieu of a metallic or gas bond, could also be accomplished at this point if that option is taken. This would require additional equipment, however.

Element Sealing

Assuming a 1000-mm-long element size, on the order of 100 elements per day must be fabricated and sealed. An automated back-fill (if a gas bond or gas tag is used), end-cap placement, and seal-welding system will be needed to fabricate this number of elements. Such a system was developed and qualified, but not installed and operated, as part of the IFR Program. Adaptation to the ATW

application should be straightforward. If a metallic bond is used, equipment will be needed to melt the bond metal and distribute it throughout the element. This equipment was also developed in the IFR program.

Element Inspection

Inspection for length, diameter, bow, and weight can be performed on an automated system similar to that developed in the IFR Program. Again, adaptation of those designs is straightforward.

Fuel Assembly Fabrication

The production rates envisioned will require output of a few assemblies per day, depending upon the number of elements in an assembly. Fuel assembly fabrication can probably be a manual operation using appropriate jigs and assembly machines, in accordance with experience and practice since the early days of the ANL EBR-II metallic fuel recycle demonstration in the 1960's.

5.5.1.2 Backup Candidate: Cast Fuel

A backup means of fuel fabrication should be carried along in the ATW program should some characteristic of the primary fuel choice or its fabrication prevent its deployment. Currently this backup method consists of injection casting of long metallic slugs in a manner similar to that employed for fabricating EBR-II fuel since the 1960's and was intended, more recently, for IFR fuel. This fuel would have the disadvantage of not being as thermal-shock resistant as dispersion fuel and will require processing at temperatures that are expected to volatilize significant amounts of Am and perhaps some other actinides. In many other respects, it is a simpler process, involving the following steps: (1) metallic products from the ATW fuel feed or recycle processes are combined, melted, and cast into molds; (2) the molds are removed from the metal slugs; (3) the slugs are then cleaned and inspected (with rejects going back into the alloying/casting step); (4) the slugs are inserted into cladding tubes (with metallic bond material, if appropriate); (5) the elements are sealed (with back-fill gas if appropriate); (6) the elements are inspected; and finally (7) the elements are combined into assemblies.

Alloy Formation And Casting

TRU material and Zr from the ATW fuel feed or recycle processes are placed in a crucible in quantities appropriate for the desired fuel alloy composition and melted in an evacuated furnace. After the batch is fully molten a pallet of molds is lowered into the melt and the furnace is pressurized to inject the molten fuel material into the molds, which quickly cool to solidify as slugs inside the mold cavities. After the furnace is opened the molds are transferred to another station for further processing.

This process has been used for many years to fabricate EBR-II fuel and was further developed during the IFR Program. Previously deployed capacity of ~150 molds producing ~6-mm-dia × 500-mm-long slugs would need to be scaled up by a factor of 2–4 to meet the requirements for ATW application, and handling would have to be streamlined somewhat to allow daily operation. This may present an engineering challenge but not an intractable one. A more serious concern is the loss of Am, and possibly other actinides, at the high temperatures needed to melt the ~75% Zr alloy. This means that an ancillary subsystem must be incorporated in the casting furnace gas-

handling system to trap this material so that it can be returned to the fabrication process. Another concern is mold technology. For fabrication for EBR-II and for the IFR program, precision-bore quartz tubes were employed as molds with a Y_2O_3 wash on the bore to serve as a mold release; removal was accomplished by shattering the quartz and discarding the shards as waste. This will probably be unacceptable for ATW application because the casting temperature may be too high for quartz, and the amount of process waste generated too great. At the close of the IFR program, a re-useable mold development activity was underway, but no conclusions were reached. This effort would undoubtedly have to be carried to fruition should this fuel fabrication option be implemented.

Mold Removal

The technique for mold removal, of course, depends on the type of mold used. Quartz molds can obviously be broken and the slugs reclaimed from the rubble. This was done manually in the EBR-II fuel fabrication process, and a machine was built and qualified, but never used in a production mode, to accomplish this for fabrication of IFR fuel. The several hundred slugs per day needed to be processed for the ATW application would certainly require the automated technique.

Re-useable molds may be adjacent refractory metal plates with multiple half-cylinders machined into each. Again, some kind of mold release may be needed in these grooves. Some ingenuity may be required to devise an automated line to separate these plates and remove the slugs, but again this appears to be a tractable problem. It may be possible to cast directly into Zr tubes, perhaps held in place by plates such as the ones just described, to provide support and a heat sink. These Zr-encased slugs could then be inserted into cladding tubes.

Cleaning and Inspection

Residual mold release and, in the case of quartz molds, any mold fragments will require mechanical removal (e.g., wire brushing). If mold plates, as described above, without Zr tubes are used, machining of mold flash formed in the plate separations would be necessary. This is easily accomplished in automated equipment, and the inspection should be similar to that of the dispersion fuel pellets, except that there would be many fewer of the longer slugs (several hundred rather than many thousands per day.) Individual inspection of each slug, especially for bow and for roundness in the multiple re-useable grooved plate option, may be warranted for this process. Small pick-and-place robots should easily be able to handle movement of the required number of slugs from optical measuring devices to electronic balances, etc. Development of an automated line to accomplish this should be straightforward.

Element Fabrication

This step is simply the placing of fuel slugs into a cladding tube, in much the same manner as envisioned for the dispersion fuel pellets, except that, again, many fewer operations would be required. Therefore, this step is essentially the same as for the dispersion fuel option.

Element Sealing, Inspection And Fuel Assembly Fabrication

These steps are the same as for the dispersion fuel option.

5.5.2 Fuel Fabrication Issues

It is felt that scale-up of processes that have been used successfully in remote operations, by factors of 2–4, is fairly straightforward as is adaptation of automated machinery for remote operations. Discussed here are only the areas where new knowledge is required to verify that processes can be developed to produce the desired products and that these products have the properties expected and perform adequately in the irradiation environment.

For the dispersion fuel candidate, development of the ATW fuel feed and recycle processes to provide products with the correct compositions and morphologies for both the dispersed particle and the dispersion matrix might well have the highest payoff, as this has the potential for elimination of some of the more complex powder formation steps. The potential for cost savings should be obvious. The metallurgical behavior of the TRU-Zr alloy will have to be studied in considerable detail to determine its hydriding/de-hydriding behavior, in particular the morphologies of the final as well as intermediate products. Studies of the sintering/hot-pressing behavior of this material, or that produced more directly from the ATW fuel feed and recycle processes, in the Zr matrix will have to be conducted to determine the conditions under which uniform pellets can be fabricated reproducibly.

The main fabrication development needs for the cast fuel are related to the casting process itself. A means of trapping the volatilized actinides must be developed. This may be straightforward, however. A similarly essential but perhaps more difficult problem, as discussed above, might be the development of re-usable molds for the casting process.

5.5.3 Irradiation Performance Issues

The primary performance criteria for a fuel rod in general are that for all anticipated conditions the fuel should retain its position in the core (or at least tend toward less reactive configurations) and the fuel rod should contain fission products and maintain a coolable geometry. Experience with various fuel forms (and metallic fuel, in particular) has proven the following characteristics to be important to fuel performance and lifetime.

5.5.3.1 Dimensional Stability

The dimensions of the fuel material can change dramatically if the material exhibits considerable irradiation swelling or irradiation growth. Such effects have implications for neutronic performance of the core and can introduce stresses into the cladding that lead to cladding breach.

5.5.3.2 Fission Gas Pressurization

Some fuel types, particularly metallic fuel, release large amounts of fission gas into the fuel rod plenum. Therefore, if the plenum is not adequately sized to accommodate this fission gas content, then pressure-induced stresses can lead to cladding failure, particularly during transient-induced temperature increases.

5.5.3.3 Phase Stability Or Microstructural Evolution

The high-temperature, high-flux environment of the reactor core (which also induces temperature gradients) typically alters the microstructure and/or local composition of a fuel material from its as-fabricated state. Such changes include development of porosity that can vary in morphology

across the radius of a fuel rod, establishment of different phases and redistribution of fuel constituents through the fuel material. Build-up of fission products during irradiation can also contribute to these effects. Ultimately, such effects manifest themselves by degrading thermal conductivity, by introducing local high-power zones in the fuel, by enhancing fuel-cladding chemical interaction, or by possibly affecting gas release or swelling phenomena.

5.5.3.4 Fuel Compatibility With Cladding

For many fuel forms, contact of fuel or fission products with the interior cladding surface leads to interactions that embrittle the cladding or otherwise compromise its ability to withstand stresses. This is particularly true for some metal fuel alloys, in which fuel constituents and fission products interdiffuse with cladding constituents, resulting in thinning of the effective wall of the cladding and incorporation of cladding elements into the fuel. Such interaction also leads to formation of low-melting-temperature phases or compositions in the fuel or cladding or in the interaction zone at the fuel-cladding interface; formation of such zones often has implications for fuel reliability during high-burnup, steady state operation and during certain transients.

Although a great deal of relevant experience exists in general on the irradiation performance of dispersion fuels (from research reactors and space reactor development) and metal alloy fuels (most recently from EBR-II operations and the IFR program), the alloys under consideration for use in the ATW system have not been irradiated. The single greatest uncertainty in the fuels technology subtask is the irradiation performance of the proposed ATW fuels. It must be demonstrated that the proposed fuels perform acceptably under irradiation to the target burnup. Experience shows that fuel performance almost always degrades as Pu-content of the fuel is increased. The proposed fuels employ an actinide alloy that is primarily Pu. Furthermore, the addition of such a large quantity of minor actinides into the fuel alloy has never before been tested. The proposed test program has been designed to gain initial irradiation test data on candidate fuel forms as quickly as possible.

A major complication to the irradiation test program is the lack of an appropriate test reactor in which to conduct the irradiations. No fast flux test reactor is currently operating in the U.S. (although FFTF, if restarted, would be an ideal test facility). Furthermore, no Pb-Bi test loop exists at any irradiation facility in the world.

5.5.4 ATW Fuel Research And Development Plan

A roadmap to develop a licensable fuel design has been developed and is presented here as a conceptual R&D plan. The activities that are envisioned to be part of developing an ATW fuel form to be used in a sodium or LBE-cooled ATW, including the gathering of data to support licensing of a demonstration facility, are discussed here.

5.5.4.1 Project Management and Administration

A full-time technical R&D program manager and associated administrative/clerical support will be required for the duration of the program. For the purpose of the present analysis, it has been assumed that the program will begin in FY2000 and last ~22 years, through completion of the database to support implementation and licensing of an ATW system.

5.5.4.2 Fuel Material Properties

Property Determination For Irradiation Testing

The material properties of the fuel components must be either conservatively estimated or experimentally determined in order to design meaningful irradiation experiments and demonstrate

to the irradiation facility the safety of proposed experiments. Therefore, these studies must be initiated and substantially completed before irradiation testing begins. Although conservative values of some properties can be justifiably estimated, such an approach is often undesirable (or intolerable) due to the constraints placed on irradiation conditions when uncertainties are accounted for in safety analyses.

Metallurgy

A variety of metallurgical studies will be required to determine alloy solidus-liquidus temperatures, phase equilibria, and microstructural characteristics. These properties have both safety (e.g., fuel melting) and irradiation performance (e.g., symmetrical crystal structures are often best for metal alloy dimensional stability during irradiation) implications. These studies must necessarily be conducted using actinides rather than surrogate elements, although presumably much can be done using Pu to represent all of the TRU component elements. Thus, these studies must be performed in a glovebox environment. This effort requires a technical staff member, a technician, and facility support.

Thermophysical Properties

A variety of thermophysical properties must be estimated or measured. These include densities, thermal expansion characteristics, thermal conductivity, specific heat, etc. These properties are required to enable the design of irradiation tests such that the proper thermal and irradiation conditions are achieved in the experiments. Furthermore, the thermal properties of the fuel directly impact fuel integrity, and therefore reactor safety. Many of these studies may be able to make use of surrogate elements substituting for the TRU component. However, acquiring representative data will require the use of Pu in property measurements, necessitating work in a glovebox environment. This effort requires a technical staff member, technician, and facility support.

Compatibility Testing

Compatibility between the fuel and stainless steel cladding must be confirmed. Considerable data exists in this area relative to metallic, Pu-containing EBR-II or IFR fuels and a variety of stainless steel claddings; however, a limited number of confirmatory tests will be needed. Of particular interest will be the class of stainless steel alloys in use by the Russians in LBE applications, and the effect of the minor actinides on compatibility. For the dispersion fuel form, compatibility between the fuel alloy and the Zr matrix must be demonstrated. Additionally, compatibility between the fuel and the LBE must be characterized. Although little data currently exists in this area, the relatively high solubilities of Pu and Zr in Pb and Bi indicate that dissolution of fuel material into coolant after cladding breach is an issue to be addressed. The major issues in this area must be resolved prior to beginning the irradiation test program. Use of Pu and minor actinides necessitates work in a glovebox environment. This effort requires a technical staff member, technician, and facility support.

Bond/Gap Studies

A number of issues related to the fuel-cladding gap must be resolved prior to fabricating the initial fuel for irradiation testing. For the primary dispersion fuel form, it must be determined if fabrication techniques allow for co-extrusion of the fuel and cladding, thus eliminating the need for an open gap and a thermal bond material. Should an open gap and a thermal bond material be

required or desired, a thermal bond material must be selected that is compatible with fuel, cladding and coolant materials; this material will likely be a liquid metal, such as lead-bismuth eutectic (LBE) or Na. This task is primarily a literature review and analysis to be conducted by a technical staff member, with some laboratory work anticipated to resolve open questions.

Property Determination For Safety Analysis And Fuel Qualification

While many material properties may be conservatively estimated for the purposes of obtaining approval for the initial irradiation experiments in test reactors, this will not be adequate to prepare a solid, technical safety case for an ATW core. This is due to the fact that the material properties may be very conservatively estimated for the purposes of irradiation testing, relying on the fact that only a small amount of the test fuel is being introduced into the reactor core, thus limiting the consequences of any failure that may occur to an acceptable level. Such consequences are generally not acceptable when failure is extrapolated to a situation in which the test fuel represents a significant fraction of the core. Thus, the large uncertainties associated with the conservative material property estimates must be reduced by a fairly comprehensive experimental program to measure the important fuel properties directly. The results of this measurement program, which will encompass steady-state and off-normal irradiation conditions, must be available for use in preparation of the DEMO core conversion fuel safety case.

Metallurgy, Thermophysical Properties, Compatibility Tests

The metallurgical studies, thermophysical property measurements, and compatibility testing previously begun to support irradiation testing must be continued and extended to provide a more comprehensive characterization of the fuel system. Experimental measurements should be made with the major TRU elements incorporated into the fuel alloy. These studies must be performed in a glovebox (perhaps shielded) environment. This effort requires a technical staff member, technician, and facility support.

5.5.4.3 Fabrication Studies

Fabrication Development for Irradiation Experiments

It is envisioned that the two metallic fuel forms proposed for the ATW can be fabricated using techniques previously employed for IFR-type fuels or currently under development for metal-matrix, reduced-enrichment fuel for research reactors. However, there is no experience with such fabrication of the fuel compositions being considered for ATW. Therefore, the suitability of the envisioned techniques must be demonstrated in laboratory experiments and appropriate fabrication parameters determined.

It is assumed that existing fuel fabrication facilities will be employed to fabricate the experimental fuels for irradiation testing. Since both candidate fuel forms will be tested initially, the capability to fabricate Pu-bearing dispersion fuels and injection cast rods will need to be identified. It will likely be necessary to install some new fabrication equipment, even at an existing fuel laboratory. Obviously, this task must be completed in time to begin fuel fabrication for the first irradiation experiment. This effort requires design/fabrication engineers and facility support.

Advanced Fabrication

Methods employed in fabricating the experimental fuels for irradiation testing may not lend themselves to a production scale operation, so this separate task will focus on those technologies capable of delivering fuel at production quantities and rates and is not limited to the fabrication of experimental test fuels. Initially, both fuel forms will be considered, but the task will focus on the selected form as irradiation testing data makes selection possible. Issues to be resolved for the dispersion fuel form include fuel alloy fabrication should the electrorefining process yield an alloy that is not directly usable, fuel alloy powderization, and fuel-cladding co-extrusion (if no gap is employed). For the cast alloy fuel form, acceptable high temperature molds must be developed and fabrication methods that minimize the volatilization of the minor actinides (or capture and recycle volatilized species) must be demonstrated. This task must be completed prior to the need to begin fabricating the fuel for the STF blanket conversion. This effort requires design/fabrication engineers, technicians and facility support.

Remote Fuel Fabrication Development

After the final fuel form has been selected and the fabrication methods demonstrated on a laboratory scale, the entirety of the process must be implemented in a remote, hot cell environment. This effort is primarily one of design requiring design/fabrication engineers, with some technician and facility support to allow mock-up testing of concepts. Remote fuel fabrication development must be substantially complete in time to support remote fabrication of prototypic ATW fuel for the STF blanket conversion.

5.5.4.4 Code and Model Development

A fuel performance computer code capable of accurately predicting the behavior of ATW fuel will be needed to support safety analyses of both STF and the demonstration plant. It is further assumed that such an analysis capability will be required by licensing authorities. It is not clear that any existing fuel performance code can easily be adapted for use with the fuel systems proposed for the ATW concept. Evaluation of the potential use of an existing code will be made. Adaptation of the basic structure of an existing code will be performed if possible, or development of a new code structure if not; with the advent of new computer platforms and computational techniques since the development of most of the classical fuel performance codes currently available, there is considerable motivation to construct a new code rather than adapt an existing one. This basic code structure must be available as a test bed for implementation and testing of new irradiation performance models as they are developed. Model development will be pursued as data from irradiation testing becomes available and is analyzed. This task requires the efforts of technical staff members.

5.5.4.5 ATW Fuel Feasibility Irradiation Tests

The first two irradiation tests are designed to collect data relating to the feasibility of both candidate fuel forms. As currently conceived, these irradiations would likely be conducted in the Advanced Test Reactor (ATR), which is viewed as the fastest path to gain initial performance data. Although the ATR is a thermal reactor and cannot provide the fast flux prototypic of an ATW system, it is expected that an experiment can be devised to produce relevant fission rates and temperatures which will allow an initial evaluation of the primary irradiation behavior characteristics such as swelling, fission gas retention/release, fuel constituent migration, and fuel-cladding compatibility. The first irradiation test would attempt to gain data at what are considered to be nominal conditions for an ATW fuel. If data from the first test appears promising, the second

test would attempt to evaluate more aggressive, bounding-type irradiation conditions; otherwise the second irradiation test would be used to further explore fuel form options at nominal conditions. Both candidate fuel forms would be tested.

Irradiation Pre-Test Design and Analysis

Irradiation test vehicle(s) must be designed and reactor test position(s) selected to produce desired power levels and temperatures in the test fuels. Irradiation is expected to occur in a drop-in capsule position. Thus, appropriate thermal conditions in the fuel must be achieved by means of thermal resistances (gas gaps) incorporated into the capsule design. Data relating to cladding performance (creep, compatibility with coolant) require a fast flux environment and cannot be obtained in these tests. This effort requires technical staff members and design engineers.

Fuel Rodlet Fabrication

It is assumed that the initial irradiation testing will be conducted on short fuel "rodlets" of 4 to 6 inches in length. Multiple rodlets will be incorporated into the irradiation test vehicle. Rodlets of both candidate fuel forms must be fabricated to QA standards acceptable to the irradiation facility. This task requires a technical staff member, technicians and facility support.

Irradiation

It is expected that the ATW fuel discharge burnups can be achieved in the ATR in irradiations of no more than one year. Funds must be provided for irradiation services.

Post-Irradiation Examination

Considerable postirradiation examinations (PIE) should be conducted on these test fuels following discharge from the reactor. Characterization of fuel swelling, fission gas release, fuel alloy constituent migration and fuel-cladding interaction will be necessary. Fuel burnup measurements will be required. This task requires a technical staff member, technicians and facility support at a hot cell facility equipped to conduct these forms of PIE.

Post-Test Analyses

Evaluation of the PIE data for use in subsequent irradiation test planning and model development will require technical staff members.

5.5.4.6 Integral Fuel Rod Tests in Fast Flux

Demonstration of integral fuel rod performance requires tests of prototypically-sized fuel rods irradiated in a fast flux. These irradiations would have to be performed in a foreign facility like BOR-60 (Russia), unless FFTF were available. Scale up to fabricate prototypically-sized rods and establishment of the interfaces required with a foreign irradiation facility will take some time. Thus, these tests are viewed as coming subsequent to the feasibility testing proposed for the ATR. The fast flux environment will provide conditions more prototypic of the ATW system for fuel testing. Cladding performance such as swelling and creep, which could not be adequately evaluated in the ATR's thermal flux, will be important in these tests. Fuel-cladding chemical and mechanical interactions will be characterized. Also, the fuel performance issues evaluated initially in the ATR tests will be characterized in a fast flux environment. It is conceivable that fuel alloy swelling and gas release behavior could be different in a fast flux. It is anticipated that a Pb-Bi loop will not be available at this time for testing; testing in a Na coolant is expected. Thus,

cladding-coolant compatibility issues under prototypic conditions cannot be demonstrated in these tests. As with the fuel feasibility irradiation tests, the first integral fuel rod test will target nominal ATW conditions of power and temperature, with the second test aimed at more aggressive, bounding conditions. It is anticipated that both candidate fuel forms will be carried through the first integral fuel rod test, with the decision being made to focus only on the most promising fuel form only in subsequent tests.

Irradiation Pre-Test Design And Analysis

Irradiation test vehicle(s) must be designed and reactor position(s) selected to produce desired power levels and temperatures in the test fuels. Irradiation is expected to occur in a standard driver fuel position. Appropriate thermal conditions would be achieved by coolant flow orificing. This effort requires technical staff members and design engineers.

Test Fuel Fabrication

It would be unlikely that an entire assembly of experimental fuel will be fabricated and tested; rather, a small number of experimental fuel rods would be incorporated into a standard driver assembly. Fuel rods of both candidate fuel forms (for first test) must be fabricated to QA standards acceptable to the irradiation facility. This task requires a technical staff member, technicians and facility support.

Irradiation

It is expected that the ATW fuel discharge burnups can be achieved in the BOR-60 in irradiations of no more than one year. Funds must be provided for irradiation services.

Postirradiation Examination

Considerable PIE should be conducted on these test fuels following discharge from the reactor. If irradiation occurs in BOR-60, a Russian hot cell facility would be used. Characterization of fuel swelling, fission gas release, fuel alloy constituent migration, fuel-cladding interaction and cladding dimensional changes will be necessary. Fuel burnup measurements will be required. This task requires a technical staff member, technicians and facility support at a hot cell facility equipped to conduct these forms of PIE.

Post-Test Analyses

Evaluation of the PIE data for use in subsequent irradiation test planning and model development will require technical staff members.

5.5.4.7 Integral Fuel Rod Tests in Pb-Bi

Demonstration of integral fuel rod behavior in LBE coolant requires irradiation tests of prototypically-sized fuel rods in a Pb-Bi test loop. No such loop currently exists at any test reactor facility and would have to be designed, constructed and installed. This could be done easily at the ATR, which is designed to accommodate the use of such test loops; installation of a Pb-Bi test loop would be considered at BOR-60 (and FFTF, if available), although it is assumed that loop installation in BOR-60 would be considerably more difficult than the ATR. Cladding-coolant

compatibility issues in a radiation environment would be assessed in these tests. As with the other irradiation test sequences, the first test will target nominal ATW conditions, with the second test aimed at more aggressive, bounding conditions.

Loop Conceptual Design

The candidate irradiation facilities at which a Pb-Bi loop might be installed must be surveyed and evaluated. ATR, BOR-60 and FFTF (if available), will be considered at a minimum. This effort will require a technical staff member, a number of design engineers and support from the irradiation facilities under consideration.

Loop Final Design, Fabrication and Installation

After the irradiation facility has been selected, the detailed design of a Pb-Bi loop must be developed and approved, along with its fabrication, installation and qualification at the reactor. This effort will require a team of design engineers and support from the irradiation facility selected for use.

Irradiation Pre-Test Design and Analysis

Irradiation test vehicle(s) for use in the Pb-Bi loop must be designed. This effort requires technical staff members and design engineers.

Test Fuel Fabrication

A small number of full-size rods, possibly comprising a mini-assembly, will be fabricated for testing in the Pb-Bi loop facility. Fuel rods must be fabricated to QA standards acceptable to the irradiation facility. This task requires a technical staff member, technicians and facility support.

Irradiation

It is expected that the ATW fuel discharge burnups (or cladding exposures) can be achieved in the ATR or BOR-60 in irradiations of no more than one year. Funds must be provided for irradiation services.

Postirradiation Examination

If the Pb-Bi loop is installed in a fast reactor, considerable PIE should be conducted on these test fuels following discharge from the reactor to obtain irradiation data on all aspects of fuel performance. If irradiation occurs in BOR-60, a Russian hot cell facility would be used. If the loop is installed in the ATR, PIE would be more limited, concentrating on cladding-coolant compatibility. This task requires a technical staff member, technicians and facility support at a hot cell facility equipped to conduct these forms of PIE.

Post-Test Analyses

Evaluation of the PIE data for use in subsequent irradiation test planning and model development will require technical staff members.

5.5.4.8 Tc and I (LLFP) Target Development

Motivation exists to transmute Tc and I (long-lived) fission products in addition to plutonium and the minor actinides. These fission products will be extracted from spent fuel to be introduced into an ATW system for destruction.

LLFP Conceptual Development

A decision must be made as to the form in which Tc and I will be introduced into an ATW transmutation blanket. While it may be possible to incorporate these elements into the ATW fuel proper, there are also advantages to their irradiation and transmutation separately in target elements. Depending on the pyroprocess scheme to be implemented, these fission products may emerge from the separations process as distinct streams and must be reintroduced into the fuel alloy system if desired to be integral with the fuel. If separate transmutation targets are to be used, the form of those targets must be determined. Alternatively, it is also possible that Tc fission products in recycled ATW fuel will remain with matrix Zr, to be incorporated in the fuel as a dilute alloy in the matrix Zr. It has been initially proposed that the Tc extracted from the incoming LWR fuel be irradiated as part of a noble metal alloy that should be able to perform well under irradiation to high exposure, which would be desired if a once-through deep-burn concept were pursued; otherwise, a Tc recycle concept would need to be developed. Iodine could be irradiated as a salt powder (e.g., NaI), providing for the produced Xe gas to be readily released to a pressure plenum included in the target element. Such a concept was considered for production of Xe-128 at EBR-II and appears feasible.

LLFP Target Development and Design

If the conceptual development task determines that transmutation of Tc and I via separate target elements is preferred (as is expected), these targets must be designed for subsequent irradiation testing. This effort requires a technical staff member, a design engineer, and some laboratory effort for bench-scale experiments to determine feasibility of concepts.

LLFP Target Recycle Development

It is unlikely that the Tc and I transmutation targets will be capable of achieving final burnup by irradiation in a “once-through” cycle due to physical and/or mechanical degradation of the target elements in the case of Tc and gas production in the case of I. Thus, recycle of the long-lived fission product transmutation targets will need to be investigated. Recasting of the noble metal target and fabrication into new target hardware may be required in the case of the Tc. If the iodine is irradiated as NaI, then perhaps additional fission product iodine will need to be reacted with the free Na. These recycle processes might be fairly simple. Alternatively, a more complex separation process may be required. This effort requires a fraction of multiple technical staff members and some laboratory effort for bench-scale experiments to determine feasibility of concepts.

Tc and I (LLFP) Target Testing In Fast Flux

Assuming the decision is made to transmute Tc and I fission products by means of special target assemblies, these target designs must be tested and demonstrated. These tests should demonstrate target neutronic (transmutation) performance as well as acceptable physical behavior; thus, they should ideally be performed in a fast reactor such as BOR-60, unless FFTF is available. The physical performance issues to be resolved/demonstrated are primarily associated with dimensional stability under irradiation of the noble metal alloy for the Tc target and accommodation of gas generation for the I target. The impact of changes in composition due to buildup of transmutation and capture products will also be assessed,

Irradiation Pre-Test Design and Analysis

Irradiation test vehicle(s) for use in the Tc and I target irradiations must be designed. This effort requires technical staff members and design engineers.

Test Target Fabrication

Test Tc and I targets must be fabricated. Assuming Tc and actual fission product I are used to demonstrate transmutation, hot cell fabrication will be required. For the case of I, stable naturally occurring isotopes might be able to be used if demonstration of transmutation is not an issue and gas production from the stable isotopes is adequate. This task requires a technical staff member, technicians and facility support.

Irradiation

Irradiation testing of the Tc and I targets in a fast reactor such as BOR-60 could take considerable time depending upon transmutation rates and the decision on whether the targets will function in a once-through deep-burn operation or be recycled. Up to three years irradiation time has been estimated, funds must be provided for irradiation services.

Interim and Final Postirradiation Examination

PIE will be conducted on selected Tc and I target capsules at interim exposures as well as at final discharge. Dilation of target cladding or capsules will be measured, swelling of the Tc target noble metal alloy will be characterized, and gas pressurization of the I target capsules will be measured. If irradiation occurs in the BOR-60, an hot cell facility in Russia would be used. This task requires a technical staff member, technicians and facility support at a hot cell facility equipped to conduct these forms of PIE.

Post-Test Analyses

The PIE results will be evaluated for use in subsequent irradiation test planning and final target design and analysis. These efforts will require technical staff members.

5.5.4.9 ATW Fuel Transient Performance Evaluation

The transient performance of ATW fuels must be understood to evaluate their behavior during off-normal or accident conditions. Reactivity transients, loss-of-flow or loss-of-heat-sink events generally involve the heating of fuel and cladding materials to temperatures significantly above those reached during normal operation, and may lead to cladding breach. Experience with metallic EBR-II and IFR fuels during these types of off-normal events showed that metallic alloys performed acceptably. It is anticipated that ATW fuels will also behave acceptably, but this must be confirmed.

Some safety issues that will require investigation are evident. For example, a fuel failure during ATW operations will likely expose the TRU-Zr alloy to LBE. The binary phase diagrams for Bi-Zr and Bi-Pu indicates that these binary systems have some mutual solubility; this implies that if fuel cladding is breached, fuel material may dissolve into Pb-Bi coolant or the fuel alloy may

dissolve coolant metal, either of which could exacerbate the consequences of a fuel failure. Although the consequences of such exposure will depend on phenomena in addition to fuel-LBE compatibility, the intrinsic fuel-LBE compatibility is an important characteristic to understand. Likewise, if sodium is used as a thermal bond in the fuel-cladding gap, the implications of Na-LBE interactions on the ATW system must also be understood.

The failure mechanisms, thresholds and consequences of ATW fuel must be identified and operational limits established. This can be accomplished through a combination of out-of-pile and in-pile testing. The results of these tests are needed for the preparation of the fuel safety case for the demonstration plant, and may be needed to some extent in evaluating the safety of the STF blanket conversion.

Out-Of-Pile Testing

Much can be learned regarding transient fuel behavior from out-of-pile tests. Furnace testing can simulate the over-temperature effects, allowing the study of accelerated fuel-cladding chemical interaction and cladding creep; dissolution rates of the fuel alloy in LBE can likewise be quantified. Actual irradiated fuel is needed even for these out-of-pile tests, however, for a variety of reasons (e.g., retained fission gas pressure drives many processes, fuel-cladding compatibility is affected by the presence of fission products, cladding strength is a function of fluence, etc.). This necessitates the availability of previously irradiated fuels. This task requires technical staff members, technicians and facility support at a specially equipped hot cell facility.

In-Pile Testing

Although much can be learned from out-of-pile testing, some aspects of transient fuel performance require reactor testing. The radial temperature gradient that exists in a fuel rod at power, and the resulting differential radial gas pressure within the fuel, drives potential failure mechanisms that cannot be simulated in furnace tests. Fuel-cladding mechanical interaction, which is an important potential failure mechanism, likewise depends upon temperature differences that can only be produced in a reactor environment. Over-power and run-beyond-cladding-breach tests could be conducted at steady-state reactor facilities, although testing in a LBE loop would be required; alternatively, it might be possible to use STF for some of this testing. Some testing in a transient reactor such as the Transient Reactor Test Facility (TREAT) at ANL would also likely be required. In this case, it may be possible to conduct the transient tests, which would be geared toward establishing cladding failure thresholds or assessing the consequences fuel melting without prior cladding breach, in sodium loops that already exist at TREAT. This task requires technical staff members, technicians and facility support at reactor test facilities.

5.5.4.10 Target Test Facility Fuel Fabrication

Planning

Because a major objective of TTF operation is to demonstrate the coupling of fissile material neutron multiplication to a spallation source, a tenable plan for incorporating fissile fuel into the TTF blanket must be developed. The planning effort will require technical staff member(s) having experience with the fuel form selected to be the DEMO start-up driver core.

Fuel Design

The TTF will operate with a fissile blanket having a well known and demonstrated fuel, since ATW fuel will not have been fully qualified in time to serve as the blanket fuel. It is proposed that this driver fuel be injection-cast U-10Zr, clad in a stainless steel alloy that has been demonstrated as being compatible with the LBE coolant. This fuel system is well understood, having served as the EBR-II driver fuel for many years, and easy to fabricate. The TTF blanket fuel design will require a technical staff member and design engineer.

Safety Case

The safety case for a U-10Zr driver fuel should be easily made using, for the most part, existing data. Experience can be drawn from the EBR-II safety case for a U-10Zr driver core that was used successfully for many years. Analytical tools and mechanistic models developed by ANL should be directly applicable. The only significant difference will be issues associated with LBE compatibility. This effort will require technical staff members familiar with U-10Zr metallic fuel performance.

Fuel Fabrication

It is assumed that the fabrication of the U-10Zr driver fuel for the TTF blanket would be performed in an existing facility such as the Fuel Manufacturing Facility at ANL. This facility was used to fabricate the EBR-II driver fuel and is currently still available. Much of the fabrication process can be performed in ventilation hoods, with the incorporation of the Na bond in a glovebox. This effort will require a fabrication engineer, technicians and facility support.

Fuel Performance Evaluation

A fuels engineer will be required to monitor and evaluate the performance of the fissile blanket.

5.5.4.11 ATW Demo Facility Fuel Fabrication and Testing

Planning

The major objective of DEMO operation is to demonstrate the integral performance of the accelerator-target-blanket system. The DEMO blanket must started up with a qualified driver fuel. The planning effort will require technical staff member(s) having experience with the fuel form selected to be the DEMO start-up driver core.

Startup Core

The ATW DEMO will start up with a first core having a well known and demonstrated fuel, since ATW fuel will not have been fully qualified in time to serve as the driver fuel. It is proposed that this driver fuel be injection-cast U-10Zr, clad in a stainless steel alloy that has been demonstrated as being compatible with the LBE coolant. This fuel system is well understood, having served as the EBR-II driver fuel for many years, and easy to fabricate.

Fuel Design

The ATW Demo start-up core driver fuel design will require a technical staff member and design engineer.

Safety Case

The safety case for a U-10Zr driver fuel should be easily made using, for the most part, existing data. Experience can be drawn from the EBR-II safety case for a U-10Zr driver core that was used successfully for many years. Analytical tools and mechanistic models developed by ANL should be directly applicable. The only significant difference will be issues associated with LBE compatibility. This effort will require technical staff members familiar with U-10Zr metallic fuel performance.

Fuel Fabrication

It is assumed that the fabrication of the U-10Zr driver fuel for the DEMO start-up core would be performed in an existing facility such as the Fuel Manufacturing Facility at ANL. This facility was used to fabricate the EBR-II driver fuel and is currently still available. Much of the fabrication process can be performed in ventilation hoods, with the incorporation of the Na bond in a glovebox. This effort will require a fabrication engineer, technicians and facility support.

Fuel Qualification

Qualification of the ATW DEMO driver core will be accomplished by the irradiation and periodic examination of a group of lead assemblies (or rods). In-core surveillance and post-irradiation examination of those rods will demonstrate that fuel performance is within the bounds defined by the STF safety analysis.

Lead Assembly Irradiation

Fuel qualification for the DEMO start-up core will consist of at most a few lead assemblies designed to operate at powers (burnup rate) and temperatures somewhat higher than the peak driver fuel conditions.

Interim and Post-Test Examination and Analysis

Ideally, selected fuel rods from the lead assemblies will be removed periodically for examination in a hot cell facility. Acceptable performance observed in these rods should bound the performance of the entire core. This effort will require a technical staff member, technicians and facility support at a hot cell equipped to perform PIE.

Blanket Conversion

After DEMO start-up, testing and reliable operation has been demonstrated using U-10Zr driver fuel, a blanket conversion process will begin. In this process, assemblies (or rods) of ATW fuel will be introduced a few at a time. This will allow neutronic data to be collected with actual ATW fuel.

Design and Analysis

Based on the data generated in the irradiation test program, an ATW fuel will be designed for implementation in the DEMO. This effort will require technical staff members and design engineers.

Safety Case

A safety case must be made for the ATW fuel assemblies that are introduced into the DEMO. This safety case will draw upon the data produced in the material property measurement and irradiation test programs and make use of the analytical tools, models and codes previously developed. This effort will require a number of technical staff members.

Fuel Fabrication

It is assumed that the ATW fuel to be used in the DEMO blanket conversion would come from the actual processing of LWR spent fuel in the LWR Demo Plant (LWRD). The fuel would be needed approximately one and a half years after the LWRD begins actual operations. Alternatively, if the LWRD is not capable of producing fuel assemblies at this time, the conversion core fuel would be fabricated using a Pu-Zr alloy in a "cold" (i.e., glovebox-contained rather than hot cell-contained) process, rather than the TRU-Zr alloy. The use of reactor or fuels grade Pu, having high contents of both Pu-241 and Am, would be a reasonable simulation of the TRU-Zr alloy. The cost estimate prepared for the fuel technology portion of this roadmap reflects the facility engineers and operators required to fabricate the fuel using feed obtained from LWR spent fuel, but does not include facility construction costs.

Fuel Qualification

As with the start-up core, qualification of the ATW driver fuel blanket conversion will be accomplished by the irradiation and periodic examination of a group of lead assemblies (or rods). In-core surveillance and post-irradiation examination of those rods will demonstrate that fuel performance is within the bounds defined by the ATW Demo safety analysis.

Lead Assembly Irradiation

Fuel qualification for the ATW driver fuel blanket conversion will consist of a few lead assemblies designed to operate at powers (burnup rate) and temperatures somewhat higher than the peak driver fuel conditions.

Interim and Post-Test Examination and Analysis

Ideally, selected fuel rods from the lead assemblies will be removed periodically for examination in a hot cell facility. Acceptable performance observed in these rods should bound the performance of the entire ATW fuel component. This effort will require a technical staff member, technicians and facility support at a hot cell equipped to perform PIE.

5.5.4.12 ATW Fuel Testing at Demonstration Plant with Increased Power

Planning

An objective of the operation of the Demo plant will be the demonstration of safe and reliable operations of the target and blanket system. One aspect of that demonstration will be the attaining of a considerable fuel performance database to support the licensing of a prototype ATW system (and subsequent ATW plants). The strategy for supporting the startup of the Demo plant and for attaining the fuel performance database depend on the safety case used to support the licensing of the Demo plant itself. For example, if a sufficient database regarding the performance of prototypic ATW fuel in LBE coolant cannot be established prior to Demo plant startup, then the Demo startup will be accomplished with a loading of fuel for which there is already an ample database, such U-10Zr. Addressing issues such as these with the intent of supporting Demo operation and meeting its objectives will require some amount of planning. The planning effort will require technical staff member(s) having experience with the fuel form selected to be the ATW Demo start-up driver core.

ATW Fuel Experiments

A short period of time after the ATW Demo has begun operations using the TRU-Zr start-up driver core, experimental fuel assemblies using recycled ATW fuel will be introduced. These experiments would probably not have to be extensive in number or duration, serving primarily to confirm expected behavior in recycled fuel. They would naturally lead into a conversion of the blanket to the ATW recycled fuel form. The fuels for these experiments would be fabricated at the Pyro Fuel Demo Plant (PFD), which would become operational approximately one and a half years before the test fuels would be required. This effort will require a technical staff member, technicians and facility support at a hot cell equipped to perform PIE.

Blanket Conversion

After ATW Demo start-up, testing and reliable operation has been demonstrated using TRU-Zr driver fuel, a blanket conversion process will begin. In this process, assemblies of ATW recycled fuel will be introduced into the core, displacing TRU-Zr driver assemblies.

Planning and Analysis

Based on the data generated in the irradiation test program, initial DEMO operations and the ATW fuel experiments, the ATW fuel design will be finalized. This effort will require technical staff members and design engineers.

Safety Case

A safety case must be made for the ATW fuel assemblies that will make up the converted core. This safety case will draw upon the data produced in the material property measurement and irradiation test programs, initial DEMO operational experience, and will make use of the analytical tools, models and codes previously developed. This effort will require a number of technical staff members.

Fuel Fabrication

It is assumed that the ATW fuel to be used in the ATW Demo blanket conversion would come from the actual processing of ATW spent fuel in the Pyro Fuel Demo Plant (PFD). The costs estimated in the attached chart reflect the facility engineers and operators required to fabricate the

fuel, but do not include costs of operating the separations portion of the facility or facility construction costs.

Fuel Qualification

As with the start-up core, qualification of the ATW driver fuel blanket conversion will be accomplished by the irradiation and periodic examination of a group of lead assemblies. In-core surveillance and post-irradiation examination of those assemblies will demonstrate that fuel performance is within the bounds defined by the Demo plant safety analysis.

Lead Assembly Irradiation

Fuel qualification for the ATW driver fuel blanket conversion will consist of a number of lead assemblies designed to operate at powers (burnup rate) and temperatures somewhat higher than the peak driver fuel conditions.

Interim and Post-Test Examination and Analysis

Selected fuel rods from the lead assemblies will be removed periodically for examination in a hot cell facility. Acceptable performance observed in these rods should bound the performance of the entire ATW driver fuel core. This effort will require a technical staff member, technicians and facility support at a hot cell equipped to perform PIE.

5.5.5 Differences in Fuel Development Program for a Sodium-Cooled ATW System

The fuel development tasks envisioned to develop and validate an ATW fuel for use in an LBE coolant are comprised of a series of irradiations in thermal and fast-spectrum test reactors, followed by integral fuel rod tests in a LBE loop. Such a loop would have to be designed, constructed and implemented in an existing test reactor. Should the decision be made to employ sodium coolant in the ATW system, such an irradiation test loop would be unnecessary. The initial fuel feasibility testing would still likely be performed in a thermal test reactor such as the ATR in order to quickly and inexpensively gain irradiation data on a variety of fuel types and forms. This would be followed by full size rod tests in an existing sodium-cooled fast reactor (e.g., BOR-60). Thus, the fuel development tasks and costs would be reduced if sodium were selected to be the ATW coolant over LBE.

The fuel development and validation issues to be addressed by a research and development program would be reduced if the ATW system employed a sodium coolant rather than LBE. Sodium is compatible with a wide variety of stainless steels that could be selected for use as cladding, as well as with the major fuel constituents such as plutonium and zirconium. Such compatibility must be demonstrated with LBE if it is to be used as a coolant. The major alloying elements in the reference ATW fuel form, zirconium and plutonium, are known to be compatible with sodium, but appear to have some solubility in lead and bismuth. This suggests that fuel dissolution in LBE coolant is a possibility that must be investigated, whereas dissolution in sodium is not a problem.

5.5.6 Estimated Cost and Schedule

Schedules and resources required for the Fuel Development tasks described above were estimated for the purposes of estimating the sequenced cost of developing a sodium-cooled and LBE-cooled ATW target and blanket system. The projected cost and schedule information developed in this effort, for the deployment-driven scenario assumed, are summarized in the tables contained in Attachments 8 and 9. Although year-by-year allocation is not shown in the tables, a linear distribution of the total effort and costs reflected in the tables was assumed (for simplicity); those assumptions provide the basis for the year-by-year summary of costs presented in Section 6. An average, fully-burdened effort rate of $220k per full-time equivalent year was assumed.

5.5.7 Proposed ATW Irradiation Test Program Using FFTF

If FFTF becomes available to support the ATW fuel development effort as it is currently envisioned, then the following irradiation testing program would be proposed. Because sodium coolant will be considered as a back-up option to LBE coolant, issues associated with fuel performance in either coolant must be considered. However, because the stainless steels most recently employed as cladding materials have excellent compatibility with sodium, the main issues

regarding the selected coolant will be associated with compatibility of fuel and cladding with LBE. These LBE-specific issues will be discussed at the end of the section.

5.5.7.1 Fuel Feasibility Test #1 in ATR (or FFTF)

Because there is no experience with the fuel forms currently proposed for the ATW system, an early irradiation test is required to determine if the principles that form the rationale for these forms are valid. This is most efficiently accomplished through a simple, low-cost irradiation test of small samples. Such tests have been used most recently to investigate the performance of Al-matrix, U-Mo dispersion fuels for the RERTR program, and similar tests were employed historically in research reactors such as the CP-5 to learn about irradiation effects in many fuel forms.

It is proposed that various fuel samples developed early in the program, as part of the effort to develop fabrication methods, would be irradiated in a small experiment in the Advanced Test Reactor (ATR) located at the Idaho National Engineering and Environmental Laboratory (INEEL). This irradiation would likely begin during the third year of the program. Although the ATR cannot offer a fast flux testing environment that is prototypic of the envisioned ATW blanket, it is believed that prototypic fission rates and fuel temperatures can be obtained in a properly-designed ATR experiment; although some radiation effects of interest will not be prototypically active in the ATR spectrum, the major feasibility issues can still be investigated. The perceived advantage to performing this early experiment in the ATR, rather than in the FFTF, is the low cost and quick turnaround of a simple ATR experiment. However, the possibility and desirability of performing such an experiment in the FFTF will be thoroughly considered.

5.5.7.2 Fuel Feasibility Test #2 in FFTF

A follow-on test of feasibility is proposed for insertion into the FFTF during the fourth or fifth year of the program. This experiment is envisioned to be an open-core experiment in an FFTF subassembly, containing a good proportion of mixed-oxide or U-10Zr “filler” fuel rods among the experimental capsules. Fuel Feasibility Test #2 will use small- samples, as proposed for test #1; however, the behavior noted in well-performing samples of Test #1 will be confirmed in the fast spectrum of FFTF in Test #2. The placement of fuel samples in the subassembly and careful restriction of flow around particular capsules will be used to provide conditions required for individual fuel capsules.

It is expected that Fuel Feasibility Test #2 will require about one to one-and-a-half years of irradiation time before removal for post-irradiation examination. Post-irradiation examination will be performed at a hot cell facility to be determined later.

5.5.7.3 Integral Rod Test #1 in FFTF

If the results of the two fuel feasibility tests indicate that one or more of the fuel forms considered show promise, then integral rods of these fuel forms will be prepared for an open-core subassembly irradiation in the FFTF. This experiment would likely begin irradiation in the fifth or sixth year of the program. The subassembly would likely be composed of various experimental fuel rods of the same dimension as FFTF driver rods, with as many as half, or more, of the positions filled with mixed-oxide driver rods or U-10Zr rods.

The irradiation conditions would be roughly similar to nominal conditions for FFTF driver assemblies (which are the conditions currently envisioned as nominal for an ATW system), and the irradiation time would extend to a year or two. It is quite likely that destructive and non-destructive interim examination of the experimental rods will be required to justify continued irradiation of the rods to higher burnup.

5.5.7.4 Integral Rod Test #2 in FFTF

If the results of the Integral Rod Test #1 are acceptable, then additional integral rods of these fuel forms will be prepared for a second open-core subassembly irradiation in the FFTF. This experiment is proposed to investigate performance of the fuel forms at aggressive conditions and would likely begin irradiation in the eighth or ninth year of the program. Again, the subassembly would likely be composed of various experimental fuel rods of the same dimension as FFTF driver rods, with as many as half, or more, of the positions filled with mixed-oxide driver rods or U-10Zr rods.

The irradiation conditions would be more severe than nominal for FFTF driver assemblies with respect to linear heat generation rate and fuel and cladding temperatures, and the irradiation time would extend to a year or two. It is quite likely that destructive and non-destructive interim examination of the experimental rods will be required to justify continued irradiation of the rods to higher burnup.

5.5.7.5 Additional Irradiation Experiments to be Determined

Because much of what is stated about the proposed fuel forms is speculative, ATW planners anticipate that unforeseen performance issues will manifest themselves in the feasibility and integral irradiation experiments. Such issues may include, for example, temperature effects on axial elongation of the fuel column, fuel constituent migration behavior, or effect of fission products and/or higher actinides on fuel-cladding compatibility. Many of these phenomena might best be investigated with carefully planned irradiation experiments in addition to those described above. Other issues to be addressed might include fuel response to certain types of transients (particularly those associated with reliability of the accelerator that drives the spallation neutron target), and some of these transients could conceivably be addressed with transient irradiation in the FFTF - perhaps in manner similar to that employed with the Transient Overpower Test Program in EBR-II in the early 1990's. Therefore, an additional two to four irradiation experiments during the first ten years of the program might be anticipated, although it is impossible to predict what form those experiments would take.

5.5.7.6 ATW Fuel Qualification for an ATW Demonstration Plant

Current plans call for an ATW Demonstration Plant to be constructed for the purposes of demonstrating the performance of an integrated ATW system, including accelerator, target, blanket, and fuel processing facility. At the moment, the best strategy for such a demonstration has not been determined. However, it is envisioned that the blanket initially will be configured for 30 to 100-MW operation and subsequently for 420-MW operation. Depending on the scheduling of the Demonstration Plant startup relative to that of the fuel development program, it may be necessary to qualify a fuel for the plant prior to startup. (Another possibility would be to start the plant with a blanket of U-10Zr, for which a safety case can likely be made using currently available information.) Therefore, an ATW fuel qualification program that uses FFTF may be

proposed. Such an irradiation program would likely involve irradiation of four to eight subassemblies to beyond design burnup values. The desired irradiation conditions would be those expected to be nominal for the ATW Demonstration Plant, with some of the assemblies configured for higher-than-nominal linear heat generation rates and/or temperatures.

5.5.7.7 Remarks on Fuel Testing for a LBE-Cooled ATW

As described in the earlier sections of this document, LBE is the preferred coolant for the ATW system currently proposed. If LBE were to be selected as the ATW coolant, ATW fuel development would have to include the demonstration of fuel assembly hardware and cladding compatibility with LBE Obviously FFTF cannot provide LBE coolant for open core experiments, so a LBE-cooled test loop for FFTF would be needed. If the ATW program evolves such that integral rod tests in LBE coolant are required, and if FFTF is available for irradiation testing, then it will be proposed that a LBE test loop be fabricated and utilized for at least two experiments similar to (or perhaps in lieu of) FFTF Integral Rod Tests #1 and #2 as described. These experiments would likely be irradiated during years 8 though 12 of an ATW program.

6. THE ATW T&B R&D ROADMAP: COST ESTIMATE SUMMARY

The estimated costs for each group of R&D activities that were presented in the preceding sections summarized here to provide an impression regarding the potential cost of an ATW target and blanket development program. Again, the reader is cautioned that the costs are estimates based on the judgment of a relatively small group of subject matter experts; some portions of the cost estimate have better basis than others.

The total cost estimates for programs to develop a sodium-cooled ATW and a LBE-cooled ATW are presented in Tables 6.1 and 6.2, respectively. However, the ATW Roadmap proposes that a decision be made in 2008 regarding which ATW coolant to be further developed. Therefore, the total cost estimates for the sodium-cooled, baseline system plus incremental costs for the LBE-cooled preferred system up through the year 2008 are presented in Table 6.3. After 2008, the program would then focus on one of the two coolant technologies, with costs as reflected in either Table 6.1 or Table 6.2.

Table 6.1 Cost Estimate Summary for Development of Target and Blanket Systems for a Sodium-Cooled ATW System

Fiscal Year	Nuclear Design and Safety	Heat Removal and Ancillary Systems	W Target Development	Fuel Technology	Total
2000	3,436	337	8,511	4,732	17,016
2001	3,857	3,427	30,796	4,132	42,212
2002	4,936	5,086	37,406	6,021	53,450
2003	3,565	7,034	17,207	5,849	33,655
2004	3,451	7,193	7,912	6,257	24,814
2005	3,680	9,115	5,316	5,657	23,769
2006	3,680	10,355	4,609	4,425	23,068
2007	3,497	17,041	8,282	5,230	34,050
2008	3,497	26,690	10,499	11,655	52,341
2009	3,494	24,844	0	11,074	39,411
2010	3,070	12,890	0	10,141	26,101
2011	3,069	8,171	0	5,309	16,550
2012	4,190	9,399	0	1,252	14,841
2013	4,036	8,348	0	1,551	13,935
2014	3,210	4,061	0	2,382	9,654
2015	3,554	461	0	4,276	8,291
2016	4,470	461	0	3,081	8,013
2017	4,465	461	0	11,687	16,614
2018	3,205	461	0	12,548	16,215
2019	3,196	461	0	8,283	11,941
2020	0	461	0	8,391	8,853
2021	0	461	0	4,543	5,004
2022	0	461	0	1,263	1,725
2023	0	461	0	0	461
2024	0	461	0	0	461
Total	73559	158607	130537	139740	502,444

Table 6.2 Cost Estimate Summary for Development of Target and Blanket Systems for a LBE-Cooled ATW System

Fiscal Year	Nuclear Design and Safety	Heat Removal and Ancillary Systems	LBE Chemistry and Materials Compatibility	LBE Target Development	Fuel Technology	Total
2000	3,589	5,787	3,689	10,897	4,944	28,906
2001	4,340	13,799	3,166	32,867	4,268	58,439
2002	5,620	16,193	5,269	40,811	6,598	74,491
2003	4,249	17,596	4,522	18,495	8,193	53,054
2004	4,135	15,202	1,334	15,280	9,978	45,929
2005	4,363	17,105	3,283	12,335	12,576	49,662
2006	4,363	10,995	3,497	4,609	6,929	30,393
2007	4,180	24,648	1,889	8,282	10,646	49,645
2008	4,180	50,461	1,243	10,499	14,558	80,941
2009	4,176	63,354	1,809	0	11,292	80,631
2010	3,223	44,661	2,698	0	10,261	60,843
2011	3,222	27,773	1,850	0	5,309	38,155
2012	4,343	26,257	0	0	1,252	31,852
2013	4,189	19,511	0	0	1,551	25,251
2014	3,363	11,421	0	0	2,382	17,166
2015	3,706	12,123	0	0	4,276	20,105
2016	4,623	5,915	0	0	3,081	13,619
2017	4,617	465	0	0	11,687	16,770
2018	3,205	465	0	0	12,548	16,219
2019	3,196	465	0	0	8,283	11,944
2020	0	465	0	0	8,391	8,856
2021	0	465	0	0	4,543	5,008
2022	0	465	0	0	1,263	1,728
2023	0	465	0	0	0	465
2024	0	465	0	0	0	465
Total	80881	386521	34249	154074	164810	820,535

Table 6.3 Cost Estimate Summary for Development of Target and Blanket Systems for a Sodium-Cooled ATW System Plus Incremental Costs for Co-development of a LBE-Cooled ATW System Through Year 2008

Fiscal Year	Nuclear Design and Safety	Heat Removal and Ancillary Systems	LBE Chemistry and Materials Compatibility	LBE and W Target Development	Fuel Technology	Total
2000	7,025	6,124	3,689	12,968	4,944	34,751
2001	8,196	17,226	3,166	38,293	4,268	71,149
2002	10,556	21,280	5,269	44,974	6,598	88,676
2003	7,814	24,630	4,522	26,177	8,193	71,336
2004	7,586	22,396	1,334	20,712	9,978	62,006
2005	8,044	26,220	3,283	15,171	12,576	65,293
2006	8,043	21,351	3,497	6,737	6,929	46,556
2007	7,677	39,022	1,889	10,350	10,646	69,584
2008	7,677	69,326	1,243	12,695	14,558	105,499

APPENDIX: HELIUM AS AN ATW COOLANT

Helium cooled reactors have operated in the US and Europe with varying degrees of success. In general, problems with these gas-cooled reactors have centered around the circulators rather than the reactors, which have operated well. Although designs and critical assemblies for gas-cooled fast reactor systems have been developed, to date only thermal systems (graphite moderated) have been deployed. DOE is currently funding a cooperative gas-cooled reactor development program with Russia for plutonium disposition. In addition, DOE has funded design and development efforts for a gas-cooled new production reactor, the NP-MHTGR. The Europeans are also investigating gas-cooled systems for ATW application. Some of the attractive aspects of the gas-cooled design for ATW are:

- Compatibility with metal fuel: Gas-cooling is compatible with the use of metallic fuel in a fast spectrum subcritical assembly as proposed in for the liquid metal cooled systems.
- No corrosion problems with the helium coolant
- Viable in-service inspection: In the gas-cooled assembly, the integrity of the fuel, fuel supports, heat exchangers, vessel, etc. are directly observable, which provides for enhanced safety and reliability.
- Potential elimination of steam generators by using a direct-cycle gas turbine.
- The ability to bring the beam in horizontally rather than vertically. Although likely to complicate core design, this possibility could eliminate a costly system to raise the high energy beam transport line above the level of the vessel with sufficient space to bend the beam downwards and provide the necessary magnets for beam expansion
- The helium is transparent to the neutrons so that no moderation occurs. This produces a very hard fast spectrum which provides favorable fission to capture ratios and makes for efficient fissioning of the actinides.

A.1 Description

A schematic of a helium cooled target and blanket system is shown in Figures A-1 and A-2. Although it shows a vertical beam insertion, a horizontal orientation could also be deployed. The proton beam is directed into the top of the assembly, and is directed onto the target in the middle of the core assembly to produce high-energy neutrons. As shown in table A-1, the core would be 150 cm high, with 100 cm top and bottom reflectors. The coolant would flow in the upward direction. The inlet temperature would be 300°C, and the outlet temperature would be 530°C or higher based on the use of high temperature fuel coatings. The design uses standard structural alloys such as 316 stainless steel and Inconel 718.

Table A-1. GC-ATW Assembly Design Parameters

Core	
Core Height	150 cm
Core layout	3 rings of fuel elements
Blanket Thickness	To be determined from conceptual studies
Top & Bottom reflector Height	100 cm
Coolant inlet temperature	300 ^{0}C
Coolant outlet temperature	530 ^{0}C

Helium flow rate	950 kg/sec
Average fuel operating Temp	1000^0C
Average clad/structure Temp	700 ^{0}C
Blanket Power	1000 MW-thermal

Fuel Element Design

Hexagonal shape containing GCFR type fuel rods	
Structure and clad	316 Stainless steel
External side length	3.863”
Inside distance across flats	6.491”
Outside distance across flats	6.691”
Box wall thickness	0.100”
Fuel rod OD	0.285”
Fuel rod pitch	0.389”
Clad thickness	0.019”
Number of rods per element	271
Number of rods per control element	234
Fuel element composition (relative to space in core)	
structure	15.1%
fuel	29.7%
helium coolant	44%
Void around element	11.2%
Maximum clad hotspot temperature	700^oC

Target Design Parameters

Target Type	Fluidized Bed of tungsten particles cooled by helium
Target location	Radial center of the core
Proton Beam	1000-MeV, 10-mA proton beam
Power delivered to the target	10MW
Beam Spot on target:	23 x 23-cm square or 25-cm diameter circle
Stopping length of beam in target:	67 cm
Assumed target dimensions:	Cylinder, 40 cm diameter by 80 cm high
Target helium coolant pressure:	290 psi
Target coolant inlet temp	201 ^{0}C
Target coolant outlet temp	433 ^{0}C
Target Helium coolant flow rate	5 kg/s
Average void fraction in particle bed:	0.46
Peak particle center temp:	446 ^{0}C
Tungsten-bed particle diameter:	1.5 to 2.0 mm
Bed superficial velocity:	20 m/s at exit
Neutron Source from target surface:	1.6 x 10^{14} n/cm^2-sec (E<20MeV)

The assembly would use the metallic fuel envisioned in the current roadmap, including plutonium and minor actinides. However, the amount of plutonium here would be very small if plutonium is

deep-burned in an accelerator-driven subcritical thermal-spectrum assembly as discussed earlier (in this case, the size of the fast spectrum assembly could be fairly small, possibly generating on the order of 300 to 400 MWt).

Structural metallic fuel material would be the same as currently envisioned for liquid metal assemblies, although the use of specialty claddings with high temperature tolerance would be investigated. Also as envisioned in the current roadmap, excess neutrons generated can be moderated and used in locations surrounding the transuranic blanket to transmute long-lived fission products such as Tc and I.

The energy in the helium coolant exiting the fuel would be available to either generate steam or directly drive a gas turbo-compressor and generate electricity.

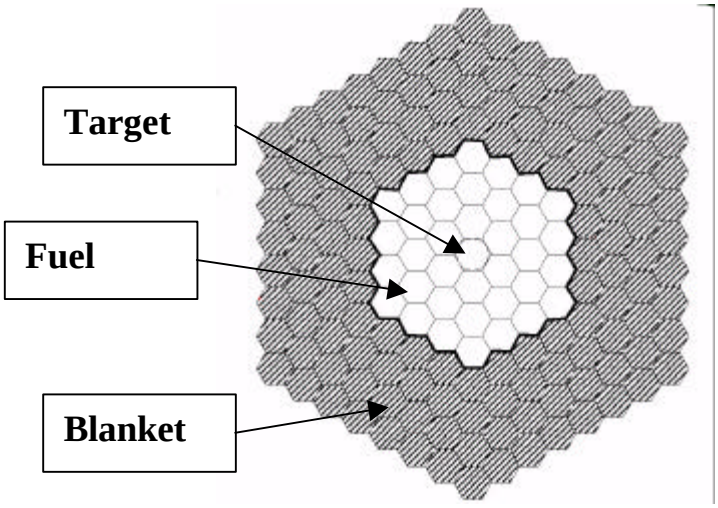

Figure A-1. Cross Section of a Subcritical, Gas-Cooled, Fast Spectrum Assembly

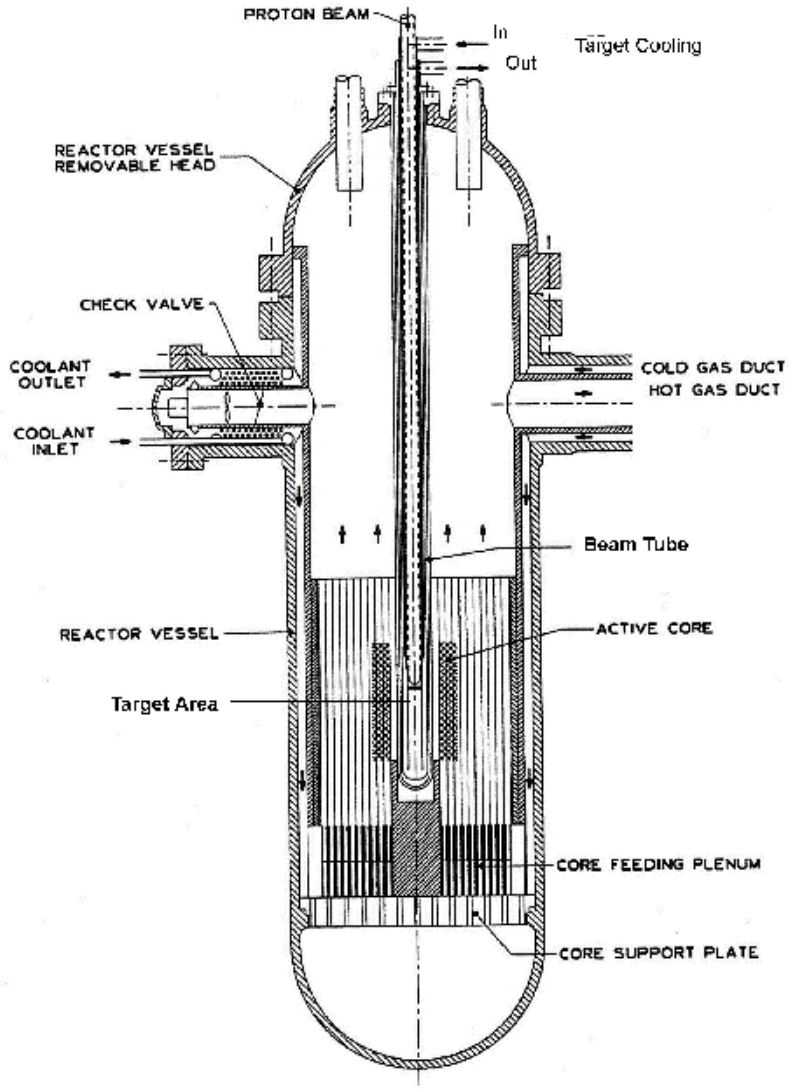

Figure A-2. Elevation of a Subcritical, Gas-Cooled, Fast Spectrum Assembly

A helium-cooled ATW target and blanket system uses a solid target such as tungsten, also cooled with helium. The target coolant would be separated from the blanket coolant in order to preclude spallation product contamination. A helium-cooled Inconel window provides the pressure boundary between the target coolant and the accelerator vacuum. A fluidized particle bed target is being evaluated for this application. As shown in table A-1, it operates between 201 and 433°C inlet and outlet temperatures, respectively. The bed enhances metal-to-gas heat transfer under forced flow conditions and provides a conduction heat removal path under low flow natural circulation conditions.

Destruction rates for the gas-cooled cycle are being developed at this time. In the fast-spectrum regime, it is anticipated that radionuclide destruction would be comparable to the one in liquid metal assemblies, or perhaps a little better because of the harder spectrum. However, preliminary calculations have been performed for an accelerator-driven (subcritical), gas-cooled, fast-spectrum assembly operating on the discharge of an accelerator-driven (subcritical), gas-cooled, thermal-spectrum assembly. The latter is assumed to operate on 1000 kg of weapons grade plutonium (WG-Pu). The results, which are summarized in the following table, are similar if LWR discharge is used instead weapons grade plutonium.

Table A-2. Destruction Rates for the GC-ATW (all units in kg).

Isotope	WG-Pu	Discharge from accelerator-driven thermal spectrum assembly	Discharge from a subsequent one year pass in accelerator-driven fast spectrum assembly	Composition after two hundred subsequent years in repository
Pu-238		2.7	2.4	0.5
Pu-239	940	3.6	1.4	1.4
Pu-240	60	41.5	28.3	27.8
Pu-241		34.9	13.9	0.0
Pu-242		63	49.8	49.9
Am-241		2.4	1.7	1.3
AM-242M		0.1	0.1	0.0
Am-243		13.8	16.8	16.5
Cm-242		1.8	0.9	0.0
Cm-243		0.0	0.0	0.0
Cm-244		4.6	8.8	0.0
Cm-245		1.7	0.9	0.9
Totals (kg)	1000	170.1	125	98.3

As can be seen in the above table, destruction of approximately 99% of the original Pu-239, and 80% of total plutonium is achieved. These numbers are of course higher with a longer irradiation time in the accelerator-driven fast spectrum assembly.

Preliminary calculations also show that an accelerator-driven fast spectrum assembly operating directly on LWR waste, without an initial accelerator-driven thermal spectrum assembly, achieves a burnup of total actinides of approximately 33% in a year.

To develop this technology for application to ATW, development activities would be necessary primarily for the target. Materials are similar to those investigated for use in the APT, but operate at higher temperatures. The effect of irradiation at high temperature is a departure from the APT experience, and would need to be investigated with irradiation tests. Also, the cooling of the target and window would need to be demonstrated. Information from the operation of a helium loop for the APT program in the beam at LANSCE is directly applicable. This test demonstrated helium clean-up techniques that are necessary to remove spallation products from the gas stream.

A.2 Helium-Cooled System Issues

A.2.1 Materials

To develop this technology for application to ATW, development activities would be necessary primarily for the target. Materials are similar to those investigated for use in the APT, but operate at a higher temperature. The effect of irradiation at high temperature is a departure from the APT experience, and would need to be investigated with irradiation tests.

There are structures of special geometries that require in-beam testing. This includes target canister prototypes, the target fluidized bed, fuel rod spacers and retention grids, and fuel assembly positioning keys.

Structural metallic fuel material would be the same as currently envisioned for liquid metal assemblies, although the use of specialty claddings with high temperature tolerance would be investigated.

A.2.3 Heat Transfer

This activity consists of verifying forced cooling processes in the gas-cooled target. This includes verifying (1) flow and pressure distributions around structures of special geometries, including beam and target vessel penetrations, target structures, and instrumentation penetrations, and (2) heat transfer properties in target and structures as a function of flows. Most of these tests can be performed out of beam and are thus reasonable in cost. Both normal operation and accident conditions need to be investigated.

A.2.4 Helium Cleanup

Information from the operation of a helium loop for the APT program in the beam at LANSCE is directly applicable and should be used as a starting point for design. For example, LANSCE testing demonstrated helium clean-up techniques that are necessary to remove spallation products from the gas stream.

Hydrogen and other noble gasses will be produced in the solid tungsten target as well as other structures directly in the path of the proton beam due to spallation reactions. At the temperatures that the structures will operate, some of the gases will diffuse into the coolant. This process needs

to be understood, and methods for controlling the buildup developed. For hydrogen removal, for example, a slipstream can be removed from the main coolant stream, and passed through a palladium membrane.

A.2.5 Power Conversion

Power conversion system development consists of verifying the performance of a direct-cycle power conversion system for the gas-cooled T&B. This includes fabrication of a turbo-compressor and bearings, heat recuperators, helium ducting prototypes, and vessel prototypes. It also includes testing the power conversion components under load conditions generated by nuclear heat, or fossil-fueled heat, or under heat of compression generated by motoring the generator on the turbo-compressor. These tests are expected to be performed as part of the international plutonium disposition program.

A.3 Helium T&B Development Plan

Major development activities for the gas-cooled T&B include fuel and target development, cooling verification, prototyping, and verification testing of special safety features. A description of these activities follows, with projected costs summarized in the table below.

A.3.1 Fuel Processes Development

The fast spectrum gas-cooled T&B would use metallic fuel of the type considered for the liquid metal option. Thus, the process for separating uranium and fission fragments from uranium and actinides, and the process for forming fuel rods, is common to both systems. Consequently, basic costs may be considered the same.

Nevertheless, recent advances in high temperature refractory materials that are stable in helium, have the potential to contain molten fuel, and provide a gas-tight enclosure for the fuel, could be considered. Thus, under fuel processes development, an activity is included to produce and test high temperature fuel coatings. This would involve extending APT materials irradiations data to high temperature regimes. Based on the gas-cooled reactor experience, a $14.6M program is envisioned. As a point of reference, this would be approximately 18% of the APT materials performance effort.

Facilities needed for this effort already exist in US national laboratories. However, a $3M cost estimate is included for facility modifications, for a total cost of $17.6M. Coatings for a thermal spectrum assembly, should it be selected for implementation, would be qualified as part of the separate international plutonium disposition program.

A.3.2 Metallic Components Development

This activity includes qualifying metallic structural materials for target and fuel support structures. Much of the data for the design of these components can be obtained from the APT project, and a significant part of the effort would be common to the liquid metal program.

However, gas-cooled assemblies are subjected to lower pressure differentials and, therefore, lower mechanical loads than liquid metal assemblies. In addition, helium poses significantly less risk of

corrosion and other chemical attack challenges to metallic components than liquid metals. Thus, the common part of this development effort is probably a modest fraction of the liquid metal effort.

On the other hand, it is recognized that there will be structures of special geometries that may require prototype irradiation testing. This may include target canister prototypes, fuel rod spacers and retention grids, and fuel assembly positioning keys.

Based on the gas-cooled reactor and APT experience, an estimate of $5M is included for these prototype tests. Facilities needed for this effort already exist in US national laboratories.

A.3.3 T&B Prototyping

This activity consists of verifying fabricability of gas-cooled metallic target and blanket prototypes, including the components discussed above: target canister prototypes, fuel rod spacers and retention grids, and fuel assembly positioning keys.

Based on APT experience with target and blanket prototypes, an estimate of $5M is included for these prototypes. Facilities for this effort exist in industry and US national laboratories.

A.3.4 T&B Cooling

This activity consists of verifying forced cooling processes in the gas-cooled T&B. This includes verifying (1) flow and pressure distributions around structures of special geometries, including beam and target vessel penetrations, target assemblies, fuel assemblies, and instrumentation penetrations, and (2) heat transfer properties in target and fuel assemblies as a function of flows.

The cost of the proposed development effort is based on the use of an air flow facility with electric heating to perform the tests on mockups, and adjusting the results to helium and gas-cooled assembly properties and conditions.

Based on gas-cooled reactor experience, an estimate of $2 M is included for these tests. Facilities for this effort exist in industry, US national laboratories, and universities.

A.3.5 Power Conversion

Power conversion system development consists of verifying the performance of a direct-cycle power conversion system for the gas-cooled T&B. This includes fabrication of a turbo-compressor and bearings, heat recuperators, helium ducting prototypes, and vessel prototypes. It also includes testing the power conversion components under load conditions generated by nuclear heat, or fossil-fueled heat, or under heat of compression generated by motoring the generator on the turbo-compressor.

These tests are expected to be performed under the international plutonium disposition program, and, therefore, no costs are included for this task.

Should the schedules for the plutonium disposition program not be compatible with the ATW program, power conversion would be accomplished using proven steam generators and steam turbines operating at modern fossil-fueled thermodynamic conditions. This would leave the cost

for completing the verification of circulator design to be covered, which based on gas-cooled reactor experience is estimated at $15M, including modifications to existing industrial facilities.

A.3.6 Shutdown Cooling System

This activity consists of verifying cooling processes in the gas-cooled T&B under low flow conditions. This includes operation of low power forced circulation blowers, and natural circulation. As in the case of verifying forced cooling processes above, this includes verifying (1) flow and pressure distributions around the entire fuel and target circuits, and (2) heat transfer properties in the same circuits.

It also includes verifying natural circulation in the T&B building, a process that provides the ultimate heat sink to heat exchanger panels on building walls.

The cost of the proposed development effort is based on the use of an air flow facility with electric heating to perform the tests on mockups, and adjusting the results to helium and gas-cooled assembly properties and conditions. Based on gas-cooled reactor experience, an estimate of $5 M is included for these tests. Facilities for this effort exist in industry, US national laboratories, and universities.

A.3.7 Fuel Handling and Storage

This activity consists of verifying remote handling of fuel and target systems. This includes operation of cranes, positioners, transfer casks, fuel and target pond cooling, etc.

Based on gas-cooled new production reactor and APT experience, an estimate of $5M is included for these tests. Facilities for this effort exist in industry and US national laboratories.

A.3.8 Plant Control and Protection

Although control, instrumentation and protection systems do not represent special challenges in the gas-cooled T&B, a modest program is proposed to qualify high temperature detectors, particularly thermocouples and fission chambers. Based on similar qualification programs in the gas-cooled reactor program, a $4M effort is proposed. Facilities for this effort exist in industry and US national laboratories.

Attachment 1

Summary of Estimated Cost and Schedule for Nuclear Design and Safety Tasks Proposed for Development of the Target and Blanket for a Sodium-Cooled ATW System

Attachment 2

Summary of Estimated Cost and Schedule for Nuclear Design and Safety Tasks Proposed for Development of the Target and Blanket for a LBE-Cooled ATW System

Attachment 3

Summary of Estimated Cost and Schedule for Coolant Chemistry and Materials Compatibility Tasks Proposed for Development of the Target and Blanket for a LBE-Cooled ATW System

Attachment 4

Summary of Estimated Cost and Schedule for Heat Removal and Ancillary Systems Tasks Proposed for Development of the Target and Blanket for a Sodium-Cooled ATW System

Attachment 5

Summary of Estimated Cost and Schedule for Heat Removal and Ancillary Systems Tasks Proposed for Development of the Target and Blanket for a LBE-Cooled ATW System

Attachment 6

Summary of Estimated Cost and Schedule for Tungsten Target Development Tasks
Proposed for Development of the Target and Blanket for a Sodium-Cooled ATW System

Attachment 7

Summary of Estimated Cost and Schedule for LBE Target Development Tasks Proposed for Development of the Target and Blanket for a LBE-Cooled ATW System

Attachment 8

Summary of Estimated Cost and Schedule for Fuel Development Tasks Proposed for Development of the Target and Blanket for a Sodium-Cooled ATW System

Attachment 9

Summary of Estimated Cost and Schedule for Fuel Development Tasks Proposed for Development of the Target and Blanket for a LBE-Cooled ATW System