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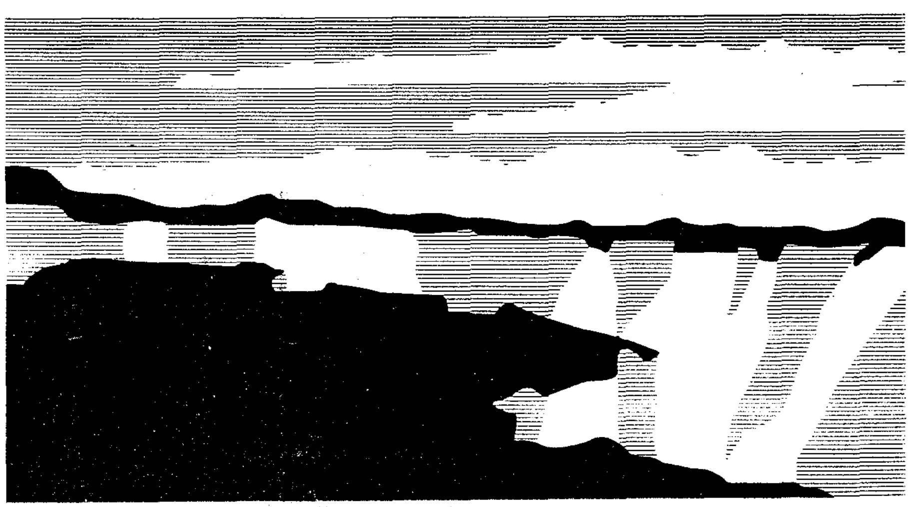

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ACCELERATOR-BASED CONVERSION (ABC) OF WEAPONS PLUTONIUM: PLANT LAYOUT STUDY AND RELATED DESIGN ISSUES

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ABSTRACT

In preparation for and in support of a detailed R&D Plan for the Accelerator-Based Conversion (ABC) of weapons plutonium, an ABC Plant Layout Study was conducted at the level of a pre-conceptual engineering design. The plant layout is based on an adaptation of the Molten-Salt Breeder Reactor (MSBR) detailed conceptual design that was completed in the early 1070s. Although the ABC Plant Layout Study included the Accelerator Equipment as an essential element, the engineering assessment focused primarily on the Target; Primary System (blanket and all systems containing plutonium-bearing fuel salt); the Heat-Removal System (secondary-coolant-salt and supercritical-steam systems); Chemical Processing; Operation and Maintenance; Containment and Safety; and Instrumentation and Control systems. Although constrained primarily to a reflection of an accelerator-driven (subcritical) variant of MSBR system, unique features and added flexibilities of the ABC suggest improved or alternative approaches to each of the above-listed subsystems; these, along with the key technical issues in need of resolution through a detailed R&D plan for ABC are described on the bases of the "strawman" or "point-of-departure" plant layout that resulted from this study.

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I. EXECUTIVE SUMMARY

Accelerator-Based Conversion (ABC) of commercial and weapons plutonium to short-term radioactive waste and net electrical energy proposes to exploit unique benefits and technical discriminators that evolve from the joining of driven (subcritical) nuclear operation with a low-inventory, fluid-fuel system. As part of an ongoing and broadening technical assessment of technical merits, an ABC Plant Layout Study was initiated to develop an early appreciation for size, inventory, operational, maintenance, safety, and general interfacial issues.

Since the molten-salt-based ABC approach is only in the earliest conceptual stage of development, this ABC Plant Layout Study relied heavily on the detailed conceptual engineering design of the Molten-Salt Breeder Reactor (MSBR) completed by the Oak Ridge National Laboratory in the early 1970s. Scaling from the MSBR design, a quantitative layout of a single (711 MWt) Target/Blanket unit for the molten-salt ABC is reported; four of these Target/ Blanket units would be driven by a single accelerator; and three such 2,844-MWt [1,263 MWe(gross); 1,074 MWe(net)] ABC systems would be required to dispose of ~50 tonne of weapons plutonium in 20 years for an average plant availability of 75%. The scaling of all key components from spallation target → primary systems (blanket and primary coolant) → secondary-coolant systems → balance of plant, including important elements of the chemical-processing system, are reported. On the basis of this scaling, the ratio of fuel salt in the blanket to that in the entire system is 0.34; the total fuel-salt power density (including exo-blanket inventory) is 57. MWt/m^3; and the ratio of containment volume to thermal power is 32.m^3/MW.

Key technical issues that have been defined in the course of this ABC Plant Layout Study are summarized, as they relate to: target-blanket longevity from both radiation-damage and chemical-corrosion view points; molten-salt chemistry issues ranging from time-varying plutonium-fuel solubilites to structural attack by soluble fission products; the (vertical) Target/Blanket maintenance scheme; the use of sodium fluoroborate secondary-coolant salt *versus* other secondary coolant options; cost (*e.g.*, high pressure) *versus* benefit (*e.g.*, high thermal-to-electric conversion efficiency) of the supercritical-steam cycle adopted from the MSBR design; the feasibility of the chemical separations for noble-gas, noble-metal, and soluble fission products that form the basis of the chemical-processing scheme adopted; the in blanket and exo-blanket disposition of gaseous, noble-metal, and soluble lathanides waste streams and the magnitude of these waste streams when used target-blanket materials are included; and general containment and safety considerations related to the fluid-fuel system adopted. These issues are quantitatively identified in the context of the ABC Plant Layout Study for elaboration by an ongoing, parallel ABC R&D planning activity.

While the main goal of the ABC Plant Layout Study is to provide early input to the ABC R&D Plan, the main goal of this report, in addition to listing all major non-accelerator engineering issues for the ABC R&D Plan, is to summarize: a) the characteristics of a basecase ABC "strawman" design; b) key scoping calculations (target, blanket, beam bending magnets, *etc.*); and c) the groundrules and scaling procedures used to translate the detailed and well-documented MSBR conceptual design into the context of ABC. The latter two contributions are of particular importance to the generation of a self-consistent ABC conceptual design (an associated cost estimates) in the future, and for these reasons the groundrules, MSBR → ABC engineering scaling relationships, and ancillary support computations have been thoroughly documented in appendices to this report.

A number of key technical and operational issues have been identified in the course of conducting this ABC Plant Layout Study. Running as a common thread through all these technical issues are materials concerns related to component longevity in a highly corrosive and high-radiation environment. These material issues impact all operational, safety, economic, and environmental projections for ABC. While the comprehensive, but somewhat aged, MSBR "data base" has been used extensively in the selection of structural materials, nuclear components, and molten-salt compositions, the flexibility offered by the subcritical-driven ABC approach opens possibilities not available to MSBR; unfortunately, little or no experience beyond that provided by the MSBR project is available. Recognizing this common materials thread and related database limitations, key technical issues identified by the ABC Plant Layout Study are summarized below according to the main ABC subsystem; elaborations of these points are found in the main body of the report and the appendices.

- Accelerator Equipment:
  - all physics and engineering requirements needed to assure ≥ 75% availability for a 800-1,000 MeV, 50-100 MW(beam) proton Linac that is multiplexed with four independent Target-Blanket and Balance-of- Plant systems that in effect comprise four independent ~300-MWe power stations; these issues where not included in the charter of the APC Plant Layout Study;
  - topology of High-Energy Beam Transport system that linearly in series "kicks off" four beamlets to each of the ~4 ABC power-plant modules;
  - beamlet transport, bending, expansion, and "footprint" control upon impinging each Target window after traversing the primary containment building (tertiary containment boundary) where major maintenance operations (on each ABC module) must occur;
  - need for fast-acting Beam Tube Isolation Valves (BTIVs) on a system that links directly all three confinement zones [unlike the similar Main Steam Isolation Valves (MSIVs), that connect only the outer containment zone to the environment]; incorporation of the accelerator tunnel/buildings into the three-tiered containment system adopted for ABC would be prohibitively expensive.

- Target:
  - availability of containment material with acceptable longevity in a high-temperature (?????K), corrosive (flowing liquid lead), intense radiation field (?????$\times 10^{20}$ $n/m^2/s$ high-energy neutrons) environment;
  - thermal uniformity and effectiveness of the self-cooled, integrated window that separates the Target-Blanket from the high-energy proton-beam line and Accelerator Equipment vacuum system;
  - maintenance configuration (vertically into Containment Building) and separability from molten-salt/graphite/Hastelloy-N blanket; thermal insulation between Target and Blanket systems to control heat leakage (to Target coolant system);
  - choice and configuration of Target coolant system; choice between rejection of target power (including blanket thermal in-leakage) as low-grade heat *versus* recovery by thermal-conversion cycle for addition to gross-electric output.

- Primary System:
  - Core:
    - -- configurational choice (MSBR-like homogeneity *versus* fully reflected) as related to component (moderator, moderator/reflector, internal structure, reactor vessel) longevity in a high neutron flux (?????$\times 10^{20}$ n/m^2/s);
    - -- fuel-salt/fission-product/plutonium interactions with graphite and extent of post-irradiation cleanup needed to assure minimum waste stream that can be classified as Low Level Waste;
    - -- maintenance configuration (vertically into Containment Building) and relative separability of reactor vessel from other Primary System components (pumps, fuel-salt dump tank, IHXs).
  - Fuel-Salt Pump:
    - -- straight-forward pump design, but no operating experience with pumps of capacity required by ABC;
    - -- efficacy of the pump (bowl) as a major element in the Chemical- Processing system (fission-product off-gas release from fuel salt) and volume of fuel-salt inventory in pump bowl;
    - -- length of drive shaft (???? m) needed to provide adequate distance between highly radioactive fuel salt and the radiation-sensitive pump motor, from both mechanical and maintenance viewpoints.
  - Intermediate Heat Exchanger (IHX):
    - -- feasibility of and need for the unique U-tube/U-shell configuration adopted from MSBR, that efficiently minimized fuel-salt volume but may present a non-optimal maintenance geometry for ABC;
    - -- deposition of noble-metal (relative to fluorine) fission products onto cooler surfaces of IHX, and opportunity to convert a potential problem into an option for that component of the Chemical-Processing system;
    - -- possible need to provide a tritium diffusion barrier to prevent tritium migration into the secondary coolant-salt system and beyond;
    - -- control of tube leaks and secondary-coolant-salt egress into the fuel salt.
  - Fuel-Salt Piping:
    - -- optimum pipe sizes and pipe runs that minimize further the exo-blanket fuel-salt inventory while assuring acceptable flow velocities in a configuration that optimizes an otherwise messy component-maintenance operation;
    - -- deposition and accumulation of noble-metal fission products.

- Balance of Plant (BOP):
  - Secondary-Coolant-Salt System:
    - -- Intermediate Heat Exchanger: same as discussed above under the Primary System;
    - -- Secondary-Coolant-Salt Pump: design similar to fuel-salt pump, except for need to purge pump bowl of gaseous fission products; an added complication related to SG/SR flow metering, as listed below, is identified, however.
    - -- Steam Generator (SG):
      - * feasibility of and need for the unique U-tube/U-shell configuration adopted from MSBR;
      - * design and operation of high-pressure (25.8 MPa, 810 K) supercritical-steam (SCS) cycle, and impact of steam-tube failure on Secondary-Coolant-Salt-System design (including respective cells)

    * metering of secondary coolant salt to accommodate SG/SR split using variable-speed pump motors *versus* metering valves.
  -- Steam Reheater (SR):
    * impact of tube failure and subsequent pressurization of the Secondary-Coolant-Salt System;
    * concern of freezing secondary coolant salt and need to maintain a minimum feedwater temperature.
- Power-Conversion Equipment:
  -- Steam Generator (SG): as addressed above under Secondary-Coolant-Salt System
  -- Steam System Piping:
    * design for SCS system (25.8 MPa, 810 K) and need for thick-walled pipes and long pipe runs;
    * because of the last requirement, many smaller steam tubes required to deliver steam to the Turbine Plant Equipment, with impact on the number and reliability of MSIVs.
  -- Turbine Plant Equipment:
    * economic impact of using low-capacity (316 MWe) turbines;
    * possible need to locate turbine within a containment building for reasons related to tritium migration and/or the need for multiple MSIVs with reduced ensemble reliability.
  -- need to examine operational, design, safety, and cost trade offs associated with higher-efficiency SCS cycle and a less-efficient but simpler Power-Conversion systems.

• Chemical Processing:
  - Off-Gas Processing:
    -- efficacy of helium-gas sparging in the fuel-salt pump bowl to separate gaseous fission products, compared to implementation as a separate unit;
    -- efficiency, volumes, stability, and waste streams associated with getter-bed collection on activated charcoal (MSBR) or zeolites;
    -- need and means for post-collection separations and re-introduction of specific fission products into Core for subsequent irradiation.
  - Fuel-Salt Drain Tank:
    -- need for and advisability of the multifarious role of Fuel-Salt Drain Tank *vis-á-vis* Chemical Processing (of nonvolatiles), central collection point, fueling station, standby storage during maintenance of all Primary System components, and safe storage and afterheat removal under loss of (normal) cooling conditions (to name a few);
    -- reliability and speed (both opening and flow times) of fuel-salt freeze valve that connects Core with Fuel-Salt Dump Tank;
    -- generally "captured" location in reactor-vessel cell and ability to monitor and maintain.
  - Fuel-Salt Cleanup Systems:
    -- feasibility and means of on-line removal of noble-metal fission products (cold traps, electrowinning, REDOX control, *etc.*)
    -- feasibility and means of post-irradiation batchwise cleanup of fuel salt from plutonium and higher actinides (for re-injection into the fuel salt) and soluble fission products (lanthanides);

    - -- degree to which rejected salt can be classified as Low-Level Waste, and degree to which salt recycle can be implemented.
  - Other Cleanup Operations/Systems:
    - -- target lead cleanup of spallation and corrosion products;
    - -- cleanup of Core components (mainly graphite) prior to disposal as (ideally) reduced-volume, Low-Level Waste;
    - -- cleanup and (ideally) recycle of off-gas getter beds.

- Instrumentation and Control (I&C):
  - Accelerator Equipment interface and control with Primary System, Balance-of-Plant, and Safety systems requires a detailed and self-consistent design before the myriad of control issues under both transient (scheduled or unscheduled) and steady-state conditions can be identified and assessed; a similar statement applies to the other I&C categories listed below;
  - nuclear and power control systems operated in conjunction with chemical and mechanical controllers distributed throughout the Primary System;
  - Balance-of-Plant I&C systems dealing with internal operations and safety conditions and responses of each of four power-conversion systems, electrical power distribution within each 316 MWe unit [particularly for Accelerator Equipment power requirements (152 MWe)], and distribution of reliable electrical power to the electrical grid for needed revenue generation;
  - resolution of individual and interactive I&C requirements associate with plant (Primary System) operations, on-site radwaste storage, fuel-salt conditioning and cleanup, and overall waste-stream management.

- Safety Systems:
  - increased ABC concept resolution needed to verify the feasibility of a three-tiered confinement philosophy under both operating and maintenance conditions;
  - define better the means of reactivity and power control within each core (fuel-salt composition, control/shutdown rods inserted into the Core, fuel-salt flow rate, *etc.*)
  - improve understanding of multiply-connected equipment cells (*e.g.*, reactor vessel, fuel-salt dump tank, secondary coolant salt, SCS generator, chemical processing, *etc.*) responses to failure of interfacial equipment (*e.g.*, IHXs, SGs, Fuel-Salt Dump Tank, *etc.*);
  - resolve better the multi-functional role of the primary containment building as this structure provides: a) the tertiary containment envelope; b) the systems for the last manipulations/conditioning of the high-energy proton beam; and c) the central volume and laydown area for maintenance of major equipment in the Primary System.

II. INTRODUCTION

A. Background

The use of accelerator-produced neutrons to sustain high burnup of weapons plutonium in a subcritical configuration as been proposed[1-3] as a means to dispose of this material[4,5]. When combined with a fluid-fuel blanket, the driven (subcritical) Accelerator-Based Conversion (ABC) system for the burning of plutonium and the concomitant generation of electrical power offers a number of symbiotic benefits and discriminating characteristics[3]. As summarized in Table I, these benefits and discriminators that derive from the combination of burning plutonium in a subcritical, fluid-fuel power plant center on the prospects of enhanced nuclear safety in a system that offers: a) significant operational flexibility resulting from a relaxed neutron balance; b) the prospects of a reduced long-lived waste stream, "deep" plutonium burns; c) and a shift of unit operations away from chemical processing towards physical separations. These characteristics combine to promise a safer, cleaner, and more-flexible deep-burn system with reduced far-term population doses. Many of the processes upon which these claims are build; however, remain to be taken beyond the preconceptual level and, along with the need to minimize the capital and operational costs associated with the accelerator-based neutron generator, are recognized a crucial uncertainties in need of resolution.

The ABC approach to dealing with weapons and commercial plutonium has been explored primarily at a conceptual level[1]; only relatively unintegrated target, blanket-neutronics[7], blanket thermal-hydraulics, materials[6], and chemical-separations[8] scoping calculations have so far been made. While not sufficient to commence a detailed conceptual design of an ABC, the essential elements of this system are adequately defined to begin a preliminary plant layout, given that key ABC subsystem choices are made and related assumptions can be accepted. Furthermore, the process used to make the choices and assumptions needed to advance a preliminary plant layout provides a strong focus for the development of the ABC concept. This focusing onto and identification of the main technical issues for key ABC subsystems, as well as beginning a more concrete assessment of the benefits and discriminators listed in Table I, is the primary goal and product of this ABC Plant Layout Study. This ABC Plant Layout Study, therefore, serves an important integrating function that can be applied prior to any preconceptual design activity to assure that the unique characteristics of this accelerator-driven, fluid-fuel system are fully exploited while using the best of the ideas developed in conjunction with the detailed MSBR design.

B. Scope and Approach

The main goal of this ABC Plant Layout Study is the generation of a preliminary, but self-consistent, engineering layout of a weapons-plutonium-burning ABC. This plant layout is based on the individual scoping computations of key subsystem elements; some of these scoping calculations are reported in the Appendices to this report. The technical trade offs, options, or choices required to generate this ABC plant layout are collected and prioritized to provide the main issues used to define a long-term R&D plan[9] for ABC. A basecase set of design assumptions and parameters is needed to perform this preconceptual, plant-layout task. Central to the definition of this base case is the direct adaptation and scaling of the early detailed design of the Molten-Salt Breeder Reactor (MSBR) concept[10,11]. The more-or-less direct application of the MSBR design to the ABC Plant Layout Study was made to expedite the engineering layout and the technical issues this layout defines, and not because the MSBR parameters in the context of ABC were necessarily optimal or the best choice(s). While the advantages of the molten-salt fluid fuel for these accelerator-driven systems are well known[1,2], the specific MSBR embodiment for ABC applications may not represent the best choice, as will be shown. Never-the-less, use of MSBR experience in generating the ABC base case reported herein has allowed the plant layout and associated technical assessment to proceed on the basis of the substantial, related technical work reported as part of the MSBR conceptual engineering design[10].

Central to obtaining any meaningful result from a pre-conceptual design study of the kind reported here is a clear statement of design ground rules. After a brief description of both the ABC and MSBR concepts in Sec. I.C., these design ground rules are laid out in Sec. II.D. for both the scale of the plutonium disposition task and for the ABC design options and focused base case that form the core of this ABC Plant Layout Study. Section III. reports the ABC plant layout on a subsystem-by-subsystem basis, with the details of the engineering scaling and assumptions used to generate the basecase layout being described in Appendix A. The main technical issues identified in the course of generating the MSBR-based ABC base case are described in Sec. IV. in both general and a subsystem-by-subsystem contexts. After prioritizing these issues (Sec. IV.C.), as well as identifying attractive design alternatives to the MSBR base case (Sec. IV.D.), Sec. V. concludes with a summary of "top-level" R&D requirements for an optimized ABC-based disposition of weapons-grade plutonium; Sec. V. also gives recommendation for optimal technical directions to be taken by any future, more-detailed conceptual engineering design of ABC.

C. Concept Description

Figure 1 gives a systems block diagram of the ABC plant. The ABC is divided into the following four main systems: Accelerator (ACC); the target (TAR) and blanket (BLK), which together form the reactor core and, when combined with the primary pump(s) and intermediate heat exchangers (IHXs), comprise the primary heat-transport (PHT) system; the secondary (coolant-salt) heat-transport (SHT) system; the steam and power-conversion system, which is designated here as the balance-of-plant (BOP); and the chemical plant equipment (CPE) system that is comprised of fuel-loading, off-gas (tritium and volatile fission products) handling, non-volatile fission product (physical and/or chemical separations). When superposed onto a commonly adopted Program of Cost Codes[12,13] that must be used to evaluate ultimate techno-economic trade offs[14,15], and including power and mass flows that characterize an ABC that generates net-electric power, the systems diagram given in Fig. 2 results. In the most aggregated form, the ABC plant consists of Accelerator and Reactor Plant Equipment (A/RPE), Chemical Plant Equipment (CPE), and Balance of Plant (BOP), all situated on and within Structures and Site (SITE) systems. After briefly reviewing the MSBR concept, the resulting marriage with an accelerator-base neutron source to form the ABC concept is described.

1. Molten Salt Breeder Reactor (MSBR)

The essential elements of the ABC nuclear and power-conversion systems used to define the ABC base case have been taken directly or scaled from the MSBR engineering design reported in Ref. 10. This 1,000-MWe(2,250 MWt, 44.4% efficient) power-plant design utilizes four LiF-BeF_2-$(Th,U)F_4$ primary (fuel-salt) coolant loops, that transferred the fission power to a NaF-$NaBF_4$ secondary (coolant-salt) loop; the secondary coolant salt in turn drives an advanced supercritical-steam (SCS) power conversion that relies on strong reheat from the secondary coolant salt to achieve a high thermal-conversion efficiency. Figure 3 is a composite replication of the MSBR coolant(s) and power-conversion systems, and is include here because of its frequent comparative use in the ABC Plant Layout Study. The level of conceptual-design detail available for most of the key subsystems listed on Fig. 3 made the Ref.-10 study particularly valuable as a resource with which to scale the present molten-salt ABC plant layout, despite obvious differences in application, engineering, materials, and neutronics constraints, and driving technologies. Many of these engineering and materials differences reflect the need to accommodate a neutron spallation target and the sub-critical operation of the ABC core, as is summarized in Table I. The shift from use of "chemical separations" in the MSBR to "physical separations" in the ABC design also represents an important deviation. The essential elements of a molten-salt ABC are illustrated in Fig. 4, which gives a "top-level" power- and mass-flow diagram that has been loosely adapted from the MSBR design shown in Fig. 3.

2. Accelerator-Based Conversion (ABC)

As adopted in the MSBR design, the fuel-salt dump tank (Figs. 3 and 4) serves as a focal point for most CPE-related operations, including any slip-stream operations associated with the (preferred) electrowinning (*e.g.*, electrolytic deposition) collection of noble (with respect to fluorine) metal fission products. Figure 4 also indicates of the tertiary confinement approach adopted for both the MSBR conceptual design and the ABC plant layout reported herein. Also shown is a reheat stream taken directly from the secondary coolant-salt stream for possible use in driving a supercritical-steam thermal-to-electric conversion cycle at the high efficiencies projected for the MSBR. Lastly, for the purposes of establishing design ground rules in the definition of the ABC base case, the target window (WIN), operational and maintenance (O&M) systems, and systems related to plant safety (SAF) are identified as organizational units, albeit, some connectivity between these ten ABC systems exists (Sec. II.D.2.).

The thermal and electrical power flows indicated on Figs. 2 and 4 show the conversion of electrical power P_{EA} delivered to the accelerator to beam power P_B with an overall "wall-plug" efficiency of $\eta_A = P_B/P_{EA}$. Upon passing through a window, this beam power is converted (ideally) to fission power, P_F, with a gain $P_F/P_B = \beta\, k_{eff}/(1 - k_{eff})$, where k_{eff} is the blanket neutron multiplication and $\beta \sim 1.7$-1.8 is the ratio of fission energy per fission neutron, E_F/ν, to beam energy per target neutron, E_B/Y. Typically, the target neutron yield can be approximated by $Y \simeq (E_B - E_B^o)/y$, where the fitting constants are $E_B^o \simeq 200$ MeV/p and $y \simeq 35$ MeV/n; for $E_F = 200$ MeV/n, $\nu \simeq 2.8$ n/fission, and $E_B \simeq 800$ MeV/p, it follows that $E_p/\nu = 71.4$ MeV/n, $E_B/Y = y/(1 - E_B^o/E_B) \simeq 46.7$ MeV/n, and $\beta = (E_F/\nu)/E_B/Y) = 15.3$. Once converted to total or gross electrical power, $P_{ET} = \eta_{TH}\, P_F$, and after skimming off the accelerator power, P_{EA}, and a small amount of non-accelerator BOP power, $P_{AUX} = \varepsilon_{AUX}\, P_{ET}$, the net power $P_E = P_{ET} - P_{EA} - P_{AUX}$ is delivered to the grid. With the total plant recirculating-power fraction defined as $\varepsilon = \varepsilon_{AUX} + P_{EA}/P_{ET}$, the net plant efficiency is $\eta_p = \eta_{TH}(1-\varepsilon)$; typically, ε is in the range 0.15 - 0.20 for highly multiplying, but still significantly subcritical, blanket assemblies.[14,15]. Hence, whereas $\eta_p \simeq \eta_{TH} = 0.44$ for the MSBR, the equivalent ABC plant efficiency would be reduced to ~0.34-0.37.

A direct mapping of the MSBR mass and power flows (Fig. 3) into those expected of the plutonium-burning ABC (Figs. 2 and 4) does not have a one-to-one correspondence, even if approximate size and capacity scalings are available (Appendix A). Differences in fuel-salt compositions [the ABC has no (Th,U)F_4]; a more flexible neutron economy related to the added accelerator-produced neutrons (Table I) and no need to breed ^{233}U from ^{232}Th; material problems related to operation of a high-power liquid-lead target in a molten-salt environment; impact of target and accelerator on vertical maintenance scheme; *etc.* limit the benefits of directly applying the fruits of the detailed and self-consistent MSBR conceptual engineering design to the preliminary ABC concept. This mapping, however, is guided by the ground rules described in the following Sec. II.D.

D. Design Ground Rules

Ground rules adopted for the ABC Plant Layout Study have been generated to establish more firmly the many options and opportunities available to the molten-salt, fluid-fuel ABC design(s). These ground rules are divided into two broad categories: a) those that set goal plutonium disposition rates and related ABC capacities, irrespective of the characteristics of the ABC primary, secondary, power-conversion, and chemical-plant systems; and b) those ground rules used to establish broad characteristics of the MSBR-derived ABC base case. It cannot be overstressed that this latter base case is defined and generated solely for the attributes of maximal self-consistency and utilization of the MSBR design result, rather than suggesting a design that is optimal from the view point of the ultimate ABC application. In this sense, the ABC base case generated from the ground rules given in Sec. II.D.2. should be considered a "point-of-departure" (POD) reference case.

1. Goal Disposition Capacity

Typically, the plutonium-disposition rate is determined by specifying that M_{Pu} tonnes of plutonium is to be destroyed [~ 90% fissioned, with addition of fission boost through highly enhanced uranium (HEU) near end of life (EOL) to achieve > 95% plutonium burnup] in a chronological time T_{LIF}(yr) by a system that on the average operates at full capacity for a fraction p_f of any given year. Further specification of the number of ABC units, N_{ABC}, each with N_{BLK} target-blanket modules of the kind depicted in Fig. 4, defines the system. The choices of N_{ABC} and N_{BLK} have both developmental, economic, operational, and safety implications. The number of ABC units is dictated largely by the maximum accelerator capacity, P_B(MW), and the maximum amount of electrical power to be delivered to the grid node by that unit, P_E; typically, power economics suggests $P_E \gtrsim$ 1,000 MWe, and (present-day) capacity limits suggest $P_E \lesssim$ 1,500 MWe. For P_E in this range, the number of target-blanket modules, N_{BLK}, is set by target power-density limits, target efficiency, (*e.g.*, neutron coupling, parasitic absorption) in driving a blanket with a given k_{eff}, (passive) safety and local (radioactive) inventory considerations, and cost[14,15].

The basis for the choice of target-blanket module size, P_F/N_{BLK}, in past ABC designs[1] was set by limitations imposed by the use of solid targets. These earlier ABC designs for a given value of N_{ABC} suggested ~500-MW modules and a number, N_{BLK}, of such modules. A large number of smaller modules (with the attendant potential for higher cost) sandwiched between large accelerator and balance-of-plant systems economically and operationally may not be optimal[14,15]. Furthermore, if nuclear and afterheat safety can be provided by a quick exit of fuel salt to a dump tank (Figs. 3 and 4), the blanket power capacity should not be limited by a desire for passive removal of decay heat from an otherwise unperturbed blanket. In this case, the module size could be set by blanket-criticality, target-power-density, cost, and/or other constraints (*i.e.*, scaling of developmental or prototype power increments). Given that limits imposed by target power density can be pushed upward through the use of flowing, self-cooled, liquid-metal (Pb or the lower-melting Pb-Bi eutectic) target, the thermal power per target-blanket module, P_F/N_{BLK}, can be increased from ~500 MW to values as high as 1,500-2,000 MW[7].

The magnitude of the module power, at this point in the conceptual development of ABC, is not as important as is the existence of a clear logic for determining it, as long as the module power is not too small. The following "traceable, but not unique" selection process based on the DOE guidance[4,5] in this area is used:

- The (disposition) technology shall be demonstrated in 20 years.

- A total of M_{Pu} = 50 tonnes of weapons plutonium will be disposed in T_{LIF} = 50 years; this suggests a burn time of 30 years; a more-aggressive 20 years has been adopted, which portents reduced life-cycle costs[14].

- The life-time average plant availability or capacity factor is $p_f = 0.75$.

- Given that the fissioning of 50 tonnes of plutonium will generate 128 GWyr of thermal energy (assuming complete fissioning), at $\eta_{TH} = 0.40$ thermal-conversion efficiency and a $p_f = 0.75$ plant availability, the electrical-power generation would be $N_{ABC}\,P_{ET}$ = 3,413 MWe. Furthermore, given that this power should be available in $P_E \sim$ 1-GWe chunks, $N_{ABC} \simeq 3$ such ABC units are suggested, each generating a total electric power of P_{ET} = 1,138 MWe [P_{TH} = 2,844 MWt, P_E = 967 MWe(net) if the recirculating-power fraction can be held to $\varepsilon = 0.15$; P_E = 1,063 MWe(net) of the MSBR value of $\eta_{TH} = 0.44$ is used].

- While economic consideration would favor only a few core modules operated at each of the three 967-MWe(net) ABC facilities, presumed limitations on target power density, safety, and/or reliability suggest a greater number of modules. Following the MSBE → MSBR scaling philosophy (25%, or one coolant loop)[10] and assuring that the modularization does not become too fine for reasons of lost economies of scale and cost[15], N_{BLK} = 4 modules at 2,844/4 = 711 MWt is adopted by this ABC Plant Layout Study.

It should be emphasized that the basecase ABC plant layout used in this study presumes the complete fissioning at nominally constant beam and fission power (*e.g.*, constant k_{eff}, increasing plutonium blanket inventory) of M_{Pu} = 50 tonne of weapons-grade plutonium. Although the process of final "burn down" is not considered by this study, unless highly enriched uranium is introduced near the end of life (EOL) to maintain constant power, only ~90% burnup of the original plutonium inventory is possible (Appendix E), and in fact ~56 tonne of weapons plutonium would be processed.

2. Top-Level ABC Design Options and Basecase Focus.

The main goal of the ABC Plant Layout Study is to translate the systems diagram embodied in Fig. 4 into a "strawman" plant layout using as guidance and as much as is appropriate the subsystem engineering details reported for the MSBR. Nine "top-level" ABC subsystems can be identified from Fig. 2: Accelerator (ACC); Target (TAR, including the Window, WIN); Blanket (BLK); Primary Heat Transport (PHT); Secondary Heat Transport (SHT); Power Conversion or Balance of Plant (BOP); Chemical Plant Equipment (CPE); Operations and Maintenance (O&M); and Safety (SAF). While general, this subdivision is not unique, nor are subsystem boundaries without diffusiveness. For the ABC Plant Layout Study to proceed in the spirit described above (*e.g.*, without a self-consistent and/or optimized preconceptual design), key choices must be made for each of these nine (ten if the window is considered separately) ABC subsystems.

Figure 5 lists for each of these subsystems important "top-level" design choices and the decision path taken to arrive at the base case used to generate the "strawman" plant layout described in Sec. III. The branching options listed on Fig. 5 for each of the main ABC subsystems are not all-inclusive, but many of the design decisions leading the the base case are represented. The reasons and rationale for the choices made, when they can be quantified, are elaborated in each respective subsection in Sec. III. Of equal importance are the "paths not taken" for each subsystem design decisions depicted on Fig. 5; these alternatives will emerge as part of the identification of key issues, the related prioritization of issues, and the identification of alternative design choices that may lead to improved ABC systems, as is addressed in Sec. IV.

III. ABC BASECASE PLANT LAYOUT

A. Overview of Basecase Layout

The design philosophy used to develop the ABC plant layout is based on the use of the Molten Salt Breeder Reactor (MSBR) conceptual design[10] with a minimum of modification. The design changes have been limited to only those necessary to accommodate the accelerator target, to update for inclusion of modern regulatory requirements, and to include minor design improvements. The logic behind this approach is predicated on the exploitation of the careful work of two decades ago that led to the development of the MSBR design. Furthermore, the MSBR concept is the last, and hence latest, molten-salt reactor design available. Because of limited resources available for the ABC Plant Layout Study, a commitment of similar magnitude as given to MSBR was not possible. Although a number of potential improvements (*e.g.*, substitution of an alternative secondary coolant for the sodium fluoroborate) were considered, these options are left as potential design options and not included in the base case.

Table II gives a "top-level" breakdown of key ABC molten-salt (MS) subsystems described in the following subsections. This inventory list, while more extensive than can be resolved by this ABC Plant Layout Study, provides a mechanism for generating an aggregation into key subsystems to be include explicitly in the study. Accordingly, the plant has been categorized into eight major subsystems: Accelerator, Target, Primary System (mainly the PHT subsystems), Balance of Plant, Chemical Processing, Operations and Maintenance, Instrumentation and Controls, and Safety [mainly I&C and Containment Systems (CS)]. As is indicated on Table II, in some cases a clearly defined boundary between these main subsystems does not exist, and, in general, interfacial issues can be important. The following subsections give each design basis and/or rationale base on quantitative scaling information derived largely from earlier accelerator and target/blanket studies and from the MSBR conceptual design. Many of the scaling relationships derived from the MSBR and applied to size ABC components are given in Appendix A; the accelerator scaling, *per se*, is described briefly and heuristically in Appendix B.

B. Main Subsystem Descriptions

A brief description of the main ABC subsystems is given in this section. Key dimensions and capacities, as they relate primarily to the plant layout, are collected in Table III. This table is intended to provide a collection point or parameter "depot" for the ABC Plant Layout Study, and, in terms of completeness and/or self-consistency, should not be considered an ABC design table *per se.*

1. Accelerator (ACC)

a. Overview

The essential elements of the accelerator system needed to provide the design, steady-state current to each of N_{BLK} ABC targets in a spatial distribution that meets both target power density and blanket neutron flux requirements are illustrated in Fig. 6. This figure indicates the technology development required to deliver the linear proton accelerator needed by

ABC from the present or near-future LAMPF device[16,17]. In addition to being uniquely suited for delivering high proton currents (~100 mA) at the requisite energies ($\gtrsim$ 600 MeV), the linear accelerator (Linac) adopted for ABC and embodied in LAMPF has the highest efficiency for converting "wall-plug" AC power, P_{EA}, to beam power, $P_B = I_B E_B$ ($\eta_A = P_B/P_{EA} \sim 0.5$), as well as exhibiting the lowest beam-loss factor [$< 2\times10^{-7}$/m for most coupled-cavity Linacs (CCLs)].

The Injector System (IS, Fig. 6) consists of duoplasmatron, duopigatron, or electron-cyclotron-resonance-heated (ECRH) volumetric ion sources that are capable of steady-state proton currents of >500 mA. The proton beam is extracted from the ion source at $\gtrsim$100 keV for injection into a Radiofrequency Quadrupole (RFQ) accelerator that bunches and accelerates the proton beam to 2.5 MeV. The bunched proton beam emerging from the RFQ is then accelerated to ~20 MeV by a Drift-Tube Linac (DTL). The LAMPF uses an older technology based on Cockroft-Walton injectors that feed a 100-MeV DTL. The DTL was invented a half a century ago, and this well-understood and well-developed machine has since been used on all high-current accelerators. After a transition and matching section, or in the case of ABC a FUNneling (FUN) and Bridge-Coupled Drift-Tube Linac (BCDTL), the proton beam emerging from the injector system described above (ion source, RFQ, and DTL) enters a Coupled-Cavity Linac (CCL) developed at Los Alamos in the 1960s for efficient acceleration of protons to energies >100 MeV. More recent consideration has been given to a Coupled-Cavity Drift-Tube Linac (CCDTL) as a replacement for the BCDTL matching section of the CCL Front End (FE) injector. In addition to efficient, higher-energy, and high-current capabilities, the CCL accelerating structure is simple and rugged; Fig. 6 gives the number of RF cavities (cells) and lengths for the CCLs used for LAMPF and anticipated for ABC. Other subsystems that make up the accelerator include the RF power supplies and distribution systems, vacuum systems, cooling, beam diagnostics, control and instrumentation, High-Energy Beam Transport (HEBT) systems for beam delivery to the target, and Beam Expander/Spreader (BES) systems to assure proper beam-on-target distributions for reasons of both assuring target longevity and optimizing blanket neutron flux intensity and distribution.

Figure 6 also indicates both the essential elements of the linear proton accelerator and advances in design and performance required in progressing from LAMPF[16-18] and the ATW/ABC[1-3,14,19,20]. These accelerators do not provide a continuous current of protons to the spallation target, but instead deposit a sequence of proton "bunches", each contained in the bottom of an RF electromagnetic potential well. In addition to the degree to which each RF wave is filled, the time-averaged intensity of protons delivered to the target is determined by the fraction of the time that the RF wave-train is on (*i.e.*, duty cycle) and the spacing within a given RF wave train between RF waves that actually contain protons and thoses that are empty. Hence, the increased current required of the ABC facility can be achieved by increases in: a) the degree to which each RF wave is filled with protons (LAMPF presently is ~25% "filled" in this regard); b) the fraction of the time when a packet of RF-waves will be found (LAMPF presently has a duty factor of ~6%, not to be confused with availability, which for LAMPF is ~85%); and c) the fraction of RF-waves within a given packet that actually carry or "push along" a proton bunch (for LAMPF 25% of the RF cycles actually contain proton bunches). By filling each RF electromagnet well to the "brim", by filling all of time with a continuous train of RF waves, and by using each the of these continuous RF waves with beam bunches of ~2×10^9 protons/bunch (ppb), the

the accelerator current can be enhanced by a factor of ~250 over the E_B = 800-MeV LAMPF[16,17], as is indicated on Fig. 6. The means by which the LAMPF current can be increased by the requisite factor to meet ABC needs remains primarily an issue of cost, schedule, and the accommodation of the range of uses projected for a higher-power LAMPF[20], rather than the longer-term technology developments required to achieve full ATW conditions. The key technical issues of a high-power ATW proton linear accelerator[18-20] adopted for the ABC Plant Layout Study include:

- funneling of two single beams into the last accelerating stages.
- beam loss along the acceleration chain leading to unacceptable heat loads on and activation of accelerator structures.
- efficiency and reliability of high-power RF power supplies.
- RF operational control at high beam loadings.
- (beam) fault recovery and other off-normal conditions (*e.g.*, RF-power and AC-grid surges; CCL module failure; beam failure; events driven by HEBT, BES, or window/target/blanket malfunctions, *etc.*)
- component reliability and accelerator maintainability.

Issues of lesser importance and concern for the ABC accelerator include: RMS beam physics, peak current levels, beam brightness, beam stability, accelerating gradients, thermal loads, and RF power sources. Table III lists key accelerator parameters anticipated for the ABC, and when possible value ranges are given; a main goal of any subsequent conceptual design is to complete Table III on the basis of optimized cost, schedule, and risk. An approximate accelerator scaling relationship is developed in Appendix B to given an example of the kinds of tradeoffs needed to complete an ABC accelerator "strawman"-design table for used in subsequent conceptual design studies.

b. Target Interface

The proton beam, upon achieving full energy and undergoing splitting into beamlets for use in each 711-MWt Target/Blanket module, is carried to the secondary containment building by the High-Energy Beam Transport (HEBT) system. After passage horizontally through a Main-Beam Isolation Valve (MBIV, Fig. 2), the ~ 800-1,000 MeV, $I_B/N_{BLK} \simeq$ 20-mA beamlet must be bent downward 90° and decreased in current density by means of a drift-tube beam expander/spreader (BES). Bending would occur by passage through a horizontal magnetic field. The optimization described in Appendix F suggests a bending radius of 2.8 m and a magnetic field intensity of 1.6 T. The vertically directed beam would be transported through a field-free region of length L_{EXP} = 10 m, where the beam space charge is expected[21,22] to enlarge the beam to an acceptable footprint (??? × ??? m, ???? A/m^2) at the Window/Target. While the economics of the beam bending and expansion *per se* (Appendix F) does not appear to be an important driver, the impact on the size and cost of the secondary containment building, as well as the impact on the Target/Blanket (vertical) maintenance scheme, can be significant.

2. Target (TAR)

a. Overview

Figure 7 is a schematic of the required target function. The target converts the high-energy ($E_B \gtrsim 500$ MeV) protons generated by the accelerator to lower-energy (< 20 MeV) neutrons and transports these primary neutrons to the blanket. In performing this function, the target must be cooled sufficiently for steady-state operation and must be designed to reduce risk related to radiological release or other detrimental consequences resulting from off-normal operating conditions (*e.g.*, loss of target coolant, maladjusted beam distribution, *etc.*).

As elaborated in Appendix C, the conversion of high-energy protons to neutrons relies on intranuclear reactions between the incident protons and the nucleons in the target material. Consequently, to maximize neutron production, the target material should have a large number of nucleons per individual nucleus; a high-Z material is preferred. While the protons can interact directly with bound neutrons followed by ejection from the nucleus, the emitted neutrons tend to be peaked forward and to have very high energies (*e.g.*, in the range from 20 MeV to E_B); these neutrons are poorly used in the blanket, since the slowing-down length is large, even for efficient neutron-moderating materials, and heavy shielding behind the target is required (Fig. 7). Neutrons of more utility to a moderating, thermal-neutron blanket are created through the interaction of a proton with the nucleons in a nucleus in general, thereby leaving the nucleus in an excited state after interacting with the proton. Release of excess energy in the excited nucleus occurs by nucleon evaporation; a substantial portion of these evaporated nucleons are neutrons. These evaporation neutrons are emitted isotropically with an average energy in the range 1-2 MeV. The number of neutrons generated by a given proton energy depends on the target material.

While maximizing the generation of low-energy neutrons is important to the overall efficiency of ABC, the ultimate target performance depends on an ability to transfer usefully and efficiently these neutrons to the blanket. This efficiency depends on the neutron absorption characteristics of the target material(s) and the volumetric distribution over which the neutrons are generated (*i.e.*, whether the target produces a highly-peaked, intense neutron distribution, or whether the distribution is more evenly distributed; the volume required to achieve maximum neutron production for a given neutron-source distribution is also important).

The target absorption characteristics depend not only on intrinsic nuclear parameters, but also on the amount of thermalization that occurs in the target. The degree of thermalization in turn is strongly dependent on the type and quantity of coolant used, as well as the target geometry and configuration. Similarly, the neutron-source distribution also depends on the target material (density), coolant fraction (*i.e.*, the "effective" target density), and geometry. The ABC target design, with an overall goal of achieving high thermal neutron fluxes in an acceptable blanket volume V_{BLK} with a minimum accelerator capacity P_B, therefore, will have to optimize neutron production to minimize neutron absorption in the target; to maximize neutron leakage to a blanket of a size that is acceptable for engineering purposes; and to distribute the source as evenly as is possible over the volume of interest. Achievement of the first three goals also leads to a need to minimize the coolant fraction.

The ability of the ABC target to achieve the functional goals described above depends on three specifications: a) target material, b) target geometry, and c) target heat-removal system. All three specifications are interdependent, however, and this interrelationship must be fully understood before an effective and optimal ABC target design can be realized. Each target technical issue is discussed in Appendix C, which gives a broad technical perspective of the ABC target requirements and options.

b. Target Components

Window: The accelerator window is a crucial component in the ABC system. In the current ABC design, the window is an integral part of the target structure and is cooled by the flowing liquid-lead target material. A window failure, therefore, would cause significant downtime for cleanup of the accelerator vacuum system that would be contaminated by the lead. Because the window is cooled solely by the lead, which operates at high temperature, it must maintain strength at high (~1,000°C) temperatures. This requirement, coupled with the the need to endure a large proton and neutron fluence without serious degradation to mechanical properties, makes the material choice problematic and difficult without extensive experimental investigation. Alternate proposals exist for possible window configurations that attempt to remediate the high-temperature requirement by providing the window with a separate coolant other than the lead. This possibility and associated benefits and disadvantages is discussed in Appendix D.

Lead Cooling System: The flowing lead can easily remove the heat deposited by protons, neutrons, and gamma rays, but removal of the heat deposited in the target structure is more complicated. Generally, recovery of this power at temperatures where efficient conversion to electrical power is possible is not being considered; the beam power will be rejected to the atmosphere as low-grade heat. Lead, like other heavy metals (*e.g.*, mercury and bismuth) does not wet containment materials well. The inability to wet container surfaces causes a significant decrease in the obtainable heat-transfer coefficients, as well as causing difficulty in predicting the lead flow distributions near structural surfaces. This uncertainty generates a requirement for a large degree of experimental validation for any flowing heavy-metal target designed for the target heat fluxes and target heat fluxes and power densities (????MW/m^2, ????MW/m^3) envisaged for ABC. Another important issue with regard to the lead cooling system is the choice of secondary coolant and the associated heat-exchanger design. The secondary (target) coolant presently being considered is NaK, although some industrial cooling salts (*e.g.*, HiTech) are also under investigation. The main requirements for the secondary coolant is compatibility with the lead, in the event of a heat-exchanger leak, and operation at a low pressure while maintaining compactness in the heat-exchanger design. The poor wetting characteristics of liquid lead make the heat-exchanger design another prime candidate for experimental validation.

Lead Freeze/Thaw System: The use of a liquid metal for the neutron-producing (neutron-spallation/evaporation) target generates the requirement for an additional system for melting and freezing the lead material. As with most materials, lead expands upon melting and contracts while freezing. If the phase change is allowed to occur within the target system, damage would likely occur to the structural containers (especially in the thin-walled heat exchanger tubes) because of the additional stress that accompany the phase change, which cannot be accurately controlled. A lead storage container or reservoir, therefore, is provided to accommodate phase changes. A free surface is maintained in the reservoir, and spatially dependent heaters would be used to control the melting process.

The use of this reservoir, however, means that the lead must be maintained as a liquid while residing within the main target system, even if the accelerator beam is off. Heaters, therefore, are required to be applied on the target structure are required. The use and survivability of these heaters in the high radiation environment of the target may present a key design issue. Also, an injection/drainage system must be used to transfer the lead to and from the reservoir. The transfer medium is envisioned to be an inert gas (*i.e.*, argon) pressurization system, like those used in liquid-metal fission reactors[23].

Lead Cleanup System: Generally, lead is highly corrosive to most materials, especially in a flowing environment. The slow addition of a large number of additional chemical species that are generated through the nuclear spallation/evaporation process, adds uncertainty to the expected rates of corrosion and, therefore, uncertainty to the target structural lifetime; once again, this issue raises a need for experimental efforts to resolve the uncertainties. At some point during the ABC operation, the lead may become unusable because of extensive contamination from nuclear spallation/evaporation products, which could affect fundamental thermodynamic properties the neutron-producing ability. For the conditions envisaged for the ABC target (800 MeV, 20 mA/target), the rate of lead destruction and "impurity" injection amounts to ~ ???? kg/yr, or ~??? %/yr of the active lead inventory. If this level of contamination proves unacceptable, the lead will either have to be replaced and, therefore, contributes to a (mixed) waste stream, or the lead would have to be cleaned and recycled. No processes have been identified to clean up the lead, and if needed, will require a design and development effort.

3. Primary System

As indicated on Table II, the primary system consists of all components located inside the primary vessel (core), the intermediate heat exchangers (IHXs), the fuel salt pumps, and all the primary system piping that interconnect these components. In the parlance of the EEDB Program of Cost Accounts,[12,13] the Primary System is essentially the Reactor Plant Equipment. For the purposes of the ABC Plant Layout Study, this system is approximately defined by those components that contain an appreciable quantity of fuel salt, with the exception of the drain tank and the chemical-processing equipment. Modified Hastelloy-N is used for the entire primary system because of its compatibility with the fuel salt. This fuel-salt boundary is comparable to the fuel cladding in a conventional fission reactor, and is identified as the primary containment boundary (Fig. 4). The compatibility issue was developed on the basis of the MSBR design experience (UF_4, heavy ^{233}Th loadings); the plutonium-based salt is expected to be substantially different for ABC, especially with regard to REDOX potential. Each of the major Primary System components is described in detail below, and is shown in Fig. 8 as: Core; Fuel-Salt Pump; Intermediate Heat Exchanger; and the Reactor-Cell Vessel and all associated piping.

a. Core

The ABC core corresponds to the MSBR core in size and composition, aside from the central spallation target. As indicated on Table II, the Core consists of Target/Blanket decoupler, blanket coolant (*i.e.*, the fuel salt), the Moderator, the Reflector, the Reactor Vessel, and all control/shutdown rods. For the purposes of the ABC Plant Layout Study, the Core has been designed with an overall power density of P_F/V_{COR} = 22.2 MW/m^3, which is equal to that of the MSBR. The overall size of the core shown in Fig. 8 was determined from this power density and the basecase overall thermal power per

Target/Blanket module of P_F = 711 MW. For a square cylinder core shape, the core diameter and height are 3.5 m. These dimensions compare to the MSBR core diameter and height of 5.2 m and 4.0 m, respectively.

The Core internal structure is similar to that of the MSBR. Graphite blocks or stringers are used to moderate the neutron flux and to form flow channels for the fuel salt. Each stringer is 0.10 m on a side and has a 0.034 m hole drilled through its center. Fuel salt flows both inside and between adjacent stringers. A total of 962 such graphite stringers will be required for each ABC Target/Blanket assembly. Although incomplete, parametric neutronics studies of other graphite/fuel-salt configurations indicate an important trade off between material lifetime, quantity of nuclear waste, and operational complexity as the fuel-salt/moderator ratio is varies; this material is summarized in Appendix E, which also lists the damage rates in the graphite for a number of moderator/fuel-salt ratios.

Of the 962 graphite stringers, six are non-standard: three are designed to accommodate graphite control rods and three are fitted for boron carbide shutdown rods. The principle of the control-rod action is based on the displacement of fuel salt to adjust reactivity. These control rods are inserted to start up each driven target-blanket assembly, since the surrounding regions are undermoderated; by displacing fuel salt and introducing additional moderator, reactivity is introduced. These control rods must be removed to reduce the reactivity. Although this introduces a potential failure mechanism, because of density differences the graphite tends to float in the fuel salt unless constrained. Electromagnetic control-rod drives may be used so that in the event of an electrical failure the control rods float out of the core and reduce the reactivity. In addition to the control rods, three shutdown rods are included. These rods are composed of Hastelloy-N-clad boron carbide and are inserted to reduce the reactivity. By including these shutdown rods, the effects of an inadvertent accelerator start-up are mitigated. The shutdown rods would normally be fully withdrawn during operation.

The active core region is surrounded by a 0.75-m-thick graphite reflector. This thickness was used in the MSBR conceptual design. Although the ABC core is smaller, a similar reflector thickness was chosen. The blanket-vessel inner diameter, therefore, is 5.0 m.

b. Fuel-Salt Pump

The fuel salt moves upward through the core at a nominal velocity or v_{FS} = 0.88 m/s (ΔT = 139 K, $\dot{M}_{FS}$ = 2,150 kg/s, fuel-salt volume fraction f_{FS} = 0.13) and enters an upper plenum located between the core and the reflector. The flow is divided at this point and is passed through the radial reflector by two flow channels machined in the graphite. Two identical loops primary are used to transfer the fission heat in the fuel salt to the secondary coolant. Each loop consists of a fuel-salt pump, an IHX, and the associated piping. The fuel-salt pump design was adopted from the MSBR fuel-salt pump design[10] without change. Although the pumping requirements of the two systems differ (1.0 m^3/s for the MSBR versus 0.55 m^3/s for the ABC), the pump size has been taken to be the same as that of the MSBR. Guidance on scaling the pump size was not found, nor could it generated within the scope of the ABC Plant Layout Study; the direct adaptation of the MSBR pump design results in a conservative size allowance in the layout.

The fuel-salt pump is of the centrifugal sump-pump design and is illustrated in Fig. 9. The pump bowl is 2.0-m in diameter and is 1.5-m high. A free surface is maintained in the pump bowl because the pump bowl serves as a surge volume for the entire Primary System. Because a free surface is maintained in the pump bowl, the pump must be placed in elevation above all the Primary System components. Graphite blocks may be positioned around the pump impeller to limit the fuel salt volume held up exterior to the blanket. The fuel-salt volume in each pump has been estimated to be 1.0 m^3 or 8% of the total fuel-salt volume.

The pump motor is located on the maintenance floor several meters above the impeller. This provides ample room for shielding the motor from the intense radiation field at the level of the pump bowl. Pumps of this kind were operated for many thousands of hours as part of the MSBR project. Molten-salt pumps of this large capacity, however, have never been built. It was the consensus of the MSBR project that scale up of the pump design would not be difficult.

c. Intermediate Heat Exchanger

Fuel salt flows directly from each fuel salt pump into the associated IHX, as is shown in Fig 8. The IHX design is adopted from the MSBR design. The IHX is a shell-and-tube heat exchanger with a somewhat unconventional internal arrangement to accommodate remote maintenance and to limit the exo-blanket fuel-salt inventory. As is shown in Fig. 10, the IHX dimensions are nearly identical, with the exception of the height. Modifications to this design were limited to the following items: a) shortening the tubesheet-to-tubesheet distance from 7.07 m to 5.25 meters; and b) converting the concentric secondary-salt outlet pipe into a more-conventional side outlet. The size, number, and spacing of tubes remains unchanged. The fuel salt enters through a vertical tubesheet. After traveling through the tubes, each of which assumes the shape of an inverted "L," the fuel salt flows down to the lower horizontal tube sheet. The secondary coolant salt side of the IHX was subjected to design modifications, but these modifications are described more fully in the Balance of Plant description (Sec. III.B.4.).

As is shown in Fig. 10, the IHX has an overall height of 6.55 m and a shell diameter of 1.75 m. The shell contains a central downcommer with a 0.51-m diameter. Surrounding this downcomer are 5,803 tubes arranged on a 19.1-mm pitch. Each tube has an outer diameter of 9.5-mm. The tubes are bent into a sinusoidal configuration in the upper portion of the IHX to accommodate thermal expansion. Over the remainder of their length, the tubes are knurled in a spiral pattern to enhance the overall heat-transfer coefficient.

d. Reactor Vessel and Primary System Piping

The reactor vessel is fabricated from modified Hastelloy-N alloy. The inner diameter is 5.0 m and a wall thickness is 50.8 mm. The vessel has a maximum height of 5.0 m at the center. Both the top and bottom heads are spherical, with a 16-m radius of curvature. The upper head is removable to allow replacement of the graphite stringers and inspection of the reactor internal components. The upper-head design is complicated by the need to accommodate the (removable) the target thimble. The reactor vessel is similar in design and shape to that of the MSBR. The ABC vessel is not as high and has a smaller diameter. A remote flange was used in the MSBR top-access design to lower the temperature and neutron flux on the upper head connections. A similar arrangement is expected for the ABC, although this detail is not shown in Fig. 8.

All of the primary system piping is made of modified Hastelloy-N. The piping used is 0.40 m in diameter with a 12.7-mm wall thickness. For a mass flow rate of 1,073 kg/s, the flow velocity in the primary system piping is 4.9 m/s, which is the maximum flow velocity in the primary system. The other flow velocities in the primary system are 0.88 m/s in the core, and 2.0 m/s inside the (5,803) IHX tubes.

Hastelloy-N alloy was chosen for all Primary System components because of the experience with molten-salt compatibility. This adequate compatibility, however, was developed on the basis of the MSBR design experience (UF_4, heavy ^{232}Th loadings); the plutonium-based ABC fuel salt is expected to be substantially different, particularly with respect to the REDOX potential. None of the other potential materials has undergone as extensive testing with fluoride salts. One possible exception to the use of Hastelloy-N is in the reactor vessel portion of the target thimble, however. The thimble will be exposed to a large neutron flux (???? /m^2/s), and is expected to have short lifetime (??? months). If the radiation resistance of Hastelloy-N is insufficient to provide at least a one-year operational lifetime, an alternate material may have to be used. Modified 9Cr-1Mo ferritic alloy has been considered for this application because of a superior irradiation performance. The molten-salt compatibility of this alloy, however, is not as good as that of Hastelloy-N, but its corrosion lifetime in molten-salt may prove to be greater than the Hastelloy irradiation lifetime in the high-flux region of the target thimble.

4. Heat-Removal Systems

The Heat-Removal systems (Table II) include the secondary coolant system and its associated equipment, the steam generator, the steam reheater, and the supercritical-steam (SCS) power-conversion system. All of these systems may be considered to comprise the BOP and corresponds in large part to the designation often attributed to the non-nuclear portion of nuclear power plants. The intermediate coolant loop is not found on current generation light-water fission reactors (LWRs) and is, therefore, somewhat difficult to characterize, although detailed designs for the Liquid-Metal Breeder reactor[23] (LMBR) are applicable here. While the secondary coolant will contain appreciable radioactivity as a result of activation of the coolant in the IHXs, it should contain neither fuel nor fission products unless leaks occur in the IHX tubes. The secondary salt loop is included in the ABC design for the same reason it was incorporated into the MSBR design: primarily to increase the overall system safety margin.

a. Secondary-Coolant System

A secondary coolant system is incorporated into the ABC design because, first, the secondary salt loop helps meet the three-barrier requirement for containment of fuel salt. Secondly, this loop reduces the probability of transporting fuel or fission products into the turbine and related equipment in which radioactive material containment cannot be accommodated. Lastly, the secondary loop reduces the chance of fissile material precipitation by reducing the probability of steam ingress into the primary system. For all the potential benefits, however, the secondary system is not without drawbacks related primarily to added system complexity, reduced overall conversion efficiency, and added cost.

The secondary salt chosen for the ABC is taken from the MSBR design and is a sodium fluoride, sodium-fluoroborate eutectic mixture. This coolant is commonly referred to as sodium fluoroborate, with the assumption of an eight percent sodium fluoride addition. Extensive testing was performed on this salt as part of the MSBR project. Sodium fluoroborate combines good heat-transport and fluid-flow properties with low cost, acceptable chemical and radiation stability, and compatibility with Hastelloy-N. This material is not an ideal choice for a secondary coolant, however, but its combination of advantages was determined to outweigh its disadvantages.

Most of the drawbacks to use of sodium fluoroborate are well known and were studied extensively as part of the MSBR program. One of the concerns is the requirement for a cover gas. Sodium fluoroborate undergoes a thermal decomposition that evolves BF_3. This gas must be reintroduced, along with an inert cover gas, to prevent changes in the $NaF\text{-}NaBF_4$ ratio. The mole fraction of NaF must be controlled to prevent gross changes in the fluid properties of the eutectic mixture. The BF_3 evolved is a chemical hazard, but compared to the other chemical and radiological hazards associated with a molten-salt system, this concern is minor. The off-gas system for the secondary salt loop will be more complex than it would be if an alternative salt were chosen, but this complication is not sufficient justification for use of a less characterized salt.

Another potential problem associated with the use of sodium fluoroborate is its corrosiveness when contaminated with water; minor steam leaks into the secondary loop may not be tolerable. In the absence of water, the corrosion rate of Hastelloy-N in sodium fluoroborate has been shown[10,11] to be approximately 5.0 µm/yr. This rate increases dramatically to over 500 µm/yr in the presence of water. It may prove impractical to prevent water ingress into the fluoroborate by way of the steam-generator and steam-reheater tubes (Fig. 3), and the moisture removal capability of the off-gas system is limited. The present design does not use duplex tubing in either component, so in-leakages are expected to occur over the lifetime of the plant. Large leaks would require shutdown and salt cleanup. Pinhole leaks, however, may be sufficient to accelerate corrosion and can reduce the secondary-loop component lifetimes. This issue must be accommodated in the detailed design either through the use of duplex tubing, more aggressive moisture removal equipment, or conservative design choices with regard to equipment wall thicknesses.

The feedwater temperature requirement is 56 K lower than the alternative secondary coolant salt, $LiF\text{-}BeF_2$. The liquidus temperature of the sodium fluoroborate is 658 K, compared to 732 K for the $LiF\text{-}BeF_2$. The feedwater requirements of the reference steam system are already sufficiently high to require a complex feedwater design[10]. Additional increases in the minimum feedwater temperature are not justified by the reduced complexity of the secondary-salt system if $LiF\text{-}BeF_2$ were to be used.

Sodium fluoroborate traps tritium gas leaking or diffusing into the secondary loop from the primary loop. Limited testing [Ref. 24, p. 57] has shown that a large fraction of the tritium that reaches the fluoroborate can be trapped and removed before diffusing into the steam system by way of the steam generator and steam reheater. The tritium is converted into a chemically combined and water-soluble form (????), and then removed by the off-gas system. Greater than 90% of the tritium added under steady state conditions was trapped in the limited tests that were performed. The actual mechanisms responsible for this trapping, however, are not understood.

The secondary-coolant-salt system consists of the shell side of the IHX (Fig. 10), the shell side of the steam generator, the shell side of the steam reheater, two coolant salt pumps, and the associated piping. Figure 11 shows two view of the steam generator, Fig. 12 depicts the steam reheater, and the secondary-coolant-salt pump is similar to that illustrated in Fig. 9 for the fuel salt. The coolant salt enters the IHX through a central inlet located at the top of the IHX. The salt then flows down through a central 0.51-m-(outside)diameter downcomer and into the lower tubesheet. The flow is directed outward and flows upward and past the tubes. A number of disk- and donut-shaped baffles are included in the shell to increase the overall heat transfer coefficient. The primary modification to the MSBR IHX design to accommodate the ABC application occurs in the top of the shell, where the concentric coolant salt outlet connection has been replaced by a more- conventional plenum and outlet through the side of the shell. The IHX shell measures 1.71 m in diameter, with a thickness of 12.7 mm. The IHX is 6.55-m high, which is slightly shorter than the 7.32-m height of the MSBR design.

The heated secondary coolant salt flows from the two IHXs and is combined into a single pipe for passage through the vessel that forms the reactor cell. This pipe leads directly to the steam-generator cell. The coolant-salt flow splits before satisfying the steam-generator and steam-reheater loads. The steam generators (Fig. 11) and the steam reheaters (Fig. 12) are similar, with only the design pressures, sizes, and thermal capacities changing. In both heat exchangers, the coolant salt enters at 894 K and exits at 728 K.

In systems using more conventional coolants, division of the secondary-coolant flow between the two unequal loads would be accomplished by the use of flow control valves. The required valves, however, have not been developed for use in high-temperature salt. In lieu of the need to develop these valves, flow control in the ABC design, would be accomplished through speed control of the two coolant- salt pumps. In each of the coolant-salt loops, a pump is located directly downstream of the respective heat exchanger (*e.g.*, the steam generator or the steam reheater). Feedback from temperature detectors on both the coolant salt and steam sides should allow adequate flow control. The pumps designed for use in the MSBR utilized variable speed motors, and should be capable of the fine adjustment necessary for flow control.

The coolant-salt piping is constructed entirely of Hastelloy-N. Two sizes of pipe are used. Pipe of 0.51-m diameter is used for all the connections within the individual cells (reactor vessel and steam generator). The two flows join before passing between the two (secondary-coolant-salt and the steam-generator) cells to minimize the number of cell penetrations. This larger piping uses a 0.61-m-diameter pipe. The flow velocities in the small-bore pipes range from 1.2 m/s in the steam reheater piping to 7.8 m/s in the steam-generator piping. The velocity in the large-bore piping is 6.0 m/s. The coolant-salt piping sizes can be increased with little additional penalty if it is determined that these velocities are too high. The coolant-salt volume does not represent a critical issue, since the sodium-fluoroborate secondary salt is relatively inexpensive and the radioactive inventories are low.

b. Steam Generator

The steam-generator design illustrated in Fig. 11 is taken from the MSBR design; a few minor modifications were made primarily to optimize the layout for the ABC design. For the ABC layout, the inlet and outlet plena were changed, as were the overall physical dimensions. The overall principle of the design remains unchanged, however, as is also shown in Fig. 11. The MSBR design philosophy led to the use of many smaller steam generators and steam reheaters to minimize the required wall thicknesses for this (high-pressure) SCS system. The steam generator was sized for 121 MW, and the reheater was sized for 36.6 MW. Sixteen steam generators and eight reheaters were used for the 2,250-MW MSBR power plant. Each ABC core develops a much lower thermal power (P_{TH}/N_{BLK} = 711 MW), and it was determined that a single larger steam generator and reheater were preferable to multiple smaller units. A single steam generator was, therefore, designed to accept the entire power load of approximately 617(???) MW.

The MSBR steam generator used a U-shell, U-tube design to minimize the diameters of the inlet and outlet plena and their associated tubesheets. This configuration has been adopted for the ABC. The results of sizing calculations are a total length, including the inlet and outlet plena, of 7.3 m, a total height of 6.0 m, and a shell diameter of 1.5 m. Because of the high pressures in the steam side of the steam generator (up to 29 MPa), thick walls are required for the inlet and outlet plena, and for the tubesheets. The inlet and outlet plena are 0.25-m in thickness, and the tubesheets are 0.5-m thick. It may be possible to reduce the tubesheet thicknesses as part of the detailed design, because conservative stress calculations were used to determine these thicknesses.

One of the major changes to the steam-generator design was the relocation of the inlet and outlet plena. This change reduces the number of curves in the steam-generator shell and simplifies the piping layout. The U-shell steam generator is oriented on the side, as is shown in Fig. 11. Hot coolant salt enters the upper leg through the side of the shell. This salt passes along the tube bundle down to the lower leg, at which point it exits through the side of the shell. The feedwater enters the lower leg through the end of the shell. The inlet plenum is hemispherical, with the tube sheet forming the flat surface. The feedwater passes through the tubesheet into the 4,115 tubes. The resulting superheated steam exits through an outlet plenum that is identical to the inlet plenum.

The steam generator has not undergone detailed design, but significant difficulties are not expected. The use of a U-shell minimizes the shell diameter and allows the use of a single steam generator for the entire plant load. Testing of this design will be required prior to construction of the full scale version.

c. Steam Reheater

Steam reheat is standard practice in supercritical-steam systems to extract the maximum work from the high-pressure fluid. The MSBR steam-system design used full-flow reheat, and this approach has also been adopted for the ABC. The MSBR reheater design (Fig. 12) is based on a conventional shell-and-tube design that uses a cylindrical shell. Eight smaller reheaters were used, and each were sized to transfer 36.6 MW. As discussed in Sec. III.B.4.b., it was determined for the ABC application that a single large component was preferable to multiple smaller units. A single reheater, therefore, is used and sized to transmit 93.6 MW. The actual reheater design is a replica of the steam generator, as is shown in Fig. 12. This choice was made primarily to simplify the BOP layout. Use of a U-shell/U-tube design is expected to increase the cost of the reheater, however, but this penalty is outweighed by the simplification in layout that results from the mirroring of flow paths to and from the steam generator and reheater.

The reheater has a shell diameter of 0.8 m, a total height of 3.0 m, and a total length (including the inlet/outlet plena) of 6.0 m. The operating pressure in the reheater is much lower than in the steam generator (4.0 MPa *versus* 25.5 MPa), so the required wall thicknesses are much reduced. The inlet and outlet plena are 0.05-m thick, and the tubesheet is 0.2-m thick. As with the steam generator, these thicknesses are likely to be reduced during the detailed design process. The calculations used to generate these parameters are intended only to provide an upper bound for use in this design.

The uncertainties in this design are few and are primarily those noted for the steam generator. Testing performed for the steam generator will for the most part be applicable to the reheater. The only difference in the two designs, other than physical size, is the steam condition. The inlet reheat steam is to be preheated to 617 K, which is actually below the freezing point of the sodium-fluoroborate coolant. Additional testing beyond that required for the steam generator will be required to show that partial freezing of the secondary coolant salt either does not occur or does not cause difficulties if it does solidify.

d. Supercritical Steam System

The steam system chosen for the ABC was taken directly from the MSBR conceptual design. The MSBR steam system depicted in Fig. 13 was adapted from the Bull Run Steam Plant design[25]. The Bull Run unit is a high-efficiency, coal-fired steam plant that utilizes supercritical steam. The steam enters the high-pressure turbine at 811 K and 24.5 MPa. These conditions are adopted for the ABC.

Supercritical feedwater enters the steam generator at 25.9 MPa and 644 K. The feedwater is heated by the sodium fluoroborate coolant salt to 811 K. The steam passes through the high-pressure turbine and exits at 4.1 MPa and 561 K. The steam then passes into a reheat steam preheater that uses first quality supercritical steam to heat the reheat steam to avoid the salt freezing in the reheater. Steam leaves the preheater at 3.8 MPa and 617 K. The reheater brings the steam back to 811 K, the temperature at which it enters the intermediate pressure turbine. The steam passes through the intermediate-pressure turbine, through the low-pressure turbine, and into the condenser. A full-flow demineralizer is used to prevent fouling of the once-through steam generator and reheater. After passing through the demineralizer, the condensate enters the feedwater heater/booster equipment. A complex system of feedwater preparation is needed because of the high feedwater temperatures required to prevent freezing in the steam generator. The majority of the pressure increase is provided by steam-turbine-driven booster pumps. The final feedwater heating occurs in feedwater mixers that blend the high-pressure steam (23.8 MPa) from the reheat steam preheater with the feedwater. Electric feedwater booster pumps are then used to increase the feedwater pressure to 25.8 MPa prior to its introduction into the steam generator.

The supercritical-steam cycle described above is complex and costly. The choice of this cycle was driven by two factors: a) the high feedwater temperature required to prevent freezing of the secondary coolant salt in the steam generator; and b) the efficiency gained by going to a supercritical-steam cycle. It was determined in the MSBR design that the high feedwater-temperature requirement was best met by blending first quality steam out of the steam generator with feedwater prior to obtaining any mechanical work. Direct mixing of condensate with steam produces violent reactions from bubble collapse. This condition is averted in the supercritical-steam cycle because the two phases are indistinguishable. A simple spherical chamber is used for mixing. The second factor in choosing the supercritical-steam cycle is gain in thermal efficiency. The best subcritical-steam cycle devised was a modified Loeffler cycle, which yielded a net plant thermal efficiency lower than the reference supercritical cycle (41.1% *versus* 44.5%). The combination of simplified feedwater mixing and higher thermal efficiency outweighs the added equipment and design complexity of the supercritical-steam cycle.

5. Chemical Processing

Like the MSBR, the chemical processing equipment is an integral part of the ABC design. The MSBR has been described as a "chemist's reactor,"[10,11] because of the importance of chemical control and separations to the design and operation. The ABC may be constrained less by chemistry limitations, because breeding is not envisioned for the system and the neutron balance is considerably relaxed in a driven (subcritical) reactor (Table I). The primary mission of the MSBR was breeding of ^{233}U from ^{232}Th for an economically attractive doubling time. The ^{233}Pa produced as an intermediate step must be removed from the fuel salt before absorbing a neutron and being lost from the fuel cycle. Furthermore and of great importance to the somewhat tenuous MSBR neutron balance, parasitic absorptions must be minimized to maintain an acceptable breeding ratio and doubling time. The removal times for ^{233}Pa and certain parasitic fission products were short, and this imposed severe restrictions on the chemical processing equipment for MSBR.

While not as serious, chemical processing, nevertheless, remains important to the ABC design. Chemical processing is needed in ABC to prevent undesirable changes in the fuel solubility (because of buildup of certain fission products), to reduce material interactions (*e.g.*, the tellurium embrittlement of Hastelloy-N), to maintain fuel concentration within the desired narrow range, and to prepare the fission products for disposal. Additionally, the cost-driven necessity to use efficiently the accelerator-produced neutrons, the desire to limit actively circulating inventories, and waste minimization are important drivers of the chemical plant equipment design. While the chemical processing equipment is expected to be neither as numerous nor as large as was required for the MSBR, it is expected to occupy a large portion of the containment. All of the chemical processing equipment is described in the following sections.

a. Fuel Processing

Fuel processing for the ABC system can be divided into two parts: a) preparation of the initial 67 mol% ^{7}LiF – 33 mol% BeF_2 fuel salt; and b) preparation of the plutonium-containing feed material. The preparation of the 67 mol% ^{7}LiF – 33 mol% BeF_2 fuel salt has been described for the MSRE and MSBR, and the same kind of system is required for the ABC system. The details of the system have been described elsewhere[10,11]. The preparation of the plutonium-containing feed material and method by which the fissile material is to be introduced into the reactor, however, are described.

The plutonium feed for the ABC fuel salt be a ^{7}LiF – PuF_3 eutectic mixture. The process for preparing such a mixture follows. The excess weapons plutonium, which would require no preprocessing, would be converted to the trifluoride by hydrofluorination in the presence of a small amount of hydrogen (*i.e.*, probably <2% H_2) in the temperature range 500 - 600°C. The hydrogen prevents the formation of the tetrafluoride and the volatile hexafluoride. The PuF_3 would be mixed with ^{7}LiF in the ratio 19.5 mol% PuF_3 – 80.5

mol% ^{7}LiF and heated above the eutectic temperature of 743°C. The mixture could be cooled and stored for later use or injected in the fuel salt. If the mixture is cooled and stored for later use, several schemes exist for introduction into the reactor. The mixture could be preheated to 750°C and the liquid blended with the fuel salt in the pump bowl, or small pellets of the eutectic mixture could be added to the pump bowl where it would dissolve in the fuel salt. Fail-safe procedures would be implemented to guard against the introduction of excess reactivity at this point in the process.

In general, the processing equipment that is required for the plutonium feed material preparation consists of a two gloveboxes, each of which being fitted with two or three furnace systems and nickel reaction chambers that is fitted with gas handling capabilities. Details relating to the size and location of the gloveboxes, the furnace systems, and the reaction chambers have not been defined. Criticality safety issues would be important during all stages of the design.

b. Drain Tank

The drain tank is located below the reactor vessel in an isolated vault. This tank is sized to accept all the fuel salt, along with a fraction of the secondary coolant salt that might enter the primary system during an IHX tube failure. The drain tank serves several purposes, each of which is described. The drain tank primarily provides a safe storage for the fuel salt under conditions where heat removal is assured. The drain tank also provides a numer of other important functions that are primarily related to the interface between the primary system and the end-of-cycle chemical processing systems. The short-term holdup volume for fission product off-gases is also provide by the drain tank. In many ways, the drain tank is at the center of the chemical processing systems and rivals the reactor vessel in size, complexity, and importance to operations and plant safety.

The MSBR drain tank was sized to contain 70.8 m^3 of fuel salt. This volume was sufficient to contain all of the fuel salt volume plus an additional 45% contingency. The average power density corresponded to 46.1 MW/m^3 with respect to fuel-salt volume . The volume of fuel salt in the ABC is estimated to be 15 m^3 (47.4 MW/m^3), with about 30% of this in the core at any one time. Allowing the same contingency volume in the drain tank, the tank must have a storage capacity of 22 m^3. This corresponds to about one-third the volume of the MSBR drain tank. The height remains the same at 6.71 m. The diameter of the tank is 2.4 meters. The result is a tall, thin tank. Retaining the same height, however, allows the entire passive cooling system to be adopted from MSBR(???). A set of parallel LiF-BeF_2 circuits are used to remove the decay heat in the drain tank passively. The LiF-BeF_2 circuits are in turn cooled by parallel water/steam loops.

c. Off-Gas Handling

The MSBR design includes a complex off-gas collection system. The primary purpose of this system is to remove volatile fission products, the most important of which is ^{135}Xe, from the fuel salt to increase the breeding ratio. Because of the small margins available in thermal breeding, reductions in the parasitic neutron losses were essential for the MSBR design.

The MSBR off-gas system design incorporates three delay zones and a final cleanup system. The first delay zone was the gas space in the fuel-salt drain tank. The heat load from the volatile fission products was used to maintain natural circulation in the drain-tank heat-removal system at all times. After decaying for two hours in the drain-tank air space, the off-gas was transferred to a short-term delay bed that was designed to allow decay of the ^{135}Xe (9.1-hr half life). After a 47-hr holdup in this second zone, a portion of the off-gas was sent to the long-delay beds, where the decay of the longer-lived isotopes occurred. The remaining fission products were then separated from the carrier-helium in a trap and bottled for long-term storage.

A system with the complexity of the MSBR off-gas system is not needed for ABC because of the absence of the fissile-fuel breeding requirement. Furthermore, the accelerator is capable of supplying the additional neutrons would be absorbed by additional fission products not removed, albeit, at a cost of increased accelerator capacity. The accelerator-produced neutrons will not be "free" in this regard, since the cost of accelerator-produced neutrons are expected to be in the range 0.3-0.5 M$/mole[14,15]. Most importantly, the off-gas will be sequestered sufficiently long for the ^{135}Xe to decay. The longer-lived isotopes will be reintroduced to the primary system as part of the cover-gas system.

The off-gas system for ABC, therefore, is not as complex in comparison to that required for the MSBR. Off-gases will be stripped from the fuel salt using a bubble generator and a bubble separator that operates on a small bleed line. The separated gases will be routed to the air space in the drain tank, where they will be held for two hours. The gases will then be taken to a short-term decay bed in which they will be held for about 47 hr. Whereas the MSBR design used activated charcoal beds, zeolite would more than likely be used for ABC. The resulting mixture of helium and longer-lives fission gases will be reintroduced into the bubble generator. The net result will be a slow buildup of longer-lived isotopes in the cover gas over time. The neutronic effects of these isotopes are not expected to be large, but the operational effects are important because of the additional shielding and remote maintenance required for the cover gas system. If these additional requirements are found to be overly restrictive in the course of more detailed designs, additional off-gas processing can be included to lower the radiation and heat load from the cover gas system.

d. On-line Separations

Electrowinning (electrolytic deposition) techniques are proposed for the on-line removal of "noble metal" and zirconium fission products that are produced in the ABC system[8]. The "noble metals" were defined during the MSRE operation[11] as those metals that form fluorides that are less thermodynamically stable than ZrF_4 and include: Mo, Nb, Ru, Rh, Ag, Cd, Tc, and other transition-metal fission products. The electrowinning method has been used extensively in other industries (*e.g.*, in aluminum production) to yield pure metals from oxide or halide feed materials that have been dissolved in a molten salt. For the ABC applicationn the purification of the molten salt instead of the production of a pure metal is of interest. A description of the electrochemical cell and the location of the cell in the ABC system follows.

The electrochemical cell consists of a consumable anode that is fabricated from beryllium metal and a nickel cathode onto which the "noble metals" and zirconium are plated. The reaction that describes the process is

$$n\,Be(s) + 2\,MF_n(d) \leftrightarrows n\,BeF_2(d) + 2\,M(s), \qquad \text{(III-1)}$$

where MF_n is a "noble metal" fluoride or Zr of valence state n and d refers to the fluoride species dissolved in the molten salt. The process is spontaneous, with the free energy difference between BeF_2 and MF_n being ΔF = ????? MJ/mole. In principle, therefore, the cell could be operated passively; however, it may be necessary to apply a small externally generated voltage between the electrodes to enhance the rate of mass transfer. The electrochemical potential of plutonium, other actinides, and lanthanide fission products falls between those of beryllium and lithium; therefore, these elements are not removed from the fuel salt. In addition to removing fission product metals, the cell would also provide control of the oxidation potential of the fuel salt. The oxidation potential of the fuel salt would be maintained as near to neutral conditions as is reasonable.

A detailed physical description of the cell is not available because only the fundamental operating principles have been established.[8] Basic design criteria, however, can be described. The electrowinning cell would be located before the intermediate heat exchanger so that the possibility of deposition or plate-out of the noble-metal fission products on the IHX tubes could be reduced. The deposition of noble-metals on the heat-exchanger tubes does not present a materials problem but could reduce the efficiency of the heat transfer system, as well as increasing the difficulty of IHX maintenance. The entire cell would consist of a series of electrochemical cells so that the removal of fission-product metals could be optimized. Maintenance of the cell would consist of the periodic replacement of the anode and cathode and must be completed by remote operations. The cathode materials that are removed from the cell during maintenance procedures could be stored as the metals, or could be oxidized, blended with silica, vitrified, and sent to a storage facility in the form of glass.

e. End-of-Cycle Separations

Ultimately the fuel salt must be processed to remove the other fission product metals that have been produced by the fission process. These fission products are primarily lanthanide, alkali, and alkaline earth metals; in general, these species are highly radioactive. The end-of-cycle separation focuses on the extraction of lanthanides and also actinides from the fuel salt, so that the salt could be used in another operation cycle. Cesium and strontium are two fission products of major concern from a radiological point of view, but these elements would remain in the fuel salt and be recycled into the core. The end-of-cycle removal times are not as demanding as the on-line separation and could be accomplished using a batch process. The extraction chemistry and a brief description of the process design criteria are given.

Removing the lanthanides and actinides (*i.e.*, primarily ^{242}Pu, Am , Cm) from the fuel salt at the end-of-cycle would be accomplished by a liquid-metal/molten-salt extraction process[8]. The process was developed for use in the MSBR[10,11] and is described by the following reaction:

$$n\,Li(d,Bi) + MF_n(d,\ 0.67\,LiF - 0.33\,BeF_2) \leftrightarrows M(d,Bi) + n\,LiF, \qquad \text{(III-2)}$$

where M is an actinide or lanthanide metal of valence n. The lithium concentration in the liquid bismuth is chosen so that the actinides are not preferentially extracted from the fuel salt, but are extracted along with the lanthanides. Multiple stage extractions would enhance the removal of the actinides and lanthanides from the fuel salt. After the extraction process is complete, the alloyed lanthanides and actinides would be separated from the unalloyed bismuth by distilling or vaporizing the pure bismuth. The remaining alloys could be oxidized, blended with silica, vitrified, and sent to a storage facility in the form of glass, or the alloys could be fluorinated and fed into an accelerator-driven waste burner.[2]

Again, a detailed description of the extraction and distillation apparatus is not available and only the fundamental operating principles have been established.[11] Unlike the on-line separation process, however, the extraction process had been consider for use in the MSBR and many of the design criteria that are discussed in Refs.10 and 11 can be applied to the ABC system. The distillation system would be a standard vacuum distillation apparatus, with appropriate changes being made to the system to accommodate working with radioactive materials.

7. Containment and Safety

The basic three-level containment philosophy adopted for ABC was illustrated generally in Fig. 4. Figure 14. gives plan and elevation views of the assemblage of the main accelerator, primary, and secondary systems described above. The compartment or cell structure used in the MSBR conceptual design is indicated, as is the secondary containment building, vertical maintenance scheme, and key containment penetrations.

The MSBR design included a full containment structure erected around the primary system; this feature was included in the ABC design. The basic philosophy applied to this design process was to provide three barriers (Fig. 4) to the release of large amounts of radioactivity to the environment. This definition was interpreted as requiring three barriers for the primary system, the chemical-processing system, and the off-gas system. In the case of the fuel salt in the primary system, the first barrier is the piping and vessel walls. The second barrier is the reactor-vessel cell boundary. Finally, the third barrier is the containment structure itself. The boundaries are similar for the remainder of the plant. The first barrier is always provided by the structure of the system (*i.e.*, piping, vessel, *etc.*), the second barrier is the cell boundary, and the third is the containment building *per se*.

The containment proper surrounding all the individual cells is of the large, dry type (????). For this containment the volume is used to ensure against failures resulting from over-pressurization. Although the maximum design pressure of this containment has not been estimated, the maximum pressure obtainable under accident conditions is not expected to be large (basis ????), and, therefore, the design pressure associated with this containment should be sufficient. The potential energy in the containment is expected to be considerably smaller than that in a standard PWR (*e.g.*, ????GJ/m^3 compared to, ???? GJ/m^3 for the ABC), since the molten salt in both the primary and secondary systems is operated at relatively low pressure.

Only a "top-level" assessment of the safety issues associated with the ABC system has been made. Because the details of the system have yet to be determined, an in-depth safety analysis cannot be performed. Nevertheless, safety considerations have played an important role in the development of the ABC plant layout reported herein. For example, providing three independent barriers to fission product release required isolation valves on several of the molten-salt and steam-piping systems, as well as for the accelerator vacuum system. A secondary cooling loop on the lead target cooling system is suggested.

Even without an in-depth analysis, a number of potential safety-related issues have been identified. These issues must be addressed as part of a future, detailed design in a way that methods and configurations for dealing with these events are included in that design. The following events/transients have been identified as requiring additional study: target-window failure; steam-generator-tube rupture and propagation; IHX-tube failure, steam-line failure within containment; primary-system rupture; criticality excursion (*e.g.*, caused by condensation/concentration of fissionable material); core blockage; loss of heat sink; loss of target cooling system; and failure of fuel-salt drain system. Most of these events were considered during the ABC Plant Layout Study, and to varying degrees means of dealing with these events have been indirectly and individually included. Further work is needed, however, to assess the potential for combinations of these events, and the ability of the system to respond to multiple failures.

8. Instrumentation and Control

A significant development effort will be needed to provide the necessary instrumentation methods and control systems to allow characterization and precise control of the ABC blanket under all normal and off-normal conditions. While this development appears to be straightforward, only limited development has been performed in this area. The MSRE[11] relied on batch sampling to characterize the fuel salt during operation. While this remains as a backup option, on-line measurement of the relevant fuel-salt properties such as constituent concentrations and REDOX potential is attractive for the system.

The requirements and outstanding issues in the I&C area, as summarized for the MSBR conceptual design, generally apply to the ABC with only minor modifications. Introduction of the accelerator-generated neutrons may complicate the neutronics monitoring, especially since k_{eff} for the system must be maintained below some nominal level (0.95-0.98). Instrumentation and control is recognized as an important subsystem of the ABC system below some nominal level (0.95-0.98). While I&C is recognized as an important subsystems, the issues appear to be developmental and not insurmountable. The development and reliability of on-line chemical analyses needed for the second-by-second control of ABC power input and output, however, is expected to present some challenges to the process and control engineer. Likewise, the control of the accelerator in a system of four thermally independent blankets driving four independent thermal-to-electric conversion systems in a way where the loss of one Target/Blanket system does not cause the entire ABC system to go off line presents additional control challenges.

IV. KEY ISSUES

The main role of the ABC Plant Layout Study is to identify global issues for the ABC approach to plutonium disposition and parallel electric-power generation prior to the initiation of a preconceptual design. Only the existence of the detailed preconceptual design of the MSBR[10,11] made this study possible. The results of modifying and/or adopting key elements of the MSBR preconceptual design to the ABC mission are summarized on Table III. Top-level technical issues that have been identified in the course of the ABC Plant Layout Study are summarized in this section. Because of the nature of this study, some of the design choices and related technical issues associated with the MSBR may have inadvertently and unnecessarily been translated into the ABC design; when possible, alternative approaches are suggested. After presenting an overview and a subsystem-by-subsystem list of technical issues, these issues are prioritized for elaboration as part of a subsequent ABC R&D Plan.[9]

A. Overview and Design Departures from MSBR

Although the ABC Plant Layout Study relied on the MSBR conceptual design study, important differences between the ABC and the MSBR designs exist. In the context of ABC and in an approximately descending order of importance these differences include: a) driven subcritical operation with a considerably enhanced flexibility in neutron balance resulting from the excess accelerator-produced neutrons and no requirement to breed fissile fuel an a prescribed rate; b) a chemical-processing philosophy that emphasizes physical separations (*e.g.*, gas-liquid, precipitation, plate-out and/or electrowinning) over chemical separations (*e.g.*, chemical extraction, REDOX); c) multiplicity of thermal-power-generating core; and d) a Li-Be fuel salt unburdened by heavy thorium loadings. For reasons related to resources or an inability to find an improved option, the ABC design retained the following key features of the MSBR: a) a strongly moderated Core consisting of ~13% fuel salt flowing through a graphite matrix housed in a Hastelloy-H vessel; b) a separate fuel-salt dump tank serving as a focal point for all elements of a multi-faceted Chemical-Processing system; c) an IHX cooled by a sodium fluoroborate secondary coolant salt; d) a generally vertical maintenance scheme for all Primary- and Secondary-System components into an overlaying Containment and/or Hot-Cell Room(s); e) a supercritical-steam power-conversion system; and f) a tertiary containment system based on a combination of operational cells or rooms (Primary System, Secondary Coolant System, Steam Generator) housed in a Containment Building and interconnected with Main Beam and Steam Isolation Valves (MBIVs, MSIVs). The ~1,000-m-long accelerator; the associated recirculating power requirement; the proton-beam transport through and bending/expansion within the Secondary Containment Building; the delivery of that beam through a thin and relatively delicate window to a spallation/evaporation neutron-generating target located centrally in the graphite/molten-salt/plutonium blanket; and the need to shield for deeply penetrating high-energy neutrons all top the list of unique technical features (and challenges) of the ABC approach to plutonium disposition.

The technical selection and winnowing of the MSBR features and the subsequent adaptation to define the unique features of the ABC has led to the particular (skeletal) design and plant layout describe in Sec. III. This design is not optimal, but, given the resolution of key (outstanding) issues, this design can with equal probability be made workable. The following section describes for each major ABC subsystem these technical issues in generally qualitative terms. An approximate prioritization of the main technical issues, as well as directions for improved ABC designs based on subcritical, fluid-fuel approach and the benefits this approach portends (Table I) are addressed in the following subsections. This technical issues are summarized and prioritized in Sec.III.C., for use in developing an ABC R&D plan.[9]

B. Main Subsystems

1. Accelerator

The 800-1,000 MeV linear accelerator used to provide the 0.05-0.10 A proton current on target with a "wall-plug" efficiency of 45-50% was included in the ABC Plant Layout Study only for the purpose of completeness. This component exerts a major influence on the site and BOP characteristics, as well as those of the Containment Building (Fig. 14). An assessment of the physics and engineering requirements of the accelerator, even at the level considered in the "top-level" systems diagram given in Fig. 6, however, was not within the scope of this study. The nominal parameters and approach of the accelerator required for each of the $N_{ABC} = 4$ ABC units (Table III) are well within the scope of systems being proposed and designed for nearer-term applications[26,27], however.

Generally, the highest technical risk for the accelerator being proposed to drive ABC resides at the low-energy front end (*e.g.*, IS, RFQ, DTL, and BCDTL or CCDTL; Fig. 6). For the blanket multiplications and beam currents envisaged for ABC (Table IV), funneling of two front ends may not be necessary, as is required in the higher-current APT[27], but two front ends may be desirable for purposes of increase reliability. The main issue for the high-energy accelerating structure is the efficiency with which RF power can be converted to beam power under conditions where beam scrape off ($< 10^{-8}$/m) and CCL activation (hands-on maintenance) can be minimized. The use of superconducting CCLs offers important advantages in this regard, in addition to promising increased reliability. Unresolved issues related to increased cost, increased development risk, and increased time to repair the superconducting accelerating structures, however, can be identified[26,27]. Additionally, cost-optimal superconducting CCL designs favor increased beam energy[26,27] with yet-to-be-resolved impacts on (increased) shielding of the more-energetic forward-scattered neutrons in the Target/Blanket system and streaming in the general direction of the crucial fuel-salt dump tank.

The plysics and technology of splitting, switching, and transport of each of the N_{BLK} high-energy beamlets to each Target-Blanket assembly remains to be resolved and may harbor technical surprises with an inconvenience rather than a fundamental feasibility impact. Operational issues related to the optimal means by which to drive N_{BLK} relatively independent thermal-electric fission systems with a single accelerator and still keep them independent remains to be understood. Although not related to the accelerator, the degree to which other ABC subsystems (*e.g.*, Electric and Turbine Plant Equipment, Chemical Plant Equipment; Fig. 3) can be multiplexed (like the accelerator) to derive important cost benefits[15] requires further study.

2. Target

The self-cooled lead target adopted for the ABC design is probably the most efficient and eloquent configuration available for this application. Power-density restrictions for the high neutron-flux conditions required for the ABC application, along with considerations of complexity and lifetime, probably preclude the use of solid[28] neutron-spallation/evaporation targets. Although unimportant from a view point of overall energy balance, use of the secondary coolant salt to remove (and recover) the majority of the beam power from the target lead would add to the simplicity of this system. A separate (?????) target cooling system, however, was adopted because of unresolved corrosion issues related to the higher lead temperature if it were cooled with the (higher-temperature) secondary coolant salt. Radiation lifetime of the target structure and the thimble structure that isolates the target assembly from the blanket is the main outstanding issue for this system. Generally, the target assembly is expected to compete favorable with the graphite moderator in the blanket in establishing Target/Blanket maintenance, availability, and waste-stream characteristics. Lastly, the strategic location of the target window; the scheme adopted for cooling this delicate item; the degree to which beam-"footprint" variability can be controlled and monitored; and the general response to beam-induced radiation damage combine to create a challenging technical issue for this system.

The self-cooled lead target was adopted for the ABC design to maximize performance while maintaining simplicity and (hopefully) minimizing maintenance. The primary issues for this system are radiation damage to the structure (especially the window) and the structural corrosion caused by the lead. Other important design and operational issues are attributable to uncertainties associated with heat removal issues because of the poor wetting characteristics of the lead and the related uncertainties in both predictions with regard to structural cooling as well as design of the heat exchanger, including selecting the secondary coolant. The majority of these issues can only be resolved through experimental efforts. A variety of design options can be envisaged, as discussed in Sec. IV.D.1., but these options invariably result in a trade-off between technical difficulty and overall performance.

3. Primary System

The Primary System design for ABC follows as closely as possible that of the MSBR. The Primary System is perhaps the most important system in the ABC plant, since it contains the majority of the radioactive material. The primary system, as described in Sec. III.B.3., consists of those components that contain an appreciable quantity of fission products. For the MSBR design, a number of outstanding issues related to the primary system were identified[10]; these issues are discussed in the following subsections.

a. Core

The Core design was taken almost directly from the MSBR design, in so far as the moderator, fuel-salt, reflector, and vessel components are concerned. The power per Target/Blanket (Core) module was scaled down from 2,250 MWt to 711 MWt, the height-to-diameter ratio was increased slightly, and the central target was added. The use of individual graphite stringers (*e.g.*, internally moderated configuration) was adopted in the course of the trade of between simplicity (*i.e.*, the externally moderated configuration and reduced graphite waste) *versus* vessel lifetime and available design detail (*i.e.*, the internally moderated MSBR-like configuration); this issue is addressed by neutronics computations in Appendix E. Specifically, differences in dpa rate and increases in plutonium inventory needed to maintain a constant power (*i.e.*, k_{eff}) blanket over a 12-(full-power) year operation are reported (Appendix E, Figs. E-6, and E-7).

One of the most important issues with respect to the Core design is the absence of experience with the ABC fuel salt that is a different fuel salt than that used in the MSBR.

The MSBR fuel salt contained a large quantity of thorium, which increased the density and changed the physical properties from the base $LiF-BeF_2$ salt. The plutonium-bearing fuel salt for ABC is similar to the basecase fuel salt use in the MSBR design, since the plutonium fraction is less than one percent. The ABC fuel salt, therefore, can be considered a $LiF-BeF_2$ with a plutonium impurity.

A large experimental effort will be required to develop an equivalent knowledge base for the plutonium fuel salt used in the ABC design. From the standpoint of fluids and heat transport, the differences between pure $LiF-BeF_2$ and salt containing plutonium, however, are not significant. For corrosion and salt stability, significant differences may exist. Tests will be required to quantify plutonium solubility in the fuel salt, both with and without fission product impurities. Some of the rare-earth fission products may compete with plutonium to an extent where plutonium solubility in the salt is reduced.

Another facet of this fuel salt experimental program would focus on determining the physical properties to an extent needed by a detailed thermal-hydraulic design of the Primary System. As noted, the values for pure $LiF-BeF_2$ were used in this study.

The graphite moderator presents a second important issue for the ABC Core. The graphite is a problem primarily because of a potential to generate high level waste over the course of its lifetime, and disposal of large volumes of fission-product-contaminated material may be problematic. As discussed in Appendix E, designs have been developed that limit the graphite in the core. These externally moderated core concepts shift the design problem from one of (contaminated graphite-moderator) waste generation to one of reactor vessel lifetime (a waste of another kind). Although the relative difficulty of the contaminated-moderator *versus* reactor-vessel waste problems in terms of volume and intensity remains to be resolved, the central issue is the feasibility of frequent reactor-vessel replacements and the need to maintain the $p_f = 0.75$ plant availability. Although the neutron spectrum for the externally moderated concept is somewhat harder than that of the internally moderated system, the significant difference in size between the two (the former is smaller) results in large increases in average flux for the externally moderated option. This configuration projects a vessel diameter of only 1 m (compared to 3.5 m for the MSBR-like internally moderated configuration), which, for the same level of total fission power, results in a substantially larger (total and fast) average neutron flux. Ultimately, an operational and cost (*i.e.*, replacement cost, availability, waste stream, etc.) trade off must be resolved that centers primarily on the average core power density and the core or reactor-vessel lifetime. Intermediate use of graphite in both the fuel-salt region and as a reflector placed between the fuel salt and the reactor vessel represents a option in need of future (thermal-hydraulic, neutronic) optimization computations, some of which are reported in Appendix E.

The degree of (credible) inherent safety of the ABC core related both to reactivity insertions and to loss of cooling represents a second important design issue for the core. Since the impact and fate of most of the afterheat is intimately related to the fate of the fluid fuel (*e.g.*, use of freeze plugs and a passively cooled dump tank[10]), attention was focused more on the former and the reactivity temperature coefficient (RTC). Preliminary computations indicated that the Beginning of Life (BOL) RTC for the internally moderated core design was (undesirably and unlicensably) positive; unlike the ^{233}U-fueled MSBR, a low-energy fission cross section in ^{239}Pu is experience as the core temperature increases, and the resulting power spike that results in passing through this resonance is thermal-mechanically unacceptable. It was this finding that led to the exploration of the externally moderated concept for ABC; the somewhat harder neutron spectrum aligned more favorably with the fission resonances in ^{239}Pu to give a negative RTC at BOL (????$). At this stage in the ABC blanket scoping study, it is not clear whether resonantly absorbing fission products will rapidly force the RTC negative irrespective of the BOL fuel, moderator, or general neutronics condition. As a compensating feature, burnable resonance absorbers, such as gadolinium or erbium, can be use to force a negative RTC at BOL. Although of initial value, the kinetic character of the core as a function of exposure may make use of burnable poisons throughout operation difficult, depending on whether such additions are competing with plutonium solubility or are being removed by the chemical-processing system.

The present APC Plant Layout Study made a choice of in-core fuel-salt fraction that favored reactor-vessel longevity over simplicity of Core design and reduced graphite/fission-product waste stream. A fully moderated Core design, therefore, was assumed, and the graphite disposal problem (*e.g.*, the added chemical processing required to reduce the contamination of the damaged graphite) will be dealt with later in the detailed design. The graphite lifetime exerts a strong impact on the quantity of material ultimately produced, but in magnitude has yet to be quantified for the ABC design. The graphite lifetime is a direct function of the neutron flux intensity and spectrum in the core, which in turn is a function of the plutonium loading, the fission product removal time, and the fuel salt graphite core fraction (i.e., degree of moderation). These core design used in this ABC Plant Layout Study was taken from the MSBR design, and has not been fully evaluated in the context of the ABC mission. Even the MSBR core design was described[10] as a preliminary design and subject to change in the course of a detailed design. It is expected that more detailed neutronic calculations will indicate improvements to the core design.

b. Fuel-Salt Pump

The fuel-salt pump design was taken unaltered from the MSBR design (Fig. 9). Since the pump design was not scaled back from the MSBR requirements (1.0 m^3/s *versus* 0.55 m^3/s), the result presents a conservatively large footprint for the ABC Plant Layout. During the MSBR design effort, the pump design was not considered an outstanding issue. The only concerns relating to the pump design were scaleup of the pumps from that used in the MSRE[11] to the size anticipated for MSBR. The pumps required for either the MSBR or the ABC are much larger than the largest (molten-salt ???) pumps that have so far been operated. The design is similar to those used in the past, but scale-up problems should be expected with such a large (×???) pump increase in scale. An additional problem for the ABC application is the loss of expertise that was available in this area over three decades since the MSBR activity concluded.

c. Intermediate Heat Exchanger

The IHX design is non-standard, with the intention to accommodate remote maintenance easily. The ABC is scaled from the MSBR design (Fig. 10), with a few modifications being made. Outstanding issues from the MSBR design remain, however, and are related primarily to the heat-transfer correlations used to size the IHX. Experiments that include large-scale tests, are needed to verify these correlations. Also, the individual tubes were knurled to increase the surface area and to enhance the heat transfer. The anticipated improvement has not been verified experimentally. These uncertainties, however, portent no "show-stopper" issues, since the size of the IHX can be increased to accommodate any design shortfall. Any increase in the IHX size, however, will increase the fuel-salt volume and reduce the in-core fraction; for the present ABC Plant Layout, ????% of the total fuel-salt inventory resides in the IHX.

4. Heat-Removal (Secondary) Systems

The Heat-Removal systems together form a number of outstanding issues. While the secondary coolant salt has some useful features, it also introduces a number of difficulties. These issues are not unique to the ABC application[10,11], but nonetheless must be addressed. Also, the design of the supercritical-steam system needed to take full advantage of the (high-temperature) molten-salt primary coolant is somewhat advanced. The SCS system presents a number of associated issues that remain to be addressed.

a. Secondary Coolant Salt

Sodium fluoride - sodium fluoroborate eutectic is the reference secondary coolant salt. This eutectic combines good thermal-hydraulic properties with low cost. This material, however, is not easy to handle, since during operation it decomposes thermally with the evolution of BF_3. The BF_3 gas must be reintroduced into the coolant-salt loop to avoid changes in the overall properties of the salt; a complex cover gas system, therefore, is required.

Other secondary coolant salts have been proposed including LiF-BeF2, HITEC ($NaNO_3$-KNO_3 ???), and Li-Be-Zr-F. While each has some advantages over the sodium fluoroborate, none were determined to be better on an overall basis. The fluoroborate salt is probably the best characterized. Other coolants including liquid metal and helium, have been suggested, but as with the range of molten-salts considered, none was judged to be as good as fluoroborate.

Another consideration that has to be taken into account in choosing a secondary coolant is trapping of the tritium generated in the course of fission and from neutron reactions with the lithium-bearing fuel salt. The sodium fluoroborate has been shown to trap greater than 90% of the tritium introduced under steady-state conditions. None of the other secondary coolants, with the exception of helium, has this capability. At the proposed operating temperatures (???? K), tritium readily diffuses through Hastelloy-N. It is expected that the majority of the tritium will migrate into the secondary coolant system through the thin-walled IHX tubes. This tritium must be prevented from reaching the steam system, since further containment cannot be assured (*e.g.*, contamination of the turbines is to be avoided). The sodium-fluoroborate coolant salt offers a potential for tritium trapping and removal. The mechanisms responsible for the trapping, however, are not understood. The MSBR project was canceled prior before investigations into the mechanisms could be completed. Before this trapping can be reliably invoked by the ABC design, an experimental program that is combined with material compatibility tests are essential.

Material compatibility tests are needed to settle another outstanding issue. The corrosion rate of the reference construction material for the entire secondary coolant system, Hastelloy-N, in fluoroborate salt is low. In the presence of moisture, however, the corrosion rate increases dramatically. Corrosion is primarily a concern for pinhole leaks or cracks in the steam generator or reheater tubes; massive failures, although to be avoided, would be easily detectable, while small leaks might introduce moisture for some time before being discovered. Tests will be necessary to determine the corrosion rates of Hastelloy-N under various moisture levels in the sodium-fluoroborate. Analysis will be required to determine the allowable moisture concentration. Finally, if the allowable moisture level is too stringent, additional moisture removal capability will have to be designed, or duplex tubing will be required in the steam generator and reheater.

b. Supercritical-Steam System

The supercritical-steam system is advanced, but the design can be readily extrapolated from the state of the art. The design adopted by the ABC Plant Layout Study is taken MSBR design, which was in turn adapted from the Bull Run Steam Plant[25]. The Bull Run plant is coal fired, but otherwise has the same steam conditions as adopted for the ABC. Operation with supercritical-steam system is complicated by the feedwater requirements that in turn is dictated by the use of fluoroborate. To prevent freezing of that coolant salt in the heat exchangers, the minimum feedwater temperatures have been set to 618 K (650°F) for the reheater (Fig. 12) and 644 K (700°F) for the steam generator. High (???) quality steam is first bled from the steam-generator outlet to generate this high temperature. The main outstanding issues for the steam system are the heat transfer in the heat exchangers and design of the reheat steam preheater.

Tests will be required to assure that either freezing does not occur in the steam generator or reheater under normal and transient conditions, or that freezing is not detrimental to the equipment. The tests will be similar for both pieces of equipment, with only the steam conditions differing.

To provide final feedwater heating, the first (???) quality steam exiting the reheat-steam preheater is blended directly with the feedwater. In a subcritical-steam system, this mixing would produce violent (mechanical) reactions from bubble collapse. For the supercritical system, however, the two phases are indistinguishable, and mixing may be accomplished in large spherical drums, as is done at the Bull Run power plant. If the reference design is changed to a subcritical-steam system, experiments will be required to verify that mixing can be accomplished without damaging the equipment.

The reference ABC design calls for a each Target/Blanket module to be serviced by an individual turbine plant of capacity 280-319 MWe. This arrangement may not be the most cost-effective for the for the overall $N_{BLK} = 4$ ABC system. Cost-based parametric studies[15] are needed to assess the optimal BOP configuration in this regard, and the operational impact and flexibility of operating with more independent units. Since all four units are driven by the same accelerator, control issues related to the desire to achieve the highest availability for electrical output are identified.

5. Chemical Processing

The chemical processing requirements for the ABC and the MSBR systems are different. Chemical processing is perhaps one of the least defined elements of the ABC design. The basic requirements have yet to be defined, since neither the necessary neutronic (burnup/burnin) analyses nor chemical transport/processing have not been performed. Basic information, however, is available to provide focus on the outstanding issues. While many of the processes are similar to those anticipated and modeled as part of the MSBR conceptual design,[10] the emphasis in chemical processing for the ABC has been driven primarily by the goals of increased simplicity and waste minimization. This shift in emphasis has been driven primarily by goals of increased simplicity and waste minimization. At the present state of ABC concept development, a number of generic issues generated from a perspective of the the above differences can be identified and described.

First, determinations are needed of the removal rates for off-gases and insoluble fission products for a fuel salt that has significant differences from that used in the MSBR design. These rates are determined by neutronic and thermal-physical considerations, as well as sizing considerations for the chemical processing equipment.

The primary concern of the chemical-processing design should be the determination of the required reprocessing rates and equipment sizes. If a batch fuel cycle can be accommodated at a frequency that is neutronically and operationally acceptable, only limited on-line processing will be required. If, however, the required removal time is short, extensive chemical processing development will be required.

One of the early design drivers for the MSBR fuel reprocessing was separation of ^{233}Pa from the fuel salt so that decay to the ^{233}U fuel could occur outside the competition of the neutron environment. Because thorium plays no role in the ABC, this design driver is not an issue. Instead, the level of parasitic neutron absorption as it impacts both the efficiency of plutonium disposition and the need for added accelerator capacity, along with solubility limits, inventories, and activity control, are key design driver for the ABC. Plutonium solubility limits are are also an issue. The lanthanide fission products complete with plutonium for fluoride ions and cause a reduction in the amount of plutonium that can be held in solution, thereby impacting reactivity (burn up) limits.

Another chemical processing issue revolves around the off-gas system requirements. As with the soluble and insoluble solid fission products, the neutronic impact of the gaseous fission products have not been assessed. Some level of off-gas processing will be required to strip the fission products from the helium cover gas. Once the requirements for this system have been determined, a number of issues become important. It is known, for example[10], that the off-gas cleanup equipment can be large (*e.g.*,activated carbon filter beds) and will require a large amount of floor space within the containment volume. The chemical processing equipment may lead to an increase in the containment size, which represents primarily an economic rather than a technical issue.

Regardless of the degree of fission-product removal that is eventually required, a minimum amount of equipment is required to prepare initial and make-up batches of fuel salt, and to cleanup salt after an off-normal situation such as a steam-generator tube leak or an IHX tube leak. The flowsheets for these operations need to be developed. Fission-product removal and cleanup is an outstanding issue because the sufficient floor area must be provided for these operations within an otherwise expensive and congested containment volume.

6. Operations and Maintenance

Operations and Maintenance is considered an outstanding issue, particularly in view of uncertainties of target-structure, graphite-moderator, and reactor-vessel radiation lifetimes. The design of the ABC allows for all components having an expected lifetime shorter than that of the plant can be replaced in a time required to assure the design plant availability factor ($p_f = 0.75$). Many of these components will require remote maintenance because of fission- and activation-product contamination. The containment cells and buildings (Fig. 14) were designed with spacings between components sufficiently large to accommodate expected maintenance operations, as well uncertainties in estimates of component sizes.

Additional laydown area and/or maintenance clearances may be required for remote-maintenance operations and should be identified early in the ABC design process. These issues, while not in the class of "show stoppers", greatly affect the component layout, cell and building volumes, and (ultimately) cost.

An operations plan is needed to identify undefined equipment needs and to complete key remaining holes in the design. For example, refueling the system will require an interface with the Primary System *vis-a-vis* the fuel-salt drain tank and the Chemical-Processing systems. Graphite-moderator and lead-target replacement equipment must be considered.

7. Instrumentation and Control

With the exception of Chemical Processing, the I&C system is most in need of definition. The I&C requirements of the MSBR design were reviewed as part of the ABC Plant Layout Study. The requirements for ABC in this area are similar to the needs anticipated for the MSBR, but important differences can be identified. Because of advances in I&C methods and technology since the completion of the MSBR program over three decades ago, the entire I&C system will require redefinition. Additionally, both the safety, neutron-economy, and Chemical-Processing complexity and waste-stream issues are expected to be relaxed for the ABC compared to MSBR.

Monitoring equipment is needed to provide fast-response and multiple point information on temperature, pressure, fluid levels, composition, moisture content impurity levels, plutonium concentration, REDOX potential, *etc.* Several of the required capabilities were never developed by the MSBR program. For example, fuel-salt composition measurements had to be made by taking samples followed by exo-reactor analysis This method was slow, and its use would dramatically affect operations and safety procedures.

Another I&C issue is the neutron monitoring in the presence of the target spallation source. Methods of monitoring k_{eff} must be developed and verified for the subcritical ABC operation. The SCRAM system has also not been designed, other than the recognition of a need to incorporate shutdown (and possibly control) rods into the Core design. The applicability of chemical "shims" on a widely variable time frame, as well as the monitoring of local power densities and temperatures within the Core present important I&C challenges.

A shutdown system has to be defined. Signals that will require a module shutdown remain to be determined. Likewise, the means by which startup, approach to full power, the long-term control of power output and spatial power distributions, and both the short-term and long-term of the Target-Blanket system and the Primary System in general in the hot-standby condition remain to be resolved. These I&C requires are primarily design rather than technology issues, however.

8. Containment and Safety

The safety philosophy used for advanced fission-reactor designs was adopted for the ABC Plant Layout Study. As is shown in Fig. 4, three barriers encompass all potential radioactive source terms. A full containment building was provided, even though a passive fuel isolation and cooling system (drain tank) is incorporated into the ABC design. Much additional effort will be required to identify the key accidents, and to assess the systems ability to deal with these accidents. This effort, however, must focus onto containment of the accelerator *per se* and prevention and/or mitigation of fluid and pressure transmissions between the main cells and buildings that comprise the ABC plant.

V. CONCLUSIONS AND RECOMMENDATIONS

A preliminary plant layout has been developed for a molten-salt-based ABC using as much as is possible and appropriate the detailed design and optimized results reported for the Molten-Salt Breeder Reactor conceptual design[10,11]. The main goal of this ABC Plant Layout Study is the identification of key technical issues for the ABC approach to weapons-plutonium disposition on the basis of a pre-conceptual design layout of key non-accelerator components. A secondary, but nonetheless important, goal of the ABC Plant Layout study is the identification of design options for a molten-salt-based ABC concept that would be appropriately used by a future ABC conceptual design. This section on conclusions and recommendations summarizes key technical issues and alternative design options for ABC.

A. Ordering of Key Issues

Although the Accelerator Equipment was include for reasons of completeness in this ABC Plant Layout Study, the identification and ordering of key technical issues for elaboration in an ABC R&D plan[9] is limited in this study to the Target, Primary System, Heat-Removal (Secondary) System, Chemical Processing, Operation and Maintenance, Instrumentation and Control, and Containment (Safety). Research and Development issues for the high-power (capacity), high-current (efficiency), and necessarily reliable (multiplexed Target-Blanket assemblies) accelerating structure, however, can not be minimized. The focus here, however, is on issues not related directly to the Accelerator Equipment, albeit, important interfacial issues and influences (Fig. 4) exists.

The MSBR conceptual design had been an essential element in defining all non-accelerator ABC plant components. In establishing a priority list of key technical issues for the molten-salt ABC, it is helpful to begin with a brief revisit of important technical issues raised by the MSBR conceptual design. While the MSBR has been considered a "chemist's dream", some aspects of that program might also be considered a "materials dream". None of the problems unveiled by the MSBR experience where considered to be "show stoppers" or "fatal flaws", with the possible exception of stretched doubling times caused by the pull of marginal neutron economics and the impact thereon of fission-product buildup related to uncertainties in the chemical-processing effectiveness. The main problems encountered during construction and operation of the MSRE and left unresolved at the time of the MSBR project closure where related to materials: a) radiation-induced helium embrittlement of the Hastelloy-N structural material; b) containment of the significant quantities of tritium formed (primarily) from neutron captures by lithium; and c) grain-boundary attack in the Hastelloy-N structural material by tellurium fission product. Solutions to these problems where left in the legacy of the MSBR program: a) immobilization of helium in Hastelloy-N by carbide precipitates; b) reduce tritium production by selection of a alternative fuel salt; and c) adjust fuel-salt REDOX potential to maintain the tellurium fission product in solution. While these singular solutions to singular problems encountered by the MSBR project do not provide global assurances that the materials problems for molten-salt systems are resolved, steady progress of this kind is encouraging.

In laying out and ordering the key (non-accelerator) technical issues for ABC, it is appropriate to begin with a general statement of materials requirements, particularly as they may differ from the MSBR experience. These general chemical and radiation-effects materials limitations impact all life-time (plant-availability and operating-cost) and waste-stream (target structure, graphite moderator, Primary-System structure, fuel salt) performance measures of ABC effectiveness. Key technical issues related to specific ABC subsystems then are listed in descending order of priority according to: Target; Blanket; Chemical Processing; key non-blanket Primary-System components; Secondary and Balance-of-Plant systems; and Containment and (related) Safety systems.

1. Materials and Fuel-Salt Chemistry

Since material and chemical issues are identified with all of the main ABC subsystems, these are first described as a generic class at the top of the of ABC technical-issues list. As summarized above for the MSBR, materials requirements and uncertainties associated with the Primary System and Chemical Processing also are expected to present a dominant concern for the molten-salt ABC. The differences in the fuel-salt composition and associated REDOX potential, however, are expected to lead to important differences in materials problems, even if the "tried-and-true" Hastelloy-N alloy is used also as the primary containment material for ABC. While the solutions to the Hastelloy-N problems described above for the MSBR may also apply to a fuel salt with the dominant PuF_3 species present almost at impurity levels and without heavy loadings of thorium, the control of tritium, gaseous fission products, and noble-metal (low-solubility) fission products through on-line processing and off-gas control is ranked as the top issue for the non-accelerator part of ABC. In addition to the control, removal, and collection of insoluble gaseous and noble-metal fission products, the control of soluble fission products (*e.g.*, lanthanides), and the impact on both the neutron utilization (*i.e.*, accelerator capacity and operating cost) and post-irradiation fuel-salt remediation and disposition define crucial operational, safety, and waste-stream issues for ABC. The degree to which long-lived and/or strongly parasiticly absorbing fission products are incorporated into/onto frequently replaced graphite core components determines both the overall neutron economy and the level of post-irradiation cleanup and the eventual classification of this potentially large-volume waste stream. Ranked close in importance with these chemical-processing issues is the control of plutonium (and actinde) solubilities in a system where the plutonium-inventory requirement can vary by factors of 5-10 over the life of irradiation (Appendix E, Fig. E-4). Hand-in-glove with these issues is that of Hastelloy-N (or other alloy) compatibility under high-radiation conditions combined with wide variability of chemical environments throughout the ABC primary system.

2. Main ABC Subsystems

a. Target

Along with the Accelerator Equipment, the Target has no MSBR counterpart. The self-cooled, liquid-lead target and associated (niobium alloy) window and structure operates at the highest heat flux (????? MW/m^2), power density (???? MW/m^3), and (high-energy) neutron flux (??? $\times 10^{20}$ $n/m^2/s$). The target performance is central to the overall efficiency (primary neutron yield, blanket neutron coupling) and availability (mean-time-to-failure and mean-time-to-replace) of the ABC. Residing operationally and physically at the interface between the accelerator and the plutonium-bearing fluid-fuel blanket in a high-importance region of that blanket, the target performance is critical to all operational and safety facets of ABC. The main technical issues associated with the application of this high-performance liquid-metal target system include:

- target material choice as related to neutron production efficiency, operating temperature, chemical compatibility (in a changing chemical environment), radionuclide production, waste generation;

- development and demonstration of an engineering configuration that assures reliable operation under high heat- and neutron-flux conditions, high- power-density operation, high thermal-mechanical stresses, and in a cross-roads environment that is central to achieving an acceptably safe, efficient, and cost-effective ABC.

- development of chemical and thermal-mechanical monitoring systems, secondary cooling systems, and single-unit remote replacement systems that assure design safety and availability standards/goals under the anticipated conditions of relatively short operational longevity of this key-stone system.

b. Blanket

While the ABC blanket configuration has been adopted largely from the MSBR design, important materials and fuel-salt differences listed in Sec. V.A.1. contribute to related issues in need of resolution. Furthermore, most of the technical issues listed above for the Target apply directly to the Blanket, particularly as related to radiation longevity, thermal/mechanical/neutronic diagnostics, reliability/availability/maintenance/inspectability (RAMI), and post-irradiation cleanup and waste-stream generation. An important issue is the degree to which the MSBR-like configuration can be re-optimized to give a simpler, reduced-waste, and increased-life ABC blanket while maintaining most of the important attributes of the MSBR approach. The material reported in Appendix E gives preliminary neutronic results on an "externally moderated" molten-salt configuration wherein the amount of graphite in contact with the fuel-salt is considerably reduced. While the Blanket is central to the fissioning of plutonium and the associated power generation, in the present design, it contains only ~30% of the active fuel salt (and associated plutonium and fission-product inventory); as important as is the Blanket, functionally, it is only a part of the overall transmutation(fissioning)/ chemical-processing system.

c. Chemical Processing

Most of the issues listed in Sec. V.A.1. pertain directly to the Chemical Processing system. In addition to control and collection of gaseous, noble-metal, and soluble fission products, as well as the time-varying plutonium concentration and the distribution of that concentration throughout the Primary Systems, Chemical Processing encompasses issues related to: plutonium feed preparation and injection; b) preparation of separated fission products for either disposal or re-injection into the Primary System; and c) remediation of all fuel salt into a "standard" waste form that is acceptable for geologic disposal. The fuel-salt dump tank played a central role in the Chemical-Processing system suggested by the MSBR conceptual design, and this central role remains in the adaptation to the ABC. In terms of fluidonic functions and scope, the dump tank is significantly more complex than the blanket, albeit, the power and neutron loads are considerably reduced. When combined with the scheduled (operational) and unscheduled (accident) use of the fuel-salt dump tank, this system takes on an importance equal to that of the Blanket *per se*. Hence, the scope of the Chemical Processing systems includes the Primary System and the fission-product separation/collection systems appended thereto, and the function of the Chemical Processing system occurs in parallel and in conjunction with that of the Primary System; they are inextricably mixed. Key technical issues related expressly to the former are:

- a chemical diagnostics network is needed that, working in conjunction with the thermal-hydraulic, thermal-mechanical, and neutronic monitoring systems, can give an accounting of all active and passive radioactive inventories throughout the Primary System and appended Chemical-Processing systems;

- demonstration of all chemical preparation (plutonium injection and fission-product re-injection), separation, and collection unit operations (*e.g.*, gas sparging, tritium barriers, zeolite storage, electrowinning, reductive extraction, *etc.*) at a scale that is relevant to ABC for both Primary System and Target cleanup;

- development and demonstration of post-irradiation cleanup and waste packaging of from Target (spallation and corrosion products, windows, thimbles and structure), Blanket (graphite, reactor vessel), zeolite beds, electrowinnig plates, and used fuel salt;

- detailed design and simulation of all combined operational and safety functions of the interactive Primary-System and Chemical-Processing components.

d. Primary System

The essential elements of the Primary-System technical issues have in one form or another been covered in the previous sections on Target, Blanket, and Chemical Processing; these systems are inextricably mixed and share many technical issues related to component longevity, waste-steam generation, operational efficiency, and safety. Aside from the above-listed items, the remaining Primary-System components have been taken directly from the MSBR conceptual design, and the main technical issues related to these reflect the need for technical risk reduction and the related need to develop prototypes. In this category are included the following Primary-System components: fuel-salt pump; molten-salt valves; molten-salt (IHX) and liquid-metal (target) heat exchangers; fuel-salt drain systems (tanks, melt-plugs, piping, gas-transfer systems, and valves); fresh fuel-salt injection; remote maintenance schemes for a wide variety of radioactive and interconnected fluid systems; instrumentation and control of a wide variety of nuclear/chemical/thermal-hydraulic fluid systems; Primary System boundary systems, including thermal insulation (if the MSBR "furnace" concept is adopted) and interconnections with the secondary and tertiary (containment building) containment volumes. While each of these components can with acceptable confidence be designed and operated alone, important steps in overall risk reduction associated with the interactive complexities of integrated operation are needed.

e. Secondary and Balance-of-Plant Systems

The Secondary and Balance-of-Plant systems include all non-accelerator components beyond the shell side of the IHX. These systems have been scaled directly from the MSBR conceptual design, and the technical issues related thereto remain identical:

- tritium mitigation (elimination of lithium from the neutron environment) and/or containment (diffusion barriers in IHX);

- impact of supercritical-steam (SCS) power-conversion on (need for) reheater and steam-generator (SG) design, as well as impact on the SG cell layout needed to accommodate steam-tube failures;

- general cost-effectiveness of increased complexity of the SCS conversion system *versus* the increased thermal-to-electric conversion efficiency;

- choice of alternative (lower-melting) secondary coolant salt.

- operational and cost tradeoffs related to number *versus* size/capacity of SG, SCS lines and MSIVs, Turbine Plant Equipment, Electric Plant Equipment, and Miscellaneous Plant Equipment.

An number of unique features of the multiple Target-Blanket ($N_{BLK} = 4$) feature of the ABC application create issues for the SHT/BOP system that were not encountered in the MSBR design. These technical issues revolve primarily around the nature of multiplexed operation and the need to maintain constant accelerator-power input and near-constant electrical-power output in event of the loss of one Target-Blanket module. Just as multiplexing of a single accelerator represents an essential economic/technological compromise in the operation of the ABC power plant, similar techno-economic tradeoffs may exist with respect to CPE and SHT/BOP systems; these tradeoffs require future elucidation.

f. Containment and Safety

Although a traditional three-tiered containment philosophy was implemented in the ABC Plant Layout Study, a number of technical issues have been identified. These issues include:

- any surface containing fuel salt is generally defined as the primary containment boundary, and at some level is considered analogous to the fuel-pin cladding in a conventional fission power plant. This analogy requires further examination, since in the case of the fluid-fuel system, a "cladding failure" can result in the ejection of an appreciable fraction of the fuel inventory into the secondary containment (*e.g.*, the Target- Blanket cell) or beyond;

- the diffuseness of the primary containment boundary requires considerably more design definition before containment integrity and the extent to which "single-point" failures can contribute to the extent and frequency of containment-boundary violation;

- the size and multiplicity of interconnectivity between containment boundaries and the method of isolation in event of an inner-boundary violation requires resolution; the (high-pressure) SCS system requires a larger number of MSIVs, and the accelerator beamlet line serially penetrates all three containment barriers with the need for a series of fast-acting beam (accelerator) isolation valves;

- frequent target maintenance, and possibly blanket maintenance, will necessitate routine opening of the primary and secondary containment envelopes, with the volume within the (tertiary) containment building being used to provide the needed laydown and transfer areas for large quantities of highly radioactive material; the safety impact of the frequency, the duration, and the source-strength magnitude (albeit, most of the fuel salt is safely stored in the dump tank) of these essentially "singly confined" activities should be assessed.

B. Alternative Design Approaches (samples)

1. Target

Alternative design approaches for the neutron-producing target are numerous. A variety of alternate liquid metals[29-31] have been proposed, such as bismuth, lead-bismuth eutectic, lead-magnesium eutectic, and lead-tin eutectic. Although the use of these materials decreases the target operating temperature, all alternatives exhibit certain disadvantages, such as additional corrosion (bismuth, lead-bismuth), lower neutron production and higher neutron absorption (lead-magnesium, lead-tin), and small information database for use in making initial target performance predictions (lead-magnesium). Use of these alternative materials, however, should not be precluded at this point, and greater consideration would be given to these alternatives if high operating temperatures required of lead becomes a primary concern. Another possibility of limiting the effect of the high lead temperature is to remove the primary structural cooling concern (*i.e.*, the window) from the lead environment and to use an alternate coolant for this structure. That alternative raises other design issues that are outlined briefly in Appendix D.

In addition to considering alternative liquid-metal candidates, the target solid target designs using a flowing coolant offer other possibilities. The majority of neutron production for these configurations would occur in the solid material, and the liquid would only remove heat. Because of the high temperatures generated in solid materials exposed to the high-energy proton beams required by the ABC system, only a few materials, such as tungsten, tantalum, and possibly thorium, can be considered for direct exposure to the full proton beam[28]. A number of solid target designs were considered early in the ABC design, such as a water-cooled tungsten (with a secondary lead annulus) target, liquid-metal-cooled (Na,K, and NaK) tantalum or tungsten targets, and a molten-salt-cooled thorium target. While all of these targets appeared potentially functional, they all possessed characteristics, such as higher neutron absorption, greater complexity, and more severe accident scenarios, that made them less attractive than the flowing lead target.

2. Primary Coolant Systems

The main criteria for the primary coolant system in ABC are: acceptable fissile-fuel (plutonium) solubility; low pressure; and tolerably low neutron absorption cross sections in a nominally thermal spectrum. The process used for the MSBR fuel-salt selection[10,32] identified two dozen elements that met the latter criterion. As noted in Ref. 10, compounds that qualify as permissible major constituents can be formed from beryllium, bismuth, ^{11}B, carbon, fluorine, ^{7}Li, ^{15}N, oxygen, and the fissionable elements. While many compounds can be prepared with these elements as major constituents, most have been eliminated[10,32] on the basis of the need to form practical (*e.g.*, sufficiently low melting, stable) melts. In the case of MSBR and the associated need for high thorium loadings, the carbonates where eliminated. Nitrates and nitrites where eliminated for MSBR on the basis of thermal stability. On the basis of these broad arguments, only fluoride salts were deemed suitable by the MSBR designers for the list of neutronically acceptable elements.

Although flourine can moderate neutrons, the moderation power is insufficient, and an additional moderator was required in the MSBR; chemical compatibility with molten-fluoride fuel mixtures led to the choice of graphite moderator. On the basis of this broad chemical and neutronic design/selection philosophy, the MSBR core design reported in Ref. 10 was generated; this design has served with little change as a key touchstone for the ABC molten-salt concept, despite obvious differences in design constraints (Table I).

While relaxed because of the subcritical, driven, non-breeding nature of the ABC application, many of the considerations that led to the choice of molten fluoride fuel salt for the MSBR apply to the ABC. Elimination of the need for high thorium concentrations, however, along with a somewhat relaxed neutron economy and a shift from UF_4 to PuF_3 chemistry, might open other options for the fuel salt used in ABC. For each new fuel salt system that is proposed, however, many hundreds/thousands of man-hours would be required to determine plutonium solubility, physical property data, materials compatibility data, and chemical processing data for the fuel salt. The benefits of a new fuel salt, whether it is a new ternary or quaternary fluoride, or perhaps a carbonate-based system, must outweigh the amount of effort that is required to qualify the new system.

Whatever the broadened choices with respect to molten-salt chemistry for the ABC application, new primary-coolant and moderator-configuration options and variations relative to the MSBR can be suggested as areas for future work. Appendix E gives the results of preliminary neutronic parametric calculations that varied the degree to which graphite moderator is co-mingled with the fuel salt. Figures E-6 and E-7 demonstrate specifically the impact of fuel/moderator ratio and geometry on burnup capability and dpa rate (in the graphite). Movement of the moderator to the periphery of the fuel-salt zone may increase moderator longevity, reduce waste, and simplify the blanket thermal-mechanical design. Options that cool an internally circulated fuel-salt with a primary (salt) coolant that contains no fissionable material remain to be examined as a means to reduce (eliminate) exo-blanket fissile and fission-product inventories.

3. Secondary Coolant System

Several alternatives have been proposed for the secondary coolant system. The alternative that is closest to that used herein is the substitution of another secondary coolant. Suggested coolants include HITEC(KNO_3/$NaNO_3$????), LiF-BeF_2, NaF-LiF-BeF_2, sodium, and helium. Each alternative has advantages and disadvantages.

The HITEC salt has a lower melting point than the sodium fluoroborate and, therefore, allows a reduction in the feedwater-temperature requirements. However, HITEC may undergo a violent reaction if contacted with the moderator graphite. The use of HITEC, therefore, would require another coolant loop positioned intermediate between the HITEC and the fuel salt. This additional complexity and reduced thermal-conversion efficiency would be somewhat counterbalanced by the simplifications allowed in the steam system. The HITEC salt may also have the capability of trapping tritium via oxidation and subsequent sequestration in the HITEC off-gas system. While not demonstrated, this tritium trapping capability would represent another advantage.

Pure LiF-BeF_2 was used as the secondary coolant on the MSBR and is the best coolant from the standpoint of compatibility with the fuel salt. The LiF-BeF_2 salt, however, has drawbacks. This fuel salt is expensive compared to most of the other alternatives, because

^7Li must be used to avoid excessive neutron absorption and tritium production. The high melting point would require a further increase in the minimum feedwater temperature, that contributes additional complications to the steam-system design. Two considerations must be added for the ABC plutonium-burner application. First, the additional absorptions in lithium if ^6Li were to be used and an IHX leak were to occur are not as critical in the ABC system because fissile-fuel breeding is no longer a concern. The additional neutron absorptions would, however, require a higher plutonium concentration and would lower the potential burnup (BSC, ??? arn't we talking about the secondary coolant???).

The second option is to replace the steam power cycle by a helium cycle. Closed-cycle helium systems have been lately been studied for Modular Helium Reactor (MHR) applications and could be readily adapted for use in the ABC system. Inclusion of a secondary system between the helium and the fuel salt would increase the safety margin of the system, but the thermal-conversion efficiency would be decreased. Additional study of this cycle is needed to assess its viability.

A ternary fluoride eutectic, NaF-LiF-BeF_2, was studied later in the MSBR study for use as a secondary coolant. This salt is cheaper than pure LiF-BeF_2 and has a much lower melting point. However, NaF-LiF-BeF_2 does not trap tritium. If some method of tritium trapping could be developed, the use of NaF-LiF-BeF_2 with a standard steam cycle would be advantageous. The NaF-LiF-BeF_2 salt is almost as compatible with the fuel salt as pure FLIBE. The sodium would increase parasitic neutron absorption, which portends problems in the event of an IHX leak, as well as the need for increased accelerator capacity; IHX in-leakage, however, would not affect the fuel salt (????). For plutonium applications, the effects of sodium addition on plutonium solubility would need to be determined. This salt could also be used in combination with a helium power-conversion cycle. The choice between LiF-BeF_2 and NaF-LiF-BeF_2 must be made based on the basis of the desirability of lower feedwater-temperature requirements *versus* the potential for adverse effects on the fuel salt in the event of mixing of the fuel and secondary coolant salts.

Liquid metals have never been considered for secondary coolants for use in combination with a molten-salt primary loop, but the use of liquid metals have been considered for use in combination with a LiF-BeF_2 secondary loop. The additional complexities associated with use of an additional coolant, however, may outweigh the advantages of a liquid metal such as sodium. One chemistry-related disadvantage of a liquid-metal coolant, for example sodium or lithium, is that if a leak between the liquid-metal loop and the LiF-BeF_2 loop occurs, the beryllium would be reduced to metallic form.

While helium also was not considered as a secondary coolant, this coolant was considered as a tertiary coolant for use in conjunction with a FLIBE secondary coolant. Helium provides one of the simplest method of tritium trapping. It is also chemically inert and fully compatible with the fuel salt (helium is to be used as the fuel-salt cover/stripping gas). If the tritium trapping can be developed for use in combination with a helium turbine system, (*i.e.*, pass helium stream over a tritium metal getter bed) this option may prove to be to be an attractive alternative.

a

4. Chemical Processing

As mentioned earlier, chemical processing is one of the least defined aspects of ABC. Without a knowledge of the required neutronic and chemical-transport parameters for the

ABC, alternative processing schemes that address valid reactor and processing problems cannot be proposed. The primary and alternate chemical-processing schemes that were chosen for the MSBR and MSRE were selected based on the criteria that were required efficient breeding of ^{233}U from ^{232}Th and to maintain the fuel-salt composition and fluoride chemical potential. Similar requirements are either not known or required for ABC. Regardless of the uncertainties of the proposed ABC system, an alternative separations technique that could be applied to chemical processing is centrifugation.

Centrifugation has been proposed to remove the low atomic weight (*i.e.*, light) fission product elements and the high-atomic-weight fission-products from the fissile material.[44] The principle of centrifugation is to create a gravitational field across a solution so that a concentration gradient according to light *versus* heavy atomic weight is established from the center of the centrifuge solution. In this application, fissile material is transported away from the center of the centrifuge, could be separated from the fission products, and then recycled to the core. The degree of separation between the fission products and fissile material could be enhanced by using a cascade of centrifuges. Fission products that are collected could be sent to an accelerator-driven waste burner. Centrifugation has been used extensively in separating mixtures that contain two phases, but application to single-phase separations is only recent.[44] The application of single-phase separations has focused on aqueous-solution, room-temperature systems and not on high-temperature molten-salt systems. The major disadvantage of the centrifuge method is that the additional chemistry and engineering problems that will be encountered with the high-temperature molten salt requires an extensive research and development program.

Another aspect of the chemical-processing subsystems of ABC that must be considered would occur if another fluid is chosen to carry the fissile material for the system; in this case the chemical processes would have to be tailored to the new fluid system. Processes that work well for molten-salt fluoride based systems probably would not work well for molten-salt chloride or molten carbonate systems.

5. Power-Conversion System

The reference power-conversion system for ABC was adopted from the MSBR design, which was in turn adopted from the Bull Run Steam Plant design[25]. The supercritical-steam power-conversion system (Figs. 3 and 14) consists of: a steam generator (actually a superheater because the feedwater is supercritical steam); a high-pressure turbine; a reheat steam preheater; a full-flow steam reheater; an intermediate-pressure turbine; a low-pressure turbine; and a (complex) feedwater demineralizer/heater system. This power-conversion cycle maximizes the benefits of the high temperatures available from the molten salt and achieves a thermal-conversion of $\eta_{TH} = 0.44$. The use of the supercritical-steam cycle, however, introduces design difficulties. The required wall thicknesses of pipes and vessels required to contain this 25-MPa steam limit the diameter of piping and steam-generator components and, therefore, limit design options. Concerns of higher pressures in the steam generator also introduces concerns of rupture and pressurization of the secondary coolant system.

As part of the MSBR program, alternate steam cycles were considered. None was pursued, however, because of the limited resources and the availability of detailed design information on the supercritical-steam cycle from the Bull Run design. Additional analysis may show that the reduction in cycle efficiency that would accompany a change to lower-pressure steam is more than offset by the reduced complexity of the steam system. This analysis must also include an examination of alternative secondary coolants. A combination of the ternary eutectic as a secondary fluid and a lower-pressure steam power conversion cycle is deserved of further analysis. Although some disagreement exists (??), it may be possible to eliminate the complex feedwater heating system because of the lower melting point of the tertiary eutectic.

Another option for the power conversion system is to use helium or nitrogen in a closed cycle. Use of either gas coolant would require larger heat exchangers, but those for the nitrogen cycle would be even larger. Some preliminary calculations show that for the assumed conditions, nitrogen may yield a slightly higher efficiency. This gain, however, would have to be balanced against the additional capital requirements of the larger equipment and containment. In this scenario, the entire power conversion system would be placed inside containment. Only the final heat sink water and the power lines would penetrate the containment. Additional analyses are required to determine the cycle efficiencies, costs, tritium removal capability, safety, *etc.* of the helium (or nitrogen) turbine power cycle.

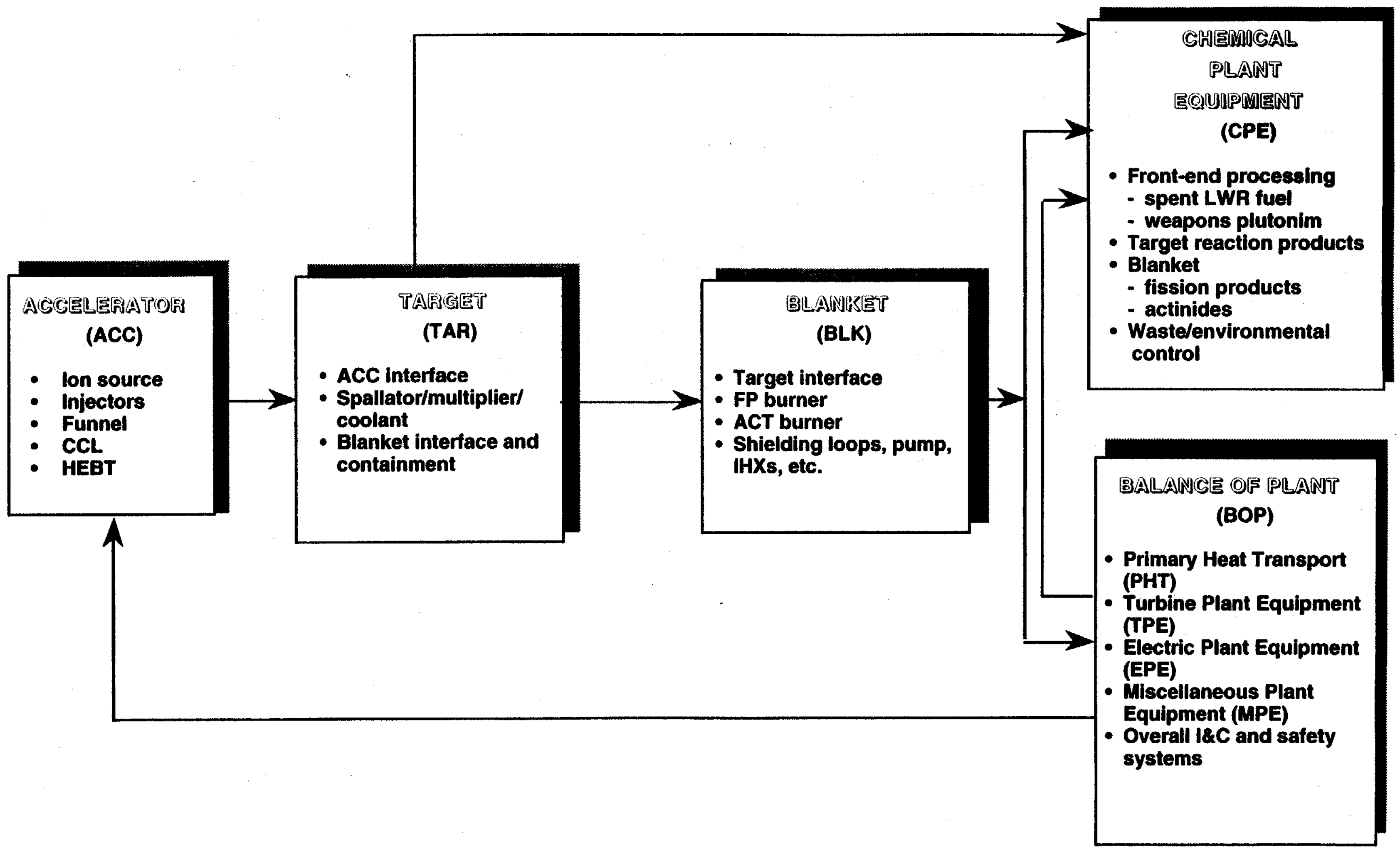


Figure 1. Top-level ATW/ABC systems diagram showing main subsystems and main mass/power flows.

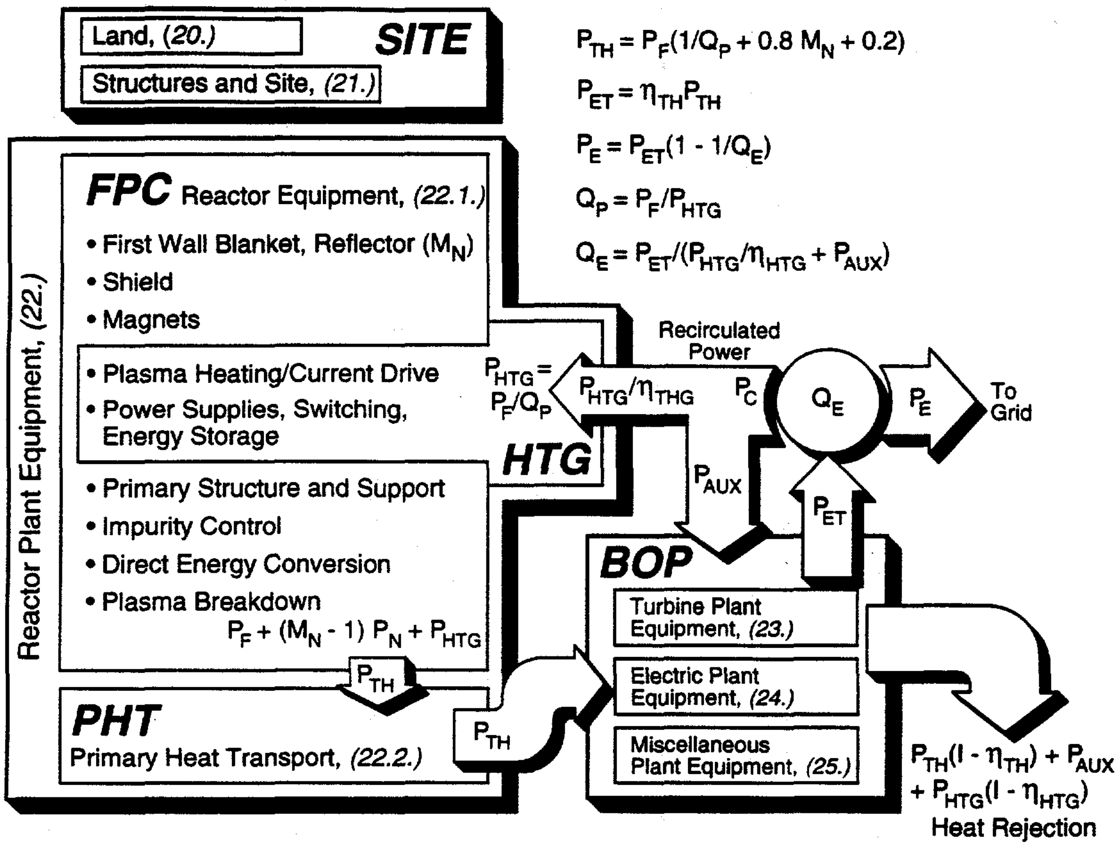


Figure 2. Re-expression of subsystems flows depicted in Fig. 1 showing main power flows and arranged according to EEDB[12,13] Program of Cost Accounts

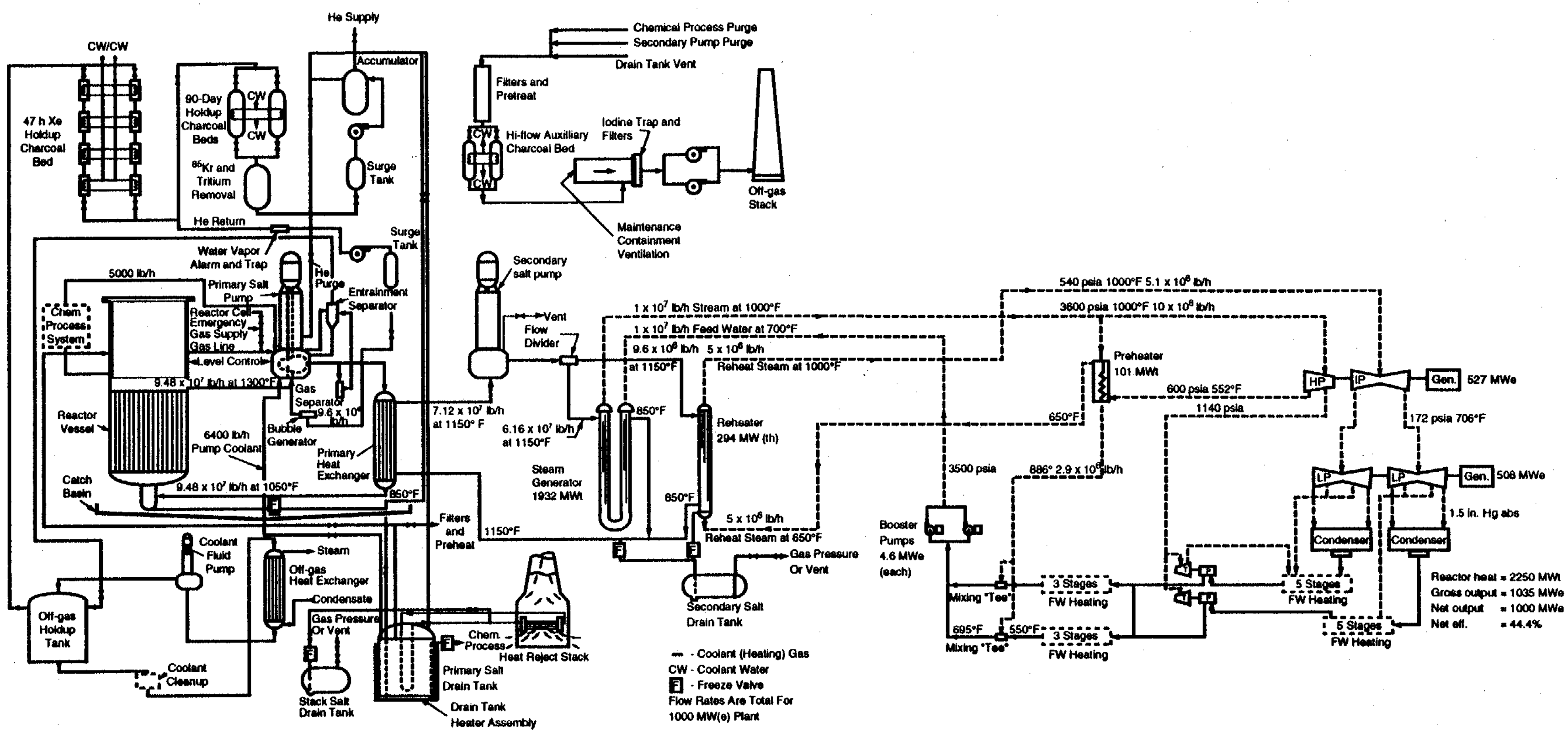


Figure 3. Composite power and mass flow diagram for the conceptual Molten-Salt Breeder Reactor (MSBR)[10] and adapted to the basecase ABC plant layout.

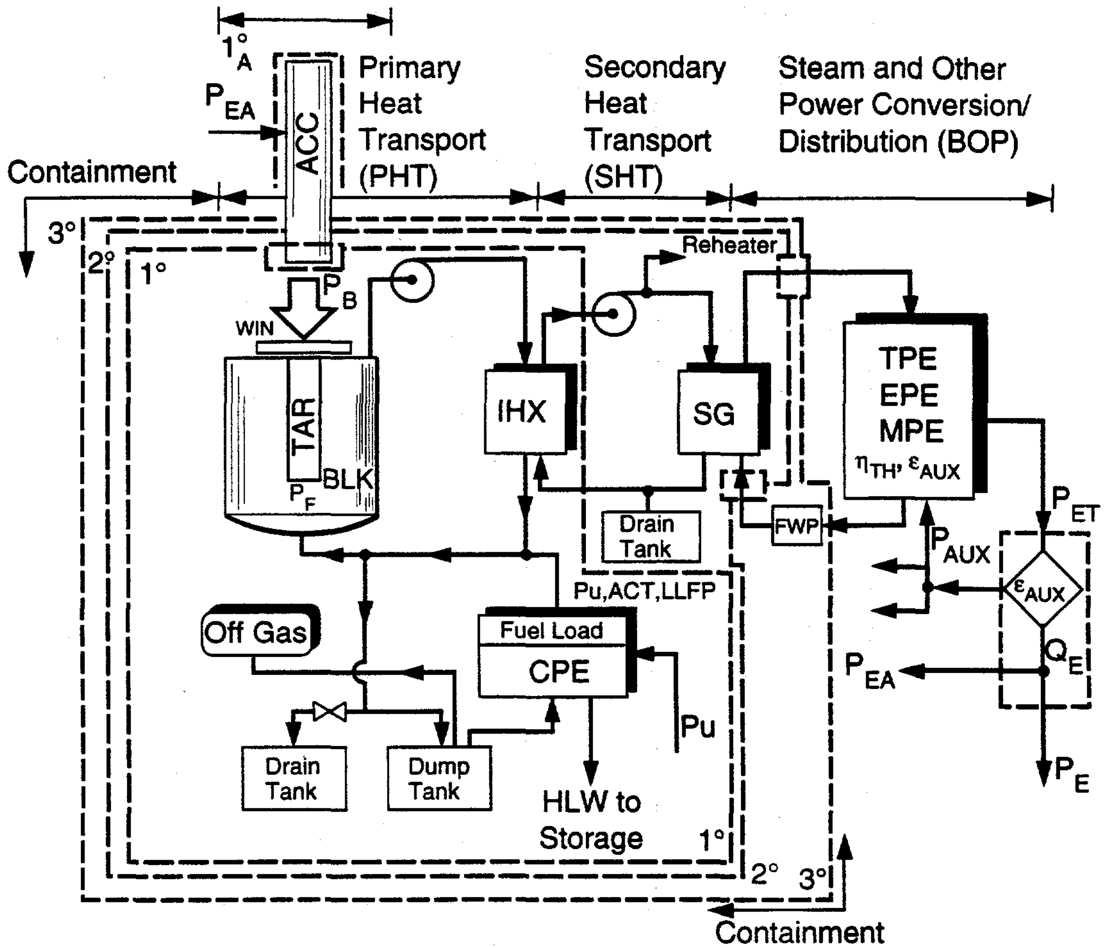


Figure 4. Essential elements of the ABC system for "deep-burn" weapons-plutonium disposition and net power production, showing main power and mass flows as well as three-level containment philosophy

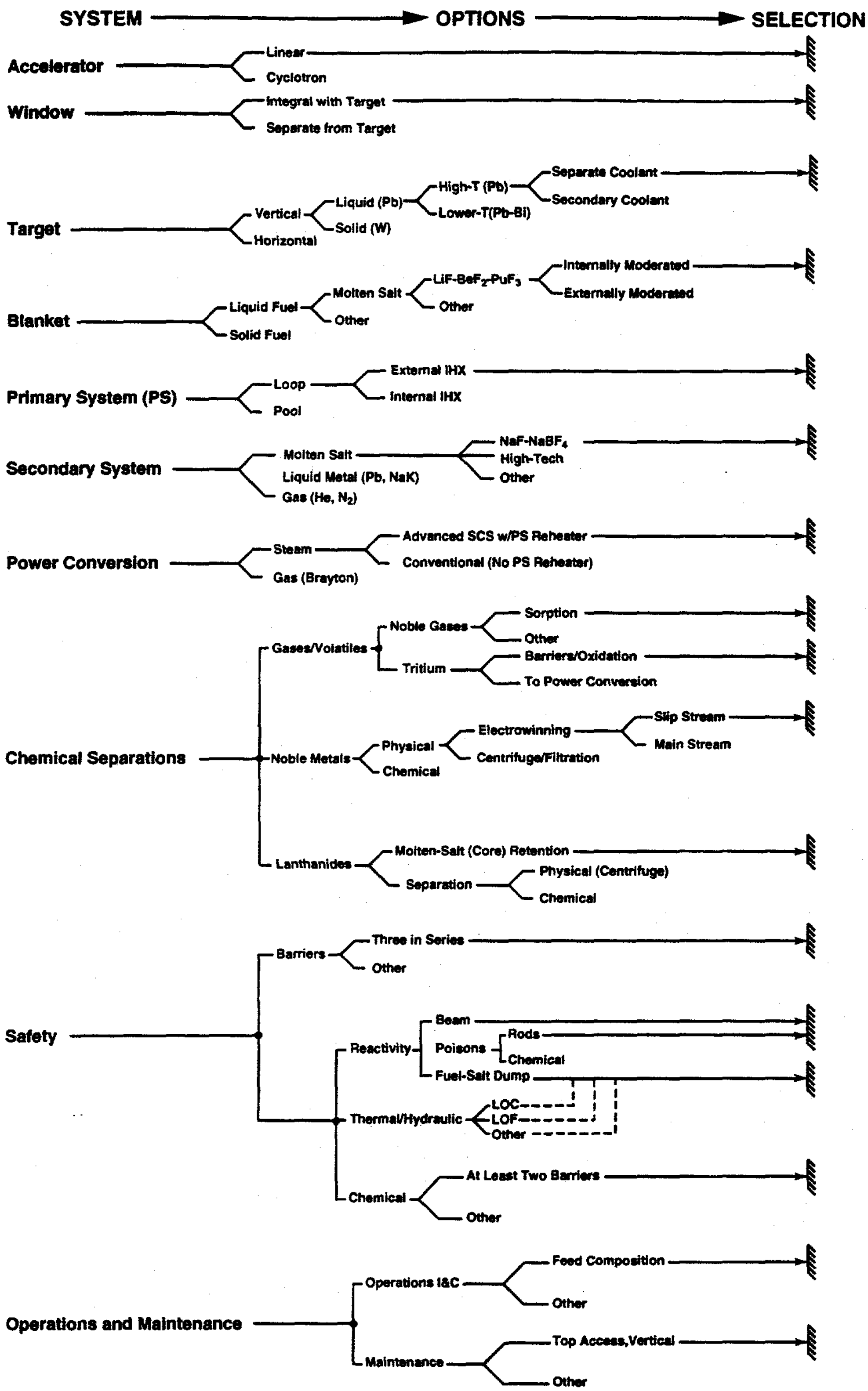
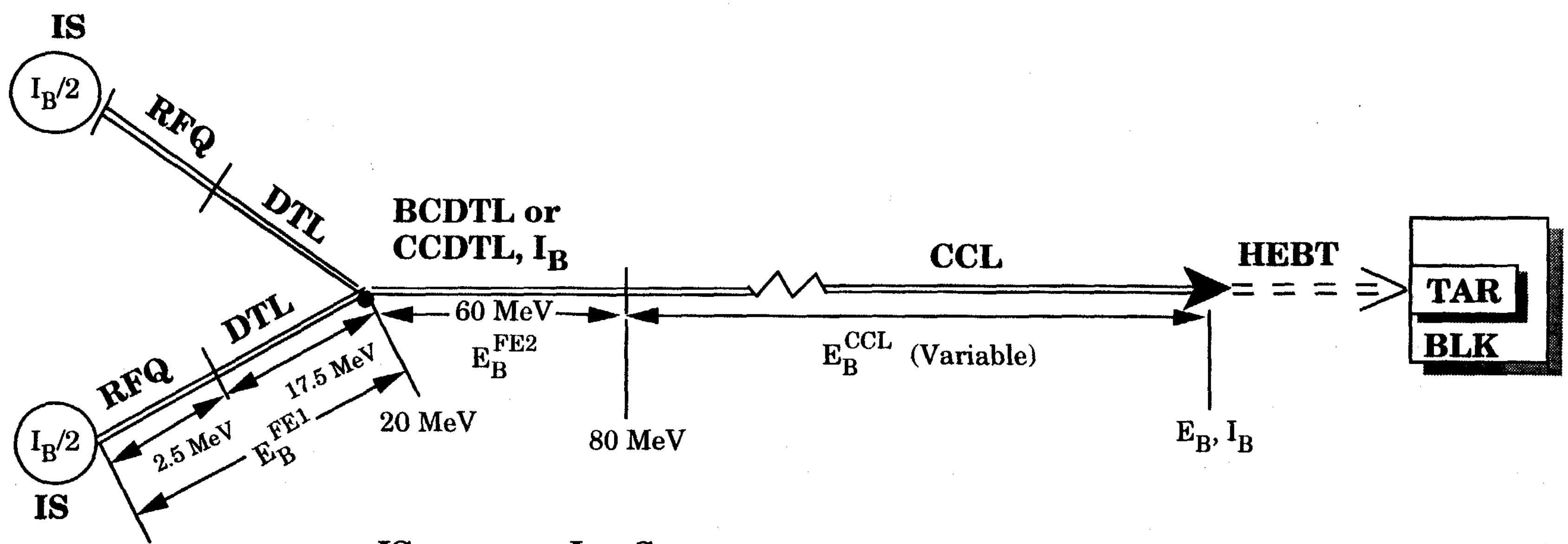


Figure 5. "Top-level" options diagram for ABC, illustrating process used to focus onto the base case used to generate preliminary ABC plant layout.



IS	=	**Ion Source**
RFQ	=	**Radio-Frequency Quadrupole**
DTL	=	**Drift-Tube Linac**
BCDTL	=	**Bridge-Coupled Drift-Tube Linac**
CCL	=	**Coupled-Cavity Linac**
HEBT	=	**High-Energy Beam Transport**
TAR	=	**Target**
CCDTL	=	**Coupled-Cavity Drift-Tube Linac**

Figure 6A. Linear accelerators proposed to drive ABC "top-level" systems diagram of ABC accelerator, showing main components.

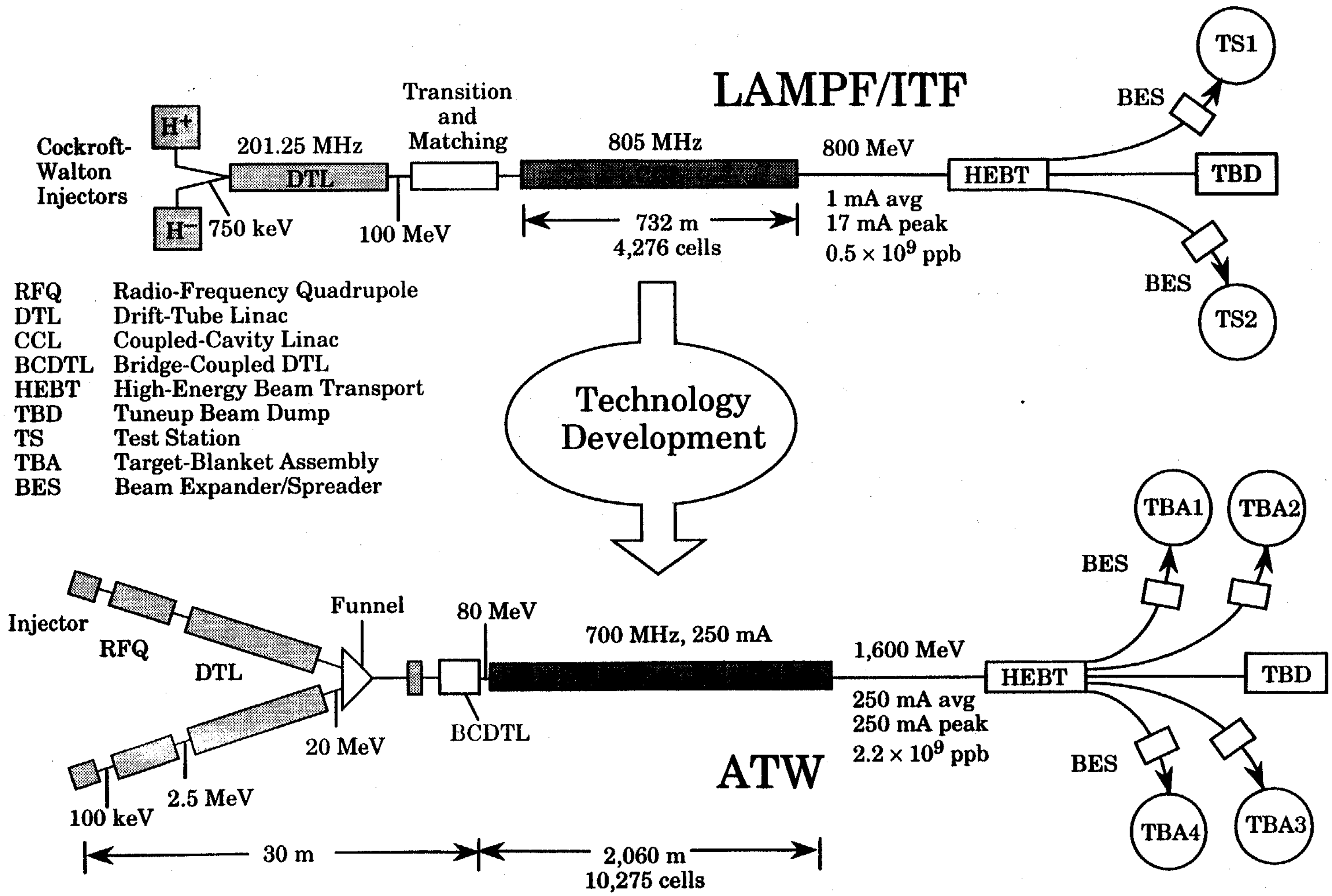


Figure 6B. Linear accelerators proposed to drive ABC:
Graphical illustration of technology development required to proceed from present LAMPF,[16,17] through an Integrated Test Facility,[18] and to a 1.6-GeV high-current accelerator.

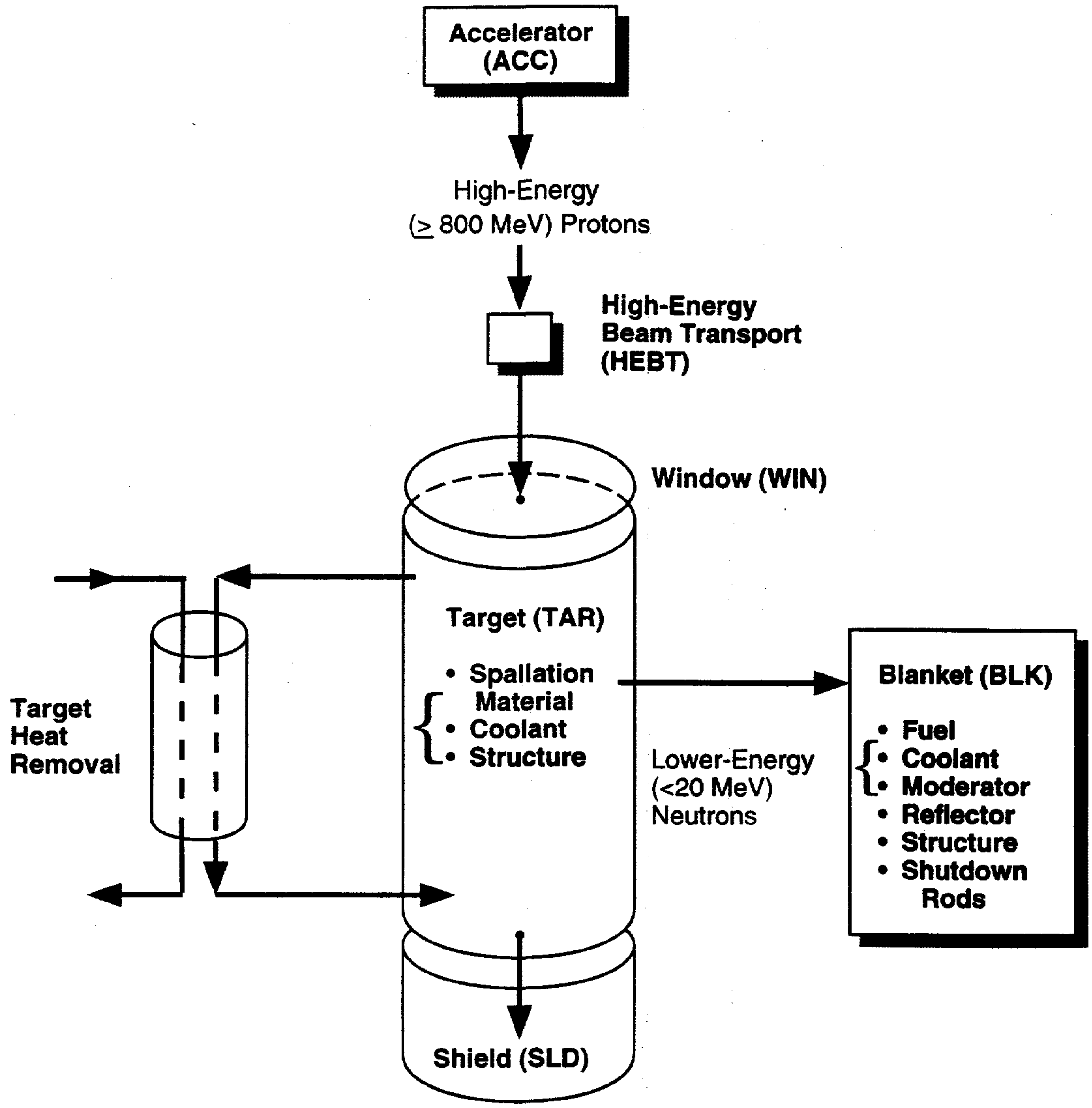


Figure 7. Schematic diagram illustrating target functional performance and connectivity with key ABC subsystems.

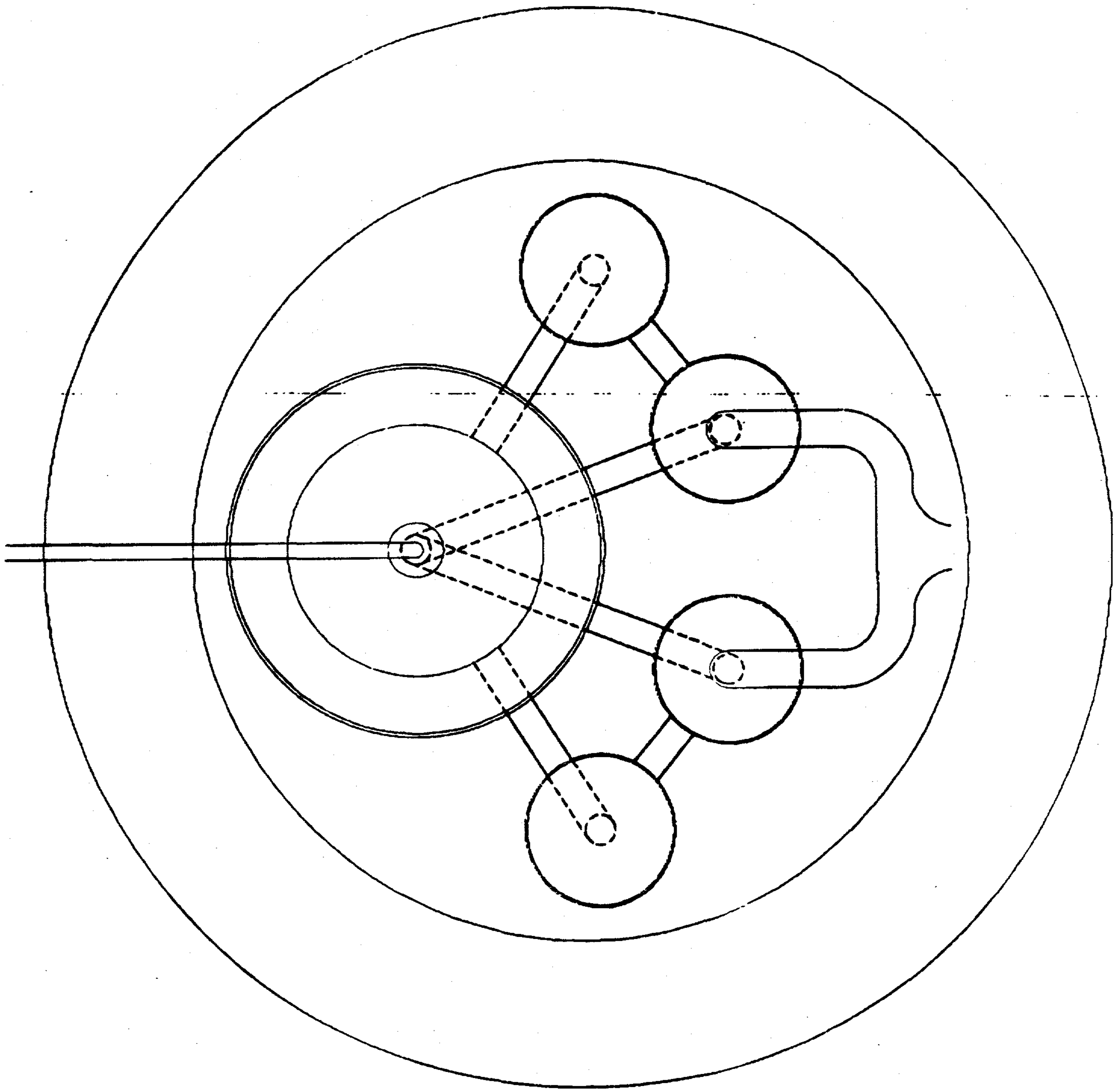


Figure 8. Plan view of Primary System: Core (Target/Blanket/Moderator/ Reflector, Reactor Vessel); Fuel-Salt Pumps; Intermediate Heat Exchanger; and interconnecting piping.

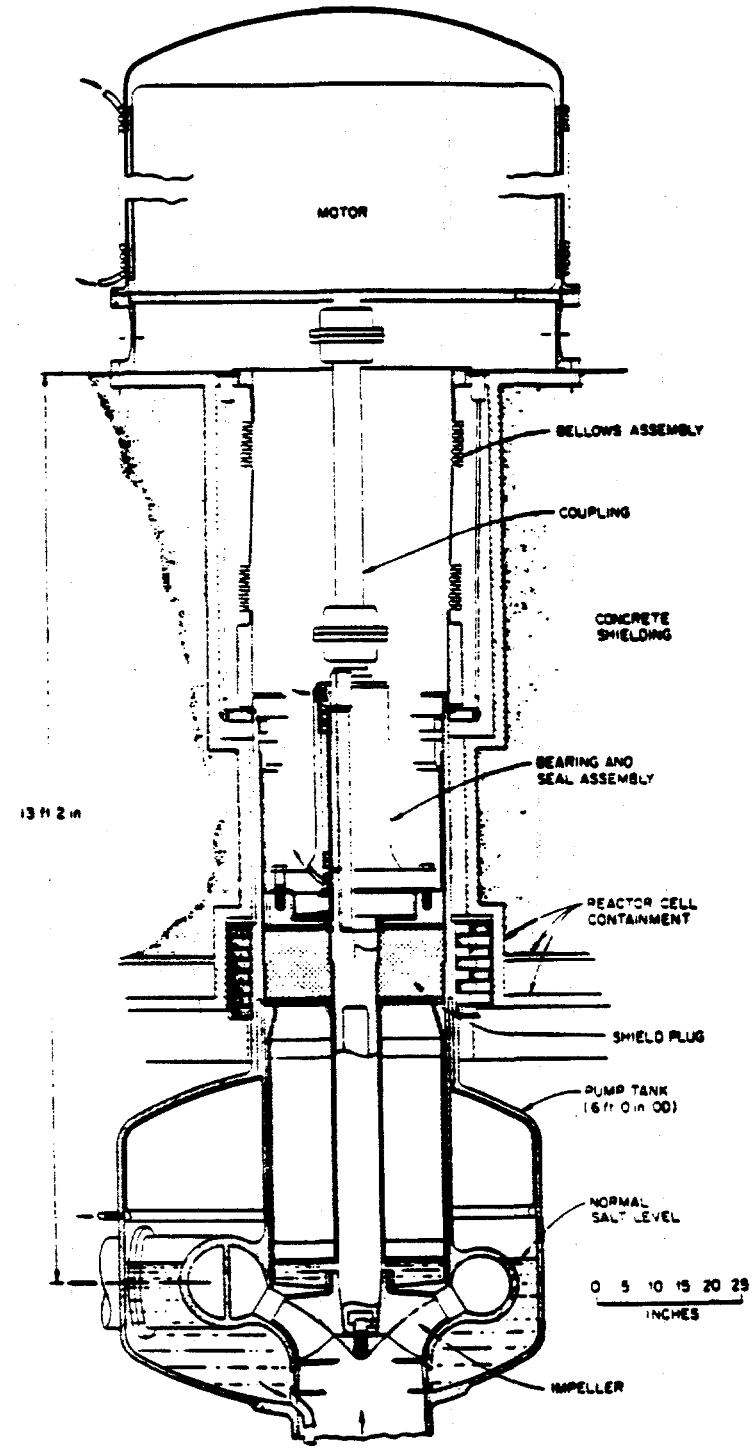


Figure 9. Detailed view of Fuel-Salt Pump; Secondary Coolant-Salt Pump is similar.[10]

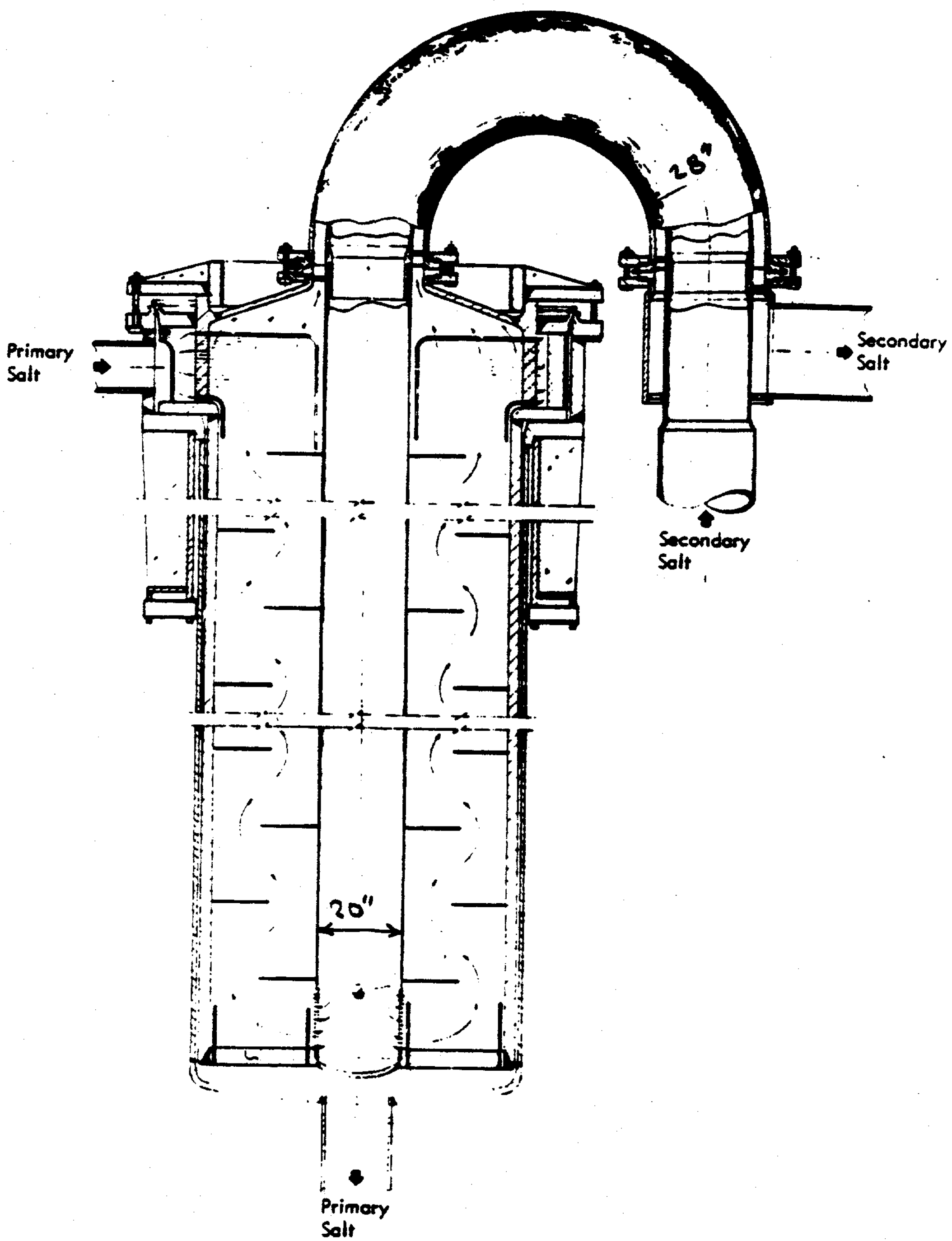


Figure 10. Detailed view of Intermediate Heat Exchanger.[10]

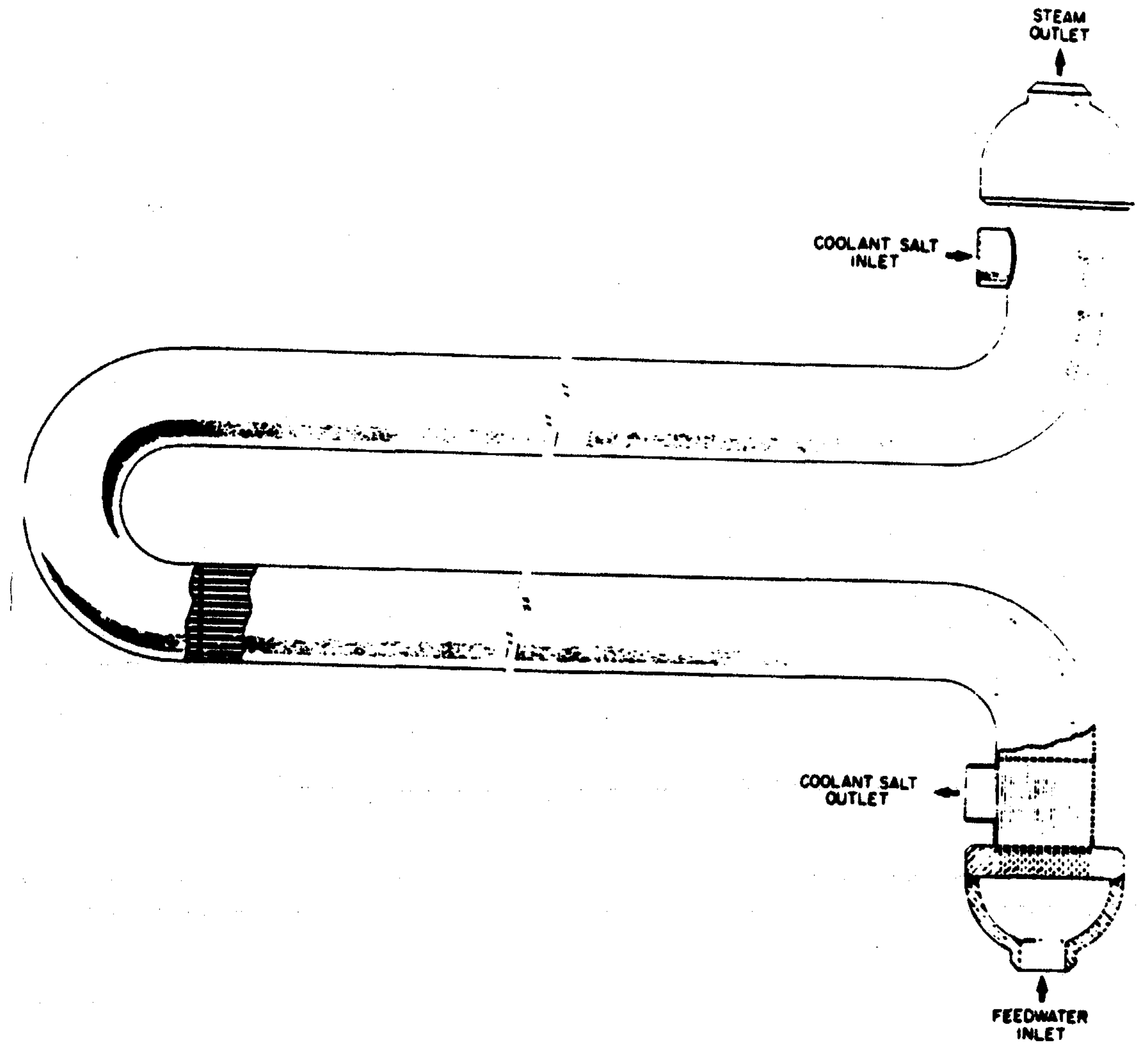


Figure 11A. Detailed view of Steam Generator: From MSBR conceptual design.[10]

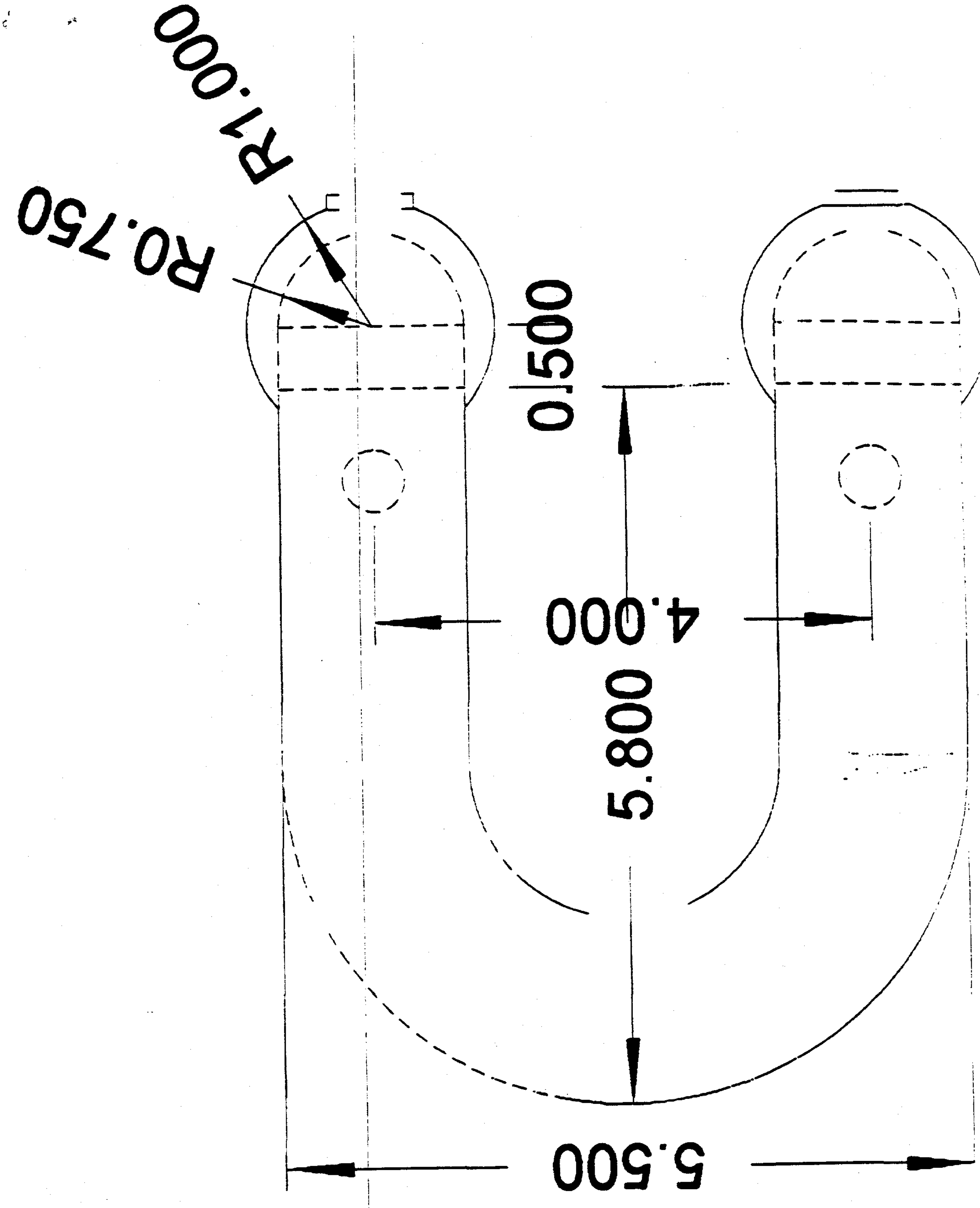


Figure 11B. Detailed view of Steam Generator: Scaled from MSBR[10] to meet ABC requirement.

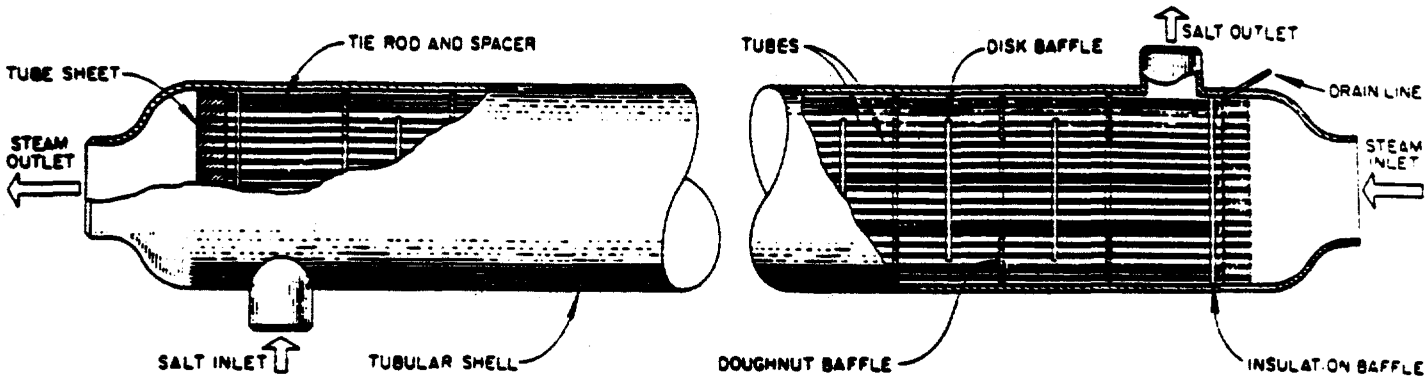


Figure 12.A Detailed view of Steam Reheater for ABC; scaled from MSBR conceptual design[10]

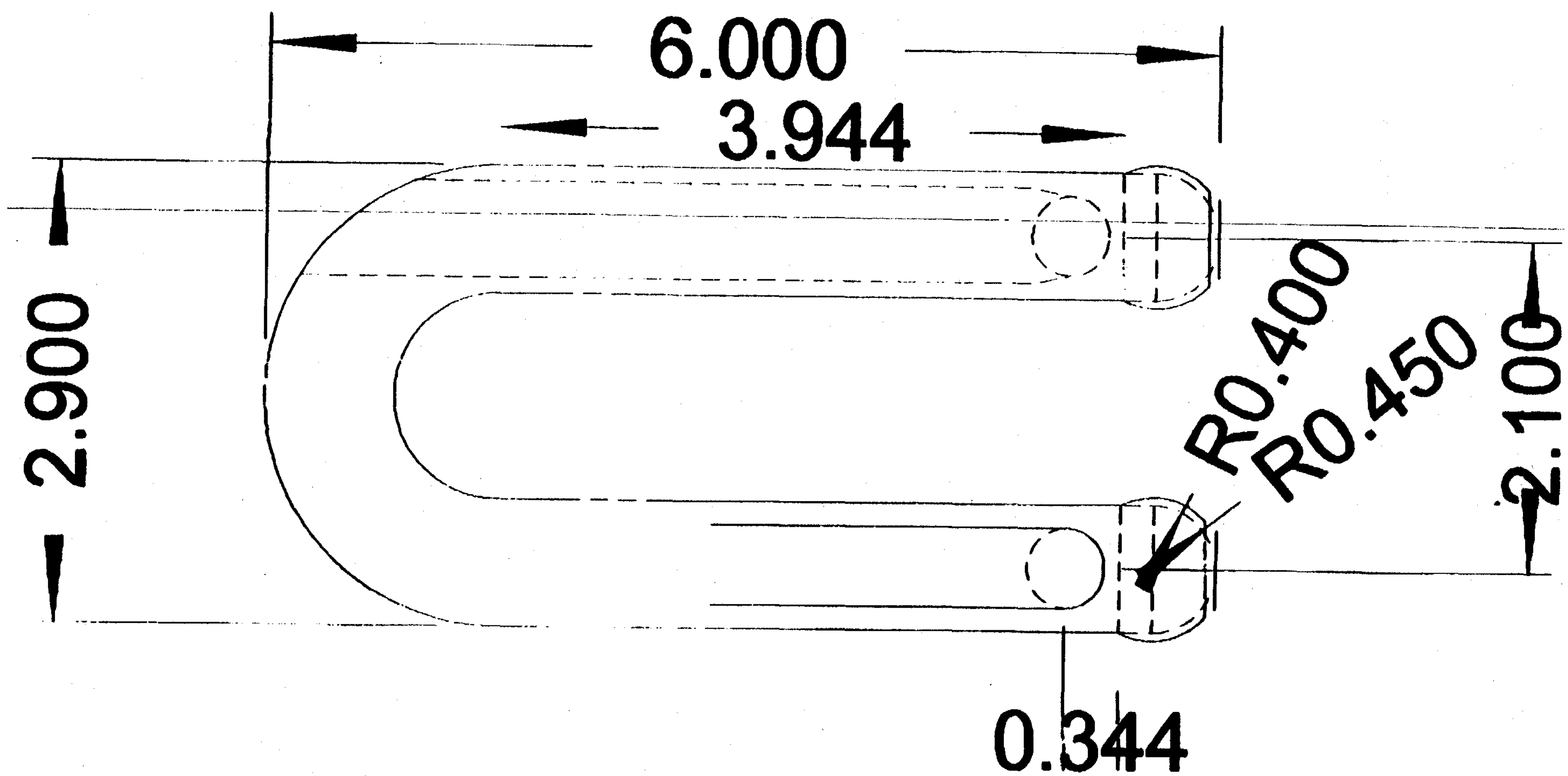


Figure 12B. Detailed view of Steam Reheater for ABC; scaled from MSBR[10] to meet ABC Requirements.

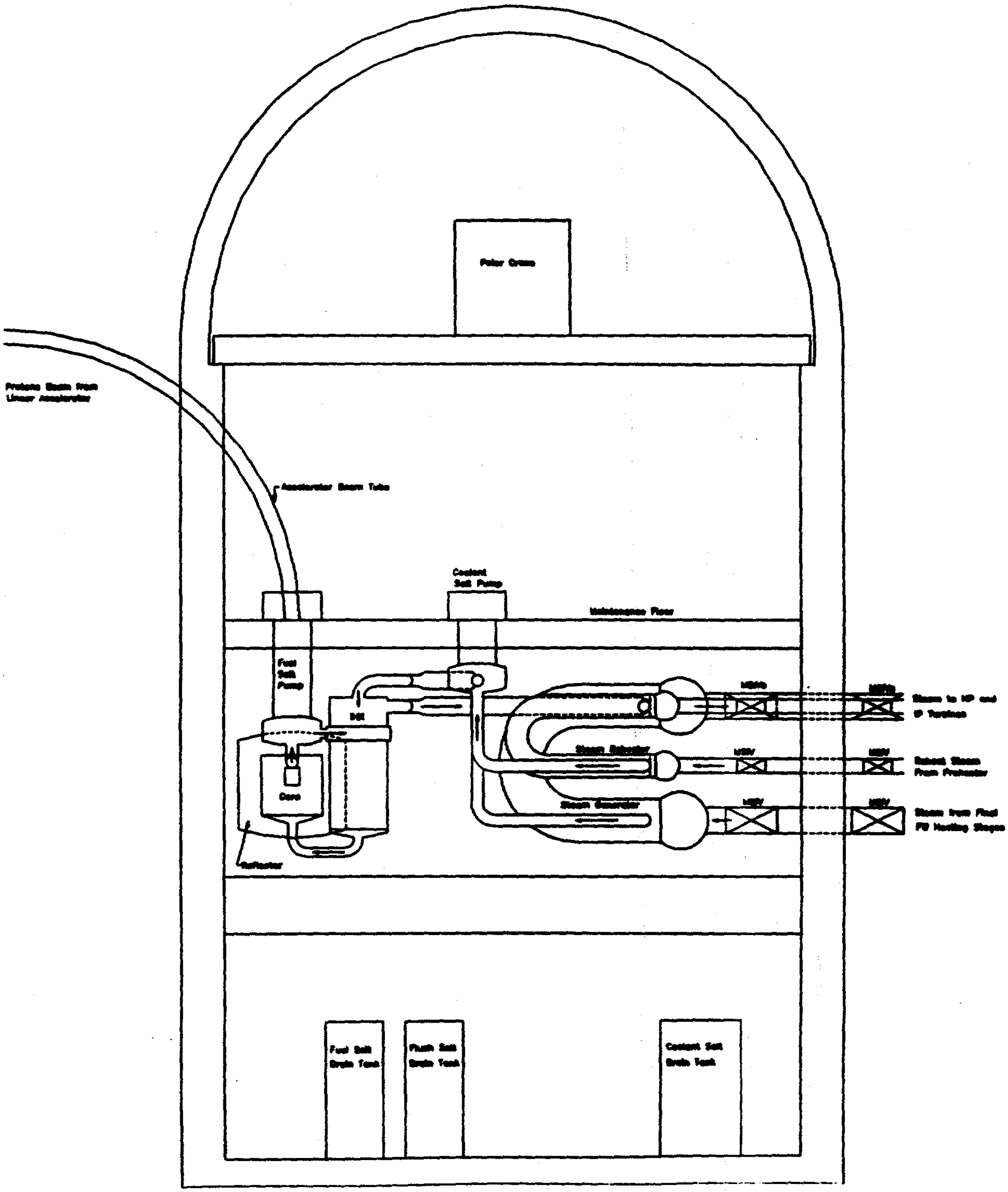


Figure 13A. Plan and elevation views of ABC plant layout: Elevation view.

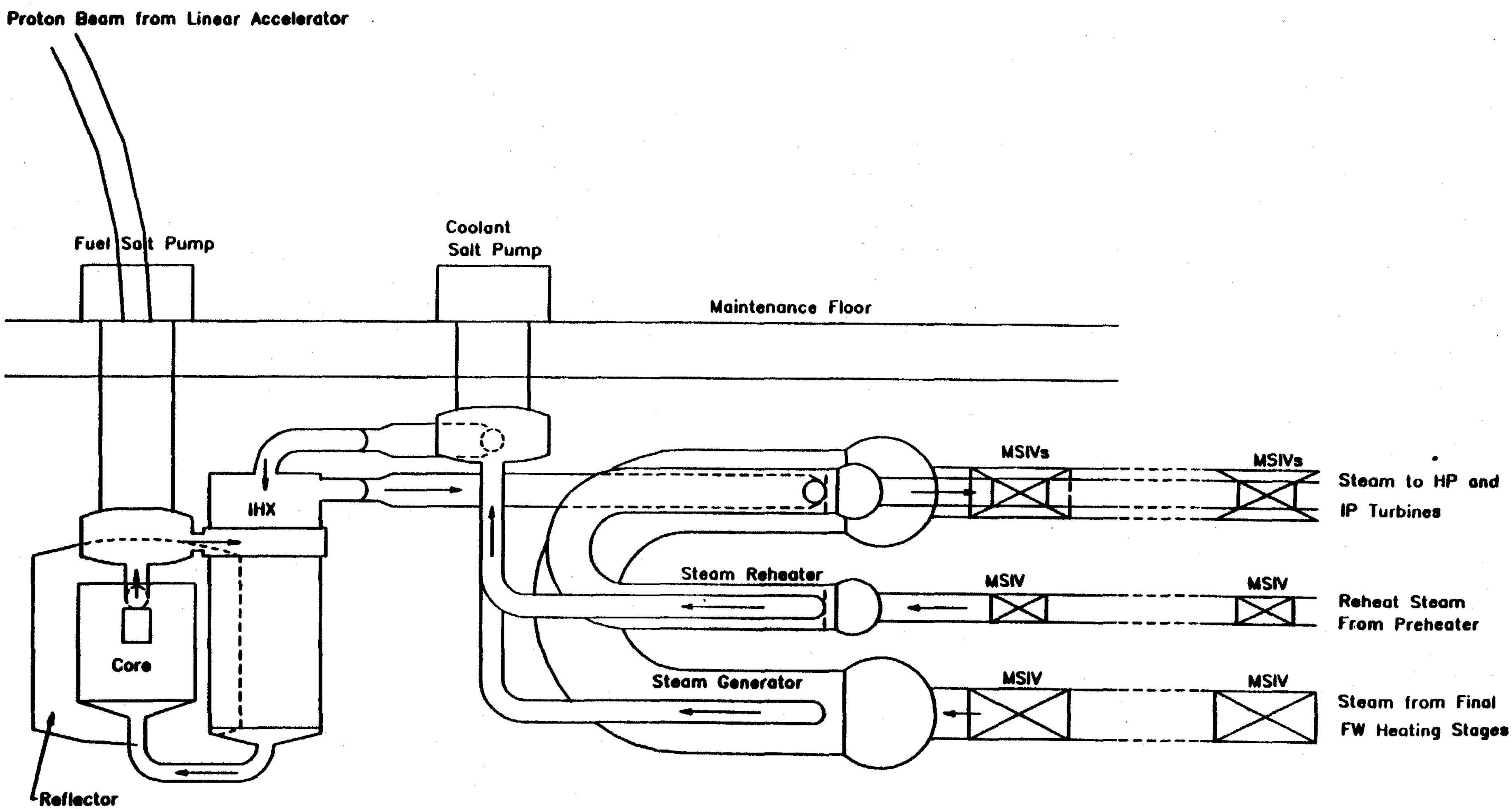


Figure 13B. Plan and elevation views of ABC plant layout: Elevation view (detailed).

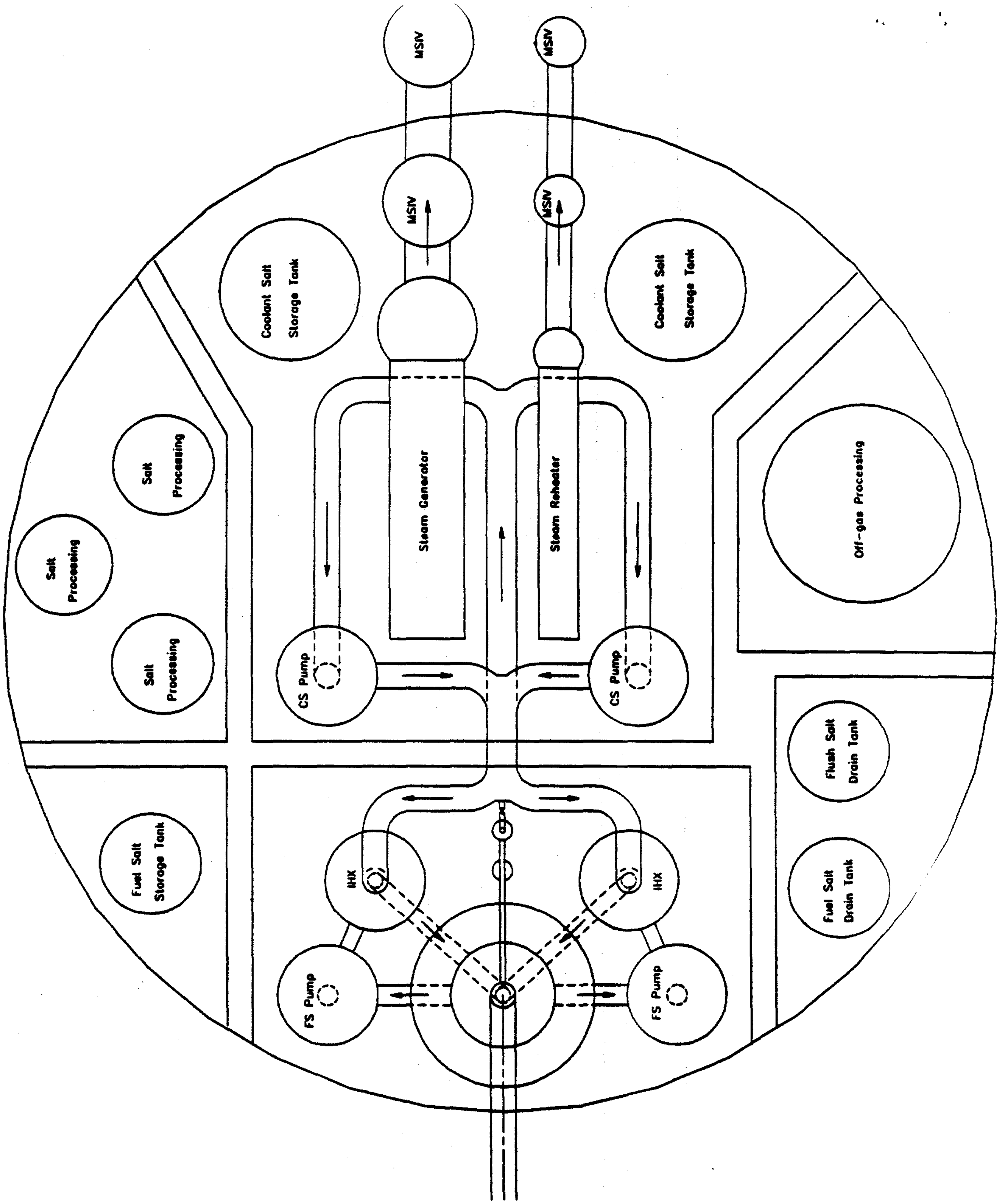


Figure 13C. Plan and elevation views of ABC plant layout: Plan view.

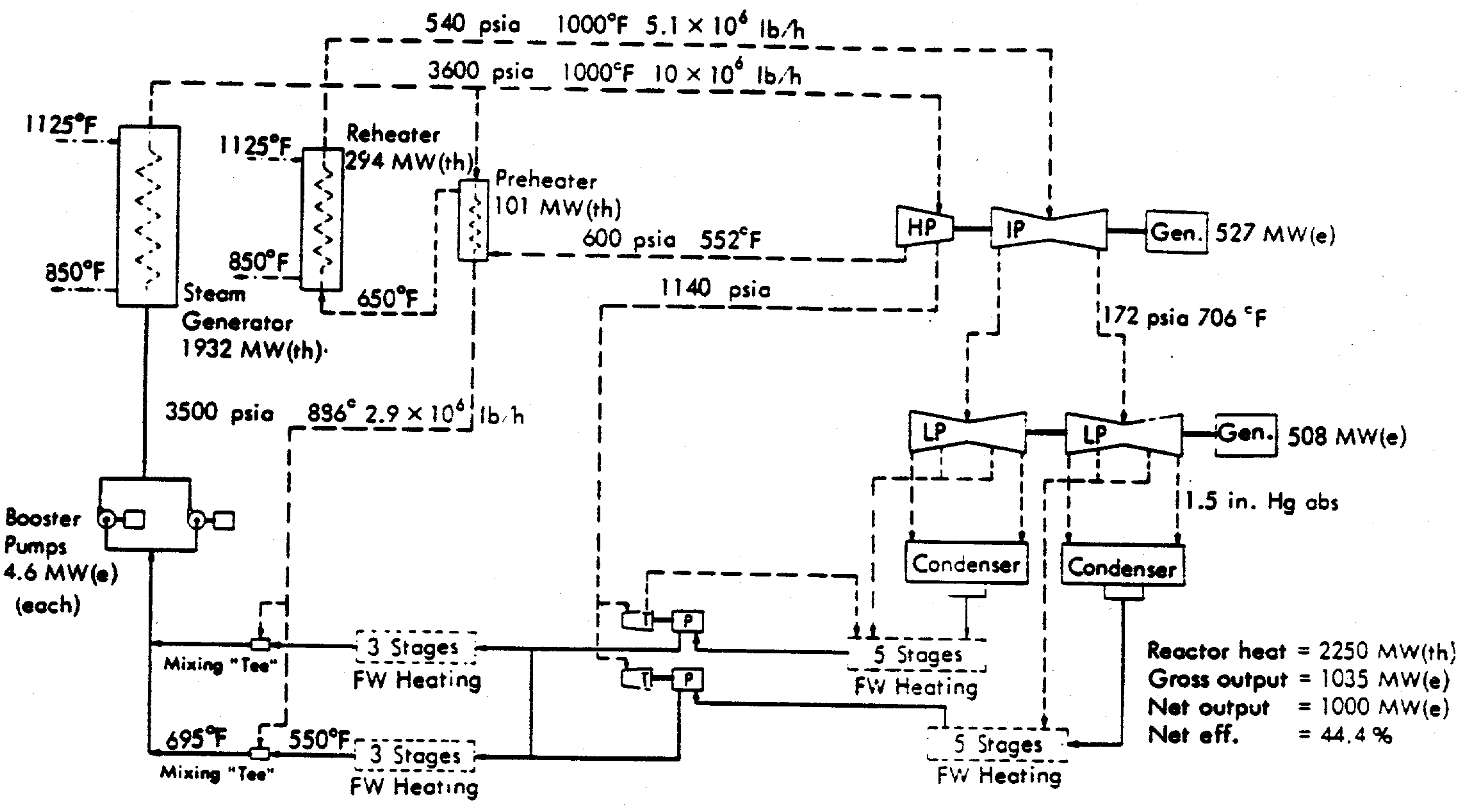


Figure 14. Secondary-coolant, steam-generation, and power conversion systems anticipated for ABC, as adapted from the supercritical-steam system[25] used in the MSBR conceptual design.[10]

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NOMENCLATURE [mks units only, with anything else enclosed by parentheses in text]

a(m)	Beam-tube radius
ABC	Accelerator-Based Conversion
ACC	ACCelerator
ACS	Absorption Cross Section
ACT	ACTinide
ADEP	Accelerator-Driven Energy Production
ADTT	Accelerator-Driven Tritium Technologies
ATW	Accelerator Transmutation of (nuclear) Waste
ATWS	Anticipated Transient Without SCRAM
ATWE	ATW Experiment
AUX	AUXiliary
B(T)	Magnetic field
BLK	BLanKet (moderator, fuel salt, reflector, structure)
BCDTL	Bridge-Coupled DTL
BES	Beam Expander/Spreader
BM	Bending Magnet
BOL	Beginning Of Life
BOP	Balanced of Plant
BTIV	Beam-Tube Isolation Valve
C_j(M$)	Cost of j^{th} component
c(m/s)	Speed of light, 3×10^8
c_j($/x)	Unit cost of i^{th} component, x = kg, W, *etc.*
CCDTL	Coupled-Cavity DTL
CCL	Coupled-Cavity Linac
COR	CORe [target, blanket (moderator, reflector, structure, salt)]
CPE	Chemical Plant Equipment
CR	Control Rod
CSP	Coolant-Salt Pump
CS	Confinement Systems
CSS	Core Support Systems
DTL	Drift-Tube Linac
e(J/eV)	Electronic charge, 1.6021×10^{-18}
E_B(MeV/p)	Proton beam energy
E_o(GeV)	Proton rest-mass energy
E_B^o(Mev/p)	Target yield fitting parameter
E_F(Mev/f)	Fission energy release
E_n(MeV/n)	"Wall-plug" energy to create a neutron
ECRH	Electron Cyclotron Resonance Heating
EEDB	Energy Economic Data Base
$(E_n)_\infty$(MeV/n)	Normalizing parameter, $y/(\eta_{DC}\eta_{RF}\eta_{WG})$
EOL	End Of Life
EPE	Electric Plant Equipment
f_{BU}	Plutonium burnup fraction
f_D	Proton-beam duty factor

f_{FS}	Volume fraction of fuel salt
FE	Front End (accelerator)
FF	Fluid Fuel
FP	Fission Products
FSP	Fuel-Salt Pump
FWP	FeedWater Pump
G(MV/m)	"Real-estate" acceleration gradient
$G_j(m^3/s)$	Volumetric flow rate
Hj(m)	Height of j^{th} system
HEBT	High-Energy Beam Transport
HEU	Highly Enriched Uranium
HLW	High Level Waste
$I_B(A)$	Proton beam current
I*(A)	Cavity → beam conversion efficiency factor, f_D $G/R_s/\cos\phi$
IHX	Intermediate Heat eXhanger
ITF	Integrated Test Facility[18]
$\ell_{EXP}(m)$	beam expansion distance
$j(MA/m^2)$	Conductor current density
k_{eff}	Blanket neutron multiplication
$L_j(m)$	Length/height of j^{th} system
LAMPF	Los Alamos Meson Physics Facility
Linac	Linear accelerator
LLFP	Long-Lived Fission Product
LLW	Low-Level Waste
LMFR	Liquid-Metal Fast Reactor
LWR	Light-Water (fission) Reactor
$L_{TUB}(m)$	Total tube length
M	Blanket fission power multiplication, $k_{eff}/(1 - k_{eff})$
M_{BM}(tonne)	Mass of beam-bending magnet
M_{TAR}(tonne)	Mass of target
$\dot{M}_{FS}$(kg/s)	Fuel-salt mass flow rate
M_{Pu}(kg)	Mass of plutonium to be destroyed
MBIV	Main Beam(let) Isolation Value
MHR	Modular Helium-cooled Reactor
MOD	MODerator
MPE	Miscellaneous Plant Equipment
MS	Molten Salt
MSBE	Molten-Salt Breeder Experiment
MSBR	Molten-Salt Breeder Reactor
MSRE	Molten-Salt Reactor Experiment
MSIV	Main Steam Isolation Valve
N_A	Avagadro's number, 6.0249×10^{26} entites/mole
N_{ABC}	Number of accelerator units
N_{BLK}	Number of target-blanket modules per accelerator unit
N_{INJ}	Number of injectors
N_{TUB}	Number of tubes
O&M	Operations and Maintenance

P^*(MW)	Beam efficiency parameter, $E_B^o I^*$
P_{AUX}(MWe)	Auxiliary (non-accelerator) plant power
P_B(MW)	Beam power
P_c(MWe)	Recirculating power, $P_{EA} + P_{AUX}$
P_E(MWe)	Net-electric power
P_{EA}(MWe)	Electrical power to accelerator
P_{ET}(MWe)	Gross or total electrical power
P_F(MW)	Fission power
P_{TH}(MW)	Thermal power to thermal-to-electric conversion, $\sim P_F$
P_Ω(MW)	Resistive power to RF cavity wall
PHT	Primary Heat Transport
POD	Point Of Departure
ppb	protons per bunch
p_f	Plant availability factor
p'(GeV)	Beam momentum, pc/e
r_B(m)	Beam radius
R_{TAR}(m)	Target radius
R_j(m)	Radius of j^{th} system
R_s(MΩ/m)	CCL shunt resistance
R&D	Research and Development
RF	RadioFrequency
RFL	ReFLector
RFP	RF Power
RFQ	RF Quadrupole
RT	Room Temperature
RTC	Reactivity Temperature Coefficient
RPE	Reactor Plant Equipment
SAF	SAFety
SCS	SuperCritical Steam
SG	Steam Generator
SHT	Secondary Heat Transport
SL	Steam Line
SLD	SHieLding
SLDA	Accelerator SHieLding
SP	Space Power
SR	Steam Reheater
T_{LIF}(yr)	Chronological time during which plutonium is disposed
TAR	TARget
TBA	Target-Blanket Assembly
TPE	Turbine Plant Equipment
TUN	TUNnel
UTS	Ultimate Tensile Strength
v_{FS}(m/s)	Fuel-salt flow velocity
V_j(m^3)	Volume of j^{th} system
VSL	VeSseL
WIN	WINdow
WG	RF WaveGuide

Y(n/p)	Net target neutron yield
YS (MPa)	Yield Strength
y(MeV/n)	Target yield fitting parameter
z(m)	Axial position
α_j	Bending magnet parameters
β	Parameter, $(E_F/\nu)/(E_B/Y)$
δ(m)	Conductor radius
ΔF(MJ/mole)	Free-energy change
ε	Recirculating power fraction, $(P_{AUX} + P_{EA})/P_{ET}$
ε_{ACC}	Accelerator power fraction, P_{EA}/P_{ET}
ε_{AUX}	Auxiliary (non-accelerator) power fraction, P_{AUX}/P_{ET}
η(ohm/m)	Resistivity of beaming magnet windings
η_A	Accelerator "wall-plug" efficiency
η_B	Cavity RF $\rightarrow$ beam efficiency
η_{DC}	AC $\rightarrow$ DC conversion efficiency
η_{RF}	DC $\rightarrow$ RF conversion efficiency
η_{WG}	RF $\rightarrow$ cavity RF transport efficiency
η_p	Net plant efficiency, $\eta_{TH}(1 - \varepsilon) = P_E/P_{TH}$
η_{TH}	Thermal-to-electric conversion efficiency
ϕ	phase angle between RF and proton beam bunch
ν	neutrons released per fission
ρ_j(kg/m^3)	density of jth component
μ_o(h/m)	premeability of free space, $4\pi\times10^{-7}$ h/m

Table I Summary of Benefits and Discriminating Features of a Driven (Subcritical, $k_{eff} < 1$) Fluid-Fuel (FF) System for Accelerator-Based Conversion (ABC) of Global Plutonium Inventories[3,9]

- Robust safety margins to reactivity variations caused by:
  - fissile-fuel burnup and fission-product burn-in
  - inadvertent reactivity insertions

- Looser/more-flexible neutron economy for $k_{eff} < 1$, resulting in:
  - destruction of LLFPs $\rightarrow$ reduced long-term dose
  - decreased fuel cleanup rate
  - fuel-form flexibility

- Increased fissile-fuel (Pu) burn-down, allowing:
  - phased deep burn using HEU
  - reduced materials specification
  - reduced handling of active core (no fuel shuffling)

- Reduced chemical processing with emphasis on physical separation:
  - gas-phase separation of xenon and krypton
  - surface collection of noble and semi-noble metals
  - batch precipitation of actinides, lanthanides, and other FPs (infrequent molten-salt processing to LLW)

⇨ SAFER, CLEANER, MORE-FLEXIBLE PROCESS WITH DEEP BURN AND REDUCED "DEEP DOSE" TO FUTURE POPULATIONS

Table II "Top-Level" Subsystem Breakdown for Molten-Salt ABC

• Site, Buildings, and Structures - Site - Accelerator Tunnel (TUN) - Containment Systems (CS) - Containment Dome Atmospheric Control - Containment Penetrations (incl. MSIVs) - Cell Ctaoninment Structure - Cell Atmosphere Control[k] - Cell Liner (Thermal) Shield and Cooling - Beam-Tube Isolation Value (BTIV) - Other Structures	CONTAINMENT
• Accelerator Systems ACC) - Ion Source (IS) - Radio-Frequency Quadrupole (RFQ) - Drift-Tube Linac (DTL) - Bridge-Coupled Drift-Tube Linac (BCDTL) - Couple-Cavity Linac (CCL) - High-Energy Beam Transport (HEBT) - Window (WIN) - Tunnel (TUN) - Shield (SLDA) -- Main Accelerator Structure -- HEBT/TAR-BLK - Accelerator Power Systems -- Power Conditioning (POWAE) -- RF Power (RFP) -- RF Power Delivery (WG) -- Thermal Power Discharge (POWAT) - I&C	ACCELERATOR EQUIPMENT
• Target (TAR) - Window (WIN) - Spallator/Coolant[a] - Structure/Decoupler[b] - High-Energy Neutron Shield - Gas Annulus Cooling/Monitoring Systems - I&C	TARGET
• Core (COR) - Target/Blanket (BLK) Decoupler - Blanket/Coolant - Moderator (MOD) - Reflector (RFL) - Vessel/Structure (VSL) - Control/Shutdown Rods (CR) - Shielding (SLD) - I&C • Primary (Fuel-Salt) Heat Transport (PHT) - Primary System Piping - Primary Pumps - Intermediate (Primary) Heat Exchanger (IHX) - I&C • Auxiliary Core Support Systems (CSS) - Fuel-Salt Drain Tank(s) - Dump Tanks - Freeze Valves - Afterheat Coolers - Drain-Tank Cooling System - Storage-Tank Cooling System - I&C	PRIMARY SYSTEMS

Table II "Top-Level" Subsystem Breakdown for Molten-Salt ABC (Cont-1)

• Chemical Plant Equipment (CPE)[c] - Offgas Control - Fission Product Plating, Particulates, and Smoke Control[d] - Tritium Control - Molten-Salt Chemistry (Redox) Control - Fuel Loading - Fuel-Salt Cleanup System[e] - Coolant-Salt Cleanup System[e] - Waste Output Preparation/Staging - I&C[f]	CHEMIAL PLANT EQUIPMENT	
• Secondary (Coolant-Salt) Heat Transport (SHT) - Coolant Pipes - Secondary Pumps - Steam Generator (SG) - Coolant-Salt Heaters[g] - Secondary-Salt Drain Tank - SG Rupture Protection[h] - I&C	HEAT REMOVAL	BALANCE OF PLANT
• Balance of Plant (BOP) - Steam Drum[i] - Turbine Plant Equipment (TPE) - Electric Plant Equipment (EPE) - Miscellaneous Plant Equipment (MPE) - I&C[j]		
• Central Control Systems (CCS) - Plant Integration, Status, and Control - Control Room(s) - Waste Management - Environmental Control	I&C	
• Cell Access/Maintenance[k] - Target (thimble) Replacement - Moderator Replacement - Reflector Replacement - Core Vessel Replacement - Primary-Pump Replacement - Piping Replacement - IHX Replacement - SG Replacement/	O&M	

Table II "Top-Level" Subsystem Breakdown for Molten-Salt ABC (Cont-2)

(a) Assumed here to be one in the same (*e.g.*, molten lead).

(b) Including target "thimble".

(c) As presently envisaged, the CPE would be a loose federation of systems designed to deal with:
- collection and trapping of volatile fission products (~25%).
- control, monitoring, and eventual removal of fission products that plate onto cooler, post-IHX surfaces (~25%).
- tritium control and collection prior to escape into the secondary coolant system and beyond.
- any chemical shimming needed to assure the molten-salt solubility of the remaining 50% of the fission products, as well as corrosion control throughout the PHT system; removal of a part of this remaining 50% of fission products by a combination of physical and chemical means remains to be specified.
- fuel preparation and loading into the PHT system.
- all on-line analytical chemistry and related diagnostics control the PHT and TAR/BLK systems.

(d) May also be part of offgas control system.

(e) Water removal, oxide removal, impurity removal ($NaBF_4$, *etc.*).

(f) Including on-line chemical analysis.

(g) Trace heaters used in steam cell, instead of oven-type heaters.

(h) Rupture-disc, blowdown diversion systems, *etc.* in event of SG rupture.

(i) Turbine must be capable of efficient operation with less than full steam flow, (*i.e.*, when one of the N_{BLK} modules is inoperable).

(j) Controls necessary to allow a trip of one module without shutdown of entire plant, may be complicated).

(k) Applies primarily to Reactor Cell; similar requirements anticipated for other cells [(*e.g.*, CPE (if any), SG, tanks, *etc.*)].

Table III. Specified and Derived ABC Parameters from Plant Layout Study

Overall Plant[a]	
Mass of weapons plutonium to be disposed, M_{Pu}(tonne)[b]	50.
Thermal energy value of plutonium to be disposed, GWyr[b]	128.
Time allowed to demonstrate disposition technology, yr	20.
Time to dispose, T_{LIF}(yr)	20.
Annual availability or plant factor, p_f	0.75
Thermal-to-electric conversion efficiency, η_{TH}	0.444
Total electrical power generation, $N_{ABC}\,P_{ET}$(MWe)	3,789.
Number of ABC units, N_{ABC}	3
Total electrical power generation per ABC unit, P_{ET}(MWe)	1,263.
Total thermal power generation per ABC unit, P_{TH}(MW)	2,844.
Number of Target/Blankets per ABC unit, N_{BLK}	4
Thermal power per Target/Blanket, P_{TH}/N_{BLK}(MW)	711.
Recirculating power fraction, $\varepsilon = P_c/P_{ET}$	0.15
• ACC recirculating power fraction, $\varepsilon_{ACC} = P_{EA}/P_{ET}$	0.12
• BOP recirculating power fraction, $\varepsilon_{AUX} = P_{AUX}/P_{ET}$	0.03
Net electrical power per ABC unit, $P_E(\text{MWe}) = (1 - \varepsilon)P_{ET}$	1,074.
Recirculated power, $P_c(\text{MWe}) = \varepsilon\, P_{ET}$	189.
• ACC power, $P_{EA}(\text{MWe}) = \varepsilon_{ACC}\, P_{ET}$	152.
• BOP power, $P_{AUX}(\text{MWe}) = \varepsilon_{AUX}\, P_{ET}$	38.
Accelerator "wall-plug" → beam efficiency, η_A	0.45
Beam power, $P_B(\text{MW}) = \eta_A\, P_{EA}$	68.4.
Beam power per Target/Blanket assembly, $P_B(\text{MW})/N_{BLK}$	17.1
Blanket multiplication	
• $M = k_{eff}/(1 - k_{eff}) = (P_F/P_B)/\beta$[c]	24.2
• k_{eff}	0.96
Beam current	
• Accelerator I_B(A)[d]	0.086
• Target, I_B/N_{BLK}(A)	0.021

Table III Specified and Derived ABC Parameters from Plant Layout Study (Cont.-1)

Accelerator

Number of injectors, N_{INJ}	1
Length of front-end, L_{FE}(m)	~20.
Efficiencies, $\eta_A = \eta_{DC}\, \eta_{RF}\, \eta_{WG}\, \eta_B$	0.45
• AC → DC, η_{DC}	0.90
• DC → RF, η_{RF}	0.65
• RF → cavity, η_{WG}	0.98
• cavity → beam, $\eta_B = 1/(1 + I^*/I_B)$	0.78
CCL parameters	
• "Real-estate" gradient, G(MV/m)	1.0
• Shunt resistance, R_s(MΩ/m)	55.
• Cosine of RF-bunch phase angle, $\cos\phi$	0.77
• Frequency, f(Mhz)	700.
• Efficiency factor, $I^*(A) = f_D\, G/R_s/\cos\phi$	0.024
• Duty factor, f_D	0.10
Accelerator length, L_{ACC}(m)	850.
High-Energy Beam-Transport length, L_{HEBT}(m)	100.(?)
Tunnel volume, V_{TUN}(m^3)	---
Support buildings	---
• Area, m^2	---
• Volume, m^3	---

Beam Entrance (Bend and Expander)[(e)]

Beam-tube radius, a(m)	0.10
Conductor radius, δ(m)	0.07
Beam radius of curvature, R(m)	2.83
Magnetic field, B(T)	1.58
Conductor current, I(MA/conductor)	0.37
Resistive power losses, P_Ω(MW)	1.50
Mass of conductor, M_{BM}(tonne)	1.10
Expander length, L_{EXP}(m)	10.

Table III Specified and Derived ABC Parameters from Plant Layout Study (Cont.-2)

Primary System

Target nominal dimensions	
• Diameter, D_{TAR}(m)	0.75
• Height, H_{TAR}(m)	0.6
Core[e] nominal dimensions	
• Diameter, D_{COR}(m)	3.5
• Height, H_{COR}(m)	3.5
• Volume, V_{BLK}(m^3)	33
Average fuel-salt fraction in core, f_{FS}	0.13
Core power densities	
• Average core, PD(MW/m^3) = $P_{TH}/N_{BLK}/V_{BLK}$	22.2
• Fuel salt, PD/f_{MS}(MW/m^3)	171
Total fuel salt volume, V_{MS}(m^3)	12.5
Fuel-salt temperatures (K)	
• Core inlet/IHX outlet	???
• Core outlet/IHX inlet	???
Fuel-salt flow rate, $\dot{M}_{FS}$(kg/s)	???
Fuel-salt pump (nominal) dimensions	
• Diameter, D_{FSP}(m)	2.0
• Height, H_{FSP}(m)	7.5
IHX (nominal) dimensions	
• Diameter, D_{IHX}(m)	1.75
• Height, H_{IHX}(m)	6.55
Fuel salt (residence) fractions	
• Core	0.34
• IHX	0.32
• Pumps	0.16
• Other	0.18
Loop-averaged power density, $\langle PD \rangle$(MW/m^3) = $P_{TH}/N_{BLK}/V_{MS}$	57.
Dump-Tank volume, V_{DT} (m^3)	45.
Reactor-Cell volume, V_{PS}(m^3)	850.

Table III Specified and Derived ABC Parameters from Plant Layout Study (Cont.-3)

Heat-Removal System

Coolant-salt volume, $V_{CS}(m^3)$	41.
Coolant-salt pump (nominal) dimensions	
• Diameter, $D_{CSP}(m)$	2.0
• Height, $L_{CSP}(m)$	7.5
Coolant-salt flow rates	
• Steam generator, $\dot{M}_{SG}(kg/s)$	???
• Steam reheater, $\dot{M}_{SR}(kg/s)$	???
• IHX, $\dot{M}_{IHX}(kg/s)$	???
Steam-generator (nominal) dimensions	
• Length, $L_{SG}(m)$	7.3
• Height, $H_{SG}(m)$	6.0
Coolant-salt temperatures (K)	
• SG/SR inlet	???
• SG outlet	???
• SR outlet	???
Steam-generator cell volume, $V_{SG}(m^3)$	1,800.
Steam flow rate, $\dot{M}_{SCS}(kg/s)$	???
Steam pressure, p_{SCS} (MPa)	25.9
Steam temperature, $T_{SCS}(K)$	810.

Chemical Plant Equipment

Annual fission product generation, $R_{FP}(kg/yr)$	1,667.
• gaseous	
– tritium	???
– noble gases	???
• noble and semi-noble metals	???
• lanthanides	???
Volume of processing equipment, V_{CP}^{j} (m^3)	
• gaseous	
– tritium	???
– noble gases	???
• noble and semi-noble metals	???
• lanthanides	???

Table III Specified and Derived ABC Parameters from Plant Layout Study (Cont.-4)

Containment Building/Envelop

Volume of containment, $V_{CB}(m^3)$	23,000.
Specific Volume, $P_{TH}/N_{BLK}/V_{CB}(MW/m^3)$	0.031
Supercritical-steam system	
• Thermal Power, $P_{TH}(MW)$	2,833.
• Number of loops, N_{SCS}	???
• Steam-generator temperatures, $T_{in}/T_{out}(K)$	???/810.
• Mass flow rate, $M_{SCS}(kg/s)$	???
• Pressure, $P_{SCS}(MPa)$	24.9
Turbine Plant Equipment	
• Number, N_{TPE}	4
Turbine ratings (MWe)	
• Gross rating	???
• Net rating	316.
• Gross electric power, $P_{ET}(MWe)$	1,263.(g)
• Net overall thermal conversion efficiency, η_{TH}	0.444.
Electric Plant Equipment	
• Net electrical power, $P_E(MWe)$	1,074.
• Recirculating power fraction, ε	0.15
• Plant efficiency, $\eta_p = \eta_{TH}(1-\varepsilon)$	0.377

(a) The parameters in this section of the table are presented in the order of determination.

(b) Assumed total distruction (fissioning) of M_{Pu} mass of weapons plutonium; burnup >90% however, will require use of highly enriched uranium (HEU) near end of life (EOL), increased accelerator power (decreased k_{eff}) or both.

(c) $\beta = [E_F/\nu]/[y/(1 - E_B^o/E_B)]$, where $y \simeq 30$ MeV/n and $E_B^o \simeq 200$ MeV/p are fitting parameters to the target neutron yield relationship, $Y(n/p) = (E_B - E_B^o)/y$; $E_F = 200$ MeV/fission; and $\nu = 2.9$ n/fission. For a beam energy $E_B = 800$ MeV/p, $\beta = 1.72$.

(d) Base on a beam energy $E_B = 800$ MeV/p.

(e) Appendix F

(f) Target thimble, fuel salt, graphite moderator, graphite reflector, control/shutdown rods, structure, reactor vessel.

(g) A single 285-MWe turbine would be used for each Target-blanket module; four of these less the recirculating power would provide $P_E = 967$ MWe to the grid; most of the BOP sizing assumed the use of a single turbine with gross capacity equal to 315 MWe.

Appendix A. Subsystem Design Bases and Equipment Scaling

A.1. Introduction

This appendix documents all calculational and design bases used to define a molten-salt-fueled ABC. Additionally, all key assumptions and groundrules are summarized. Lastly, design details and procedures not reported in the body of the report are elaborated in this appendix. In terms of developing an indepth understanding for use in a future, more-detailed conceptual design of ABC, this appendix serves both as a focal point and a resource. Generally, the ABC Plant Layout Study enphasizes the "reactor" aspects and for this reason draws heavily on earlier work performed at ORNL as part of the Molten Salt Breeder Reactor (MSBR) Program[10,11,46,47]. The Accelerator Equipment design is elaborated only to an extent needed to fulfill the goals of a plant layout study and the input such a study has to the development of an overall R&D plan for ABC[9].

The main body of this appendix consists of descriptions of the important equipment and piping systems for ABC. For each piece of equipment, the basis for the design is given along with important assumptions and caveats. Any scaling necessary for adaptation to the ABC is quantitatively described. The level of detail provided is not uniform for all systems, however; some systems are described in great detail while others are not. The determinants for this variability is not the specific importance of a given subsystem as much as the availability of information. Most of the information for this ABC design is taken from the MSBR design, as is described in Ref. 10. Areas in which detail was not available from the MSBR design are not described in great detail. The only exception to this is the Accelerator Equipment and Target system, which have be subject to only preconceptual designs and are described only superfically in this appendix.

This appendix arranged into five major sections. The history of the molten-salt reactor concept is given under Background in Sec. A.2. This background is followed by a discussion in Sec. A.3. of key assumptions used in the development of this concept. The Individual System Descriptions Sec. A.4. is divided according to eight subsystems: Accelerator; Target; Primary System; Balance of Plant; Chemical Processing; Operations and Maintenance (O&M); Instrumentation and Control (I&C); and Safety Systems. Flow calculations and the resulting mass flow rates and velocities required to size key subsystems are described in Sec. A.5. Finally, outstanding technical issues are described in Sec. A.6., which is divided into materials, design, and miscellaneous categories. A synposis of these issues is given in the Sec. I., Executive Summary.

A.2. Background

The goal of this design effort and the associated research is to combine the features of a molten-salt breeder reactor with those of an accelerator. The resulting ABC system is to fission surplus weapon plutonium to high burnup while minimizing the production of byproduct wastes. The molten-salt reactor concept was studied extensively from the 1950s to the 1970s at ORNL[10,11,45,46]. This effort was originally intended to produce a nuclear power plant for aircraft propulsion, but was later redirected toward the development of a thermal breeder based on the ^{232}Th-^{233}U fuel cycle. Although a full-scale breeder was never built, two smaller experimental reactors were constructed and operated.[11]

The first reactor was the Aircraft Reactor Experiment (ARE).[48] The ARE was a simple arrangement with a primary goal being a feasibility demonstration. Operated for approximately 221 hours in November, 1954, the favorable results led to the construction and operation of the Molten Salt Reactor Experiment (MSRE). The MSRE was larger and was designed for extended operation. It operated at power levels up to 7.4 MW from 1965-1969.[46] The MSBR design[10,11] was based to a large extent on the design and operation of the MSRE and on the subsequent development work that was carried out through the mid-1970s.

The accelerator design is taken from the Accelerator Production of Tritium (APT) design.[49] Both the ABC and the APT use linear accelerators (Linacs) to accelerate protons to high energy. The accelerator to be used with the ABC system delivers less beam power than the APT accelerator (~70 MW *versus* 200 MW). Experience exists with accelerators of this type[16,17,50] as experimental machines, but not as part of a high-power, high-availability production facility. Scaleup and design improvements to increase accelerator availability will be necessary for the implementation of the ABC system.

While the design described in the ABC Plant Layout Study is intended to be both self-consistent and conceptually feasible, this design is far from optimized. Conservatism included in the design should increase the probability of a successful implementation of the ABC approach to plutonium disposition. Scaling and extrapolation was necessary, however, to obtain capacities and dimensions of the major equipment, as applied to ABC conditions. The resources available for performing this work were not sufficient to allow original design work to proceed. The existence of the significant knowledge base developed as part of the MSBR program, as well as the documentation and maintenance of this knowledge base over the intervening years, was of immense benefit to the ABC Plant Layout Study.

A.3. Assumptions

Inherent in any pre-conceptual design are numerous and essential assumptions and groundrules. This section identifies these assumptions and groundrules. When appropriate, the basis for each is given.

The primary groundrule for the ABC Plant Layout Study is established by the intended project goal; to dispose of $M_{Pu} \simeq 50$ tonne of weapons plutonium. Assuming a twenty year development and construction period, this goal allows $T_{LIF} = 30$ yr for plutonium destruction. A more conservative approach has been adopted, however, wherein plutonium destruction is to be performed over a twenty-year period to allow for additional time for development and/or deployment, albeit, higher capacity (rate) systems will be required. By specifying M_{Pu} and T_{LIF}, the thermal-power requirement results. Assuming each ^{239}Pu fission yields on average $E_F = 200$ MeV, the total thermal power produced by the fissioning of fifty tonnes of ^{239}Pu is 4.04×10^{18} J (128 GWt yr). Assuming a twenty year burn time, with an average lifetime capacity factor of $p_f = 0.75$, the thermal capacity required for disposing of the fifty metric tons of plutonium is 8,530 MWt. For $N_{ABC} = 3$ ABC units having N_{BLK} Target-Blanket/Power-Conversion modules per ABC accelerator unit, each module will develop and convert a thermal power of $8{,}530/N_{ABC}/N_{BLK} = 711$ MW. This power is consistent with the restrictions imposed by the Target power density and neutron-generation efficiency (Appendix C). Generally the Core size (thermal power) is limited by the desire to maintain the neutronic worth of the target at a certain level. The maximum Core capacity has been estimated to be as high as P_{TH} = 2,000 MWt, but this limit is not well documented. Most of the target designs performed to date have limited the Core to 600 MWt based on safety (afterheat) consideration. Although economic benefits would be expected to accompany an increase in module size (and concomitant decrease in the number of modules), these benefits have not been quantified for the ABC and, therefore, have not been shown to counterbalance the operational flexibility provided by the smaller modules. For an accelerator-driven power plant based on the ^{232}Th-^{233}U fuel cycle, however, significant economic benefits accrue from minimizing N_{BLK} and maximizing P_{TH}[15].

The net plant thermal efficiency is taken to be the same as that of the MSBR (η_{TH} = 0.444), because of similarities in the two systems. This conversion efficiency yields a gross electrical output of P_{ET}/N_{BLK} = 316 MWe for each Target-Blanket module and a combined total of P_{ET} = 1,263 MWe for each of the three ABC systems. For the remainder of this appendix and most of the report, a "module" is defined as a 711 MWt/316 MWe combination of Target, Blanket, and Power-Conversion equipment. A "system" refers to the combination of N_{BLK} = 4 modules along with a single, supporting linear accelerator. The net thermal efficiency quoted above for the MSBR design includes the effects of plant load, both electrical and mechanical, for the reactor and associated systems. The accelerator and the associated power requirements represents a new element in the overall plant power balance that must be accounted before the net power delivered to the grid, $P_E = P_{ET}(1 - eps)$, can be estimated, where $\varepsilon = \varepsilon_{ACC} + \varepsilon_{AUX}$, ε_{AUX} is the fraction of P_{ET} needed to meet auxiliary power demands associated with the Accelerator Equipment plant, and ε_{ACC} is the fraction of P_{ET} recirculated to the Accelerator Equipment to create the energetic proton beam. Conservatively taking ε_{AUX} = 0.02 and $\varepsilon_{ACC} \simeq 0.12$, which corresponds to an accelerator "wall-plug" power of P_{EA} = 152 MW, a nominal power delivered for sale to the electrical grid from each of the N_{ABC} = 3 ABC units would be P_E = 1,074 MWe. For an accelerator "wall-plug" efficiency of $\eta_A \sim 0.45$ (Appendix B), the beam power per module would be $P_{EA}/\eta_A/N_{BLK}$ = 17 MW. These sample parameters are summarized on Table III.

Another important groundrule adopted for the ABC Plant Layout Study deals the minimization of the technical extrapolation from MSBR to ABC. Departures from the Ref.-10 MSBR design are proposed only to accommodate the accelerator beam and the target, to upgrade the design for modern safety requirements, and to incorporate molten-salt experience gained after the completion of the reference MSBR design[10]. Although the overall layout and equipment details have changed, the ABC design retains the essential elements of the MSBR design, thereby providing a firm foundation on which to develop, build, and (ultimately) operate the ABC.

Many other assumptions were evoked in the development of the ABC plant layout. For the most part, these assumptions are associated with individual systems or components and, therefore, are discussed in the relevant component descriptions given in subsequent sections of this appendix.

A.4. Individual System Descriptions

The ABC design has been divided into eight systems, as is indicated on Table II: Accelerator Equipment (AE);Target System (TAR); Primary System (PHT, for Primary Heat Transport, including the Blanket or Core); Chemical-Processing Equipment (CPE); Balance of Plant (BOP); Instrumentation and Control (I&C); Operations and Maintenance (O&M); and Safety Systems. The Accelerator Equipment includes all the equipment associated with producing the beam of high-energy (E_B = 800-1,000 MeV) protons, transporting and splitting the high-energy beam into beamlets, bending the (horizontal) beamlet to accommodate the vertically oriented target, and expanding (defocusing) the beamlet onto the target. The Target system includes the window, the self-cooled liquid-lead target *per se*, the lead (secondary) cooling system, and the lead cleanup system. The Primary System includes all those components that contain an appreciable quantity of plutonium-bearing fuel salt, with the exception of the drain tank and the chemical-processing tanks, which are listed under Chemical Processing. While other systems breakdowns of accelerator-driven nuclear systems treat the Blanket as a separate entity[14,15], for the purposes of the ABC Plant Layout Study the Blanket (or Core) is incorporated into the broader Primary System. The Balance of Plant includes the Secondary-Coolant-Salt system and all the steam-power-conversion equipment. The Chemical-Processing system includes the equipment for initial salt purification, fission-product removal, moisture removal, impurity removal, and equipment needed to meet all other salt cleanup requirements. Instrumentation and Controls includes the equipment needed for operation, surveillance, and protection of the plant. The Operations and Maintenance system includes equipment for such activities as refueling, replacing the graphite neutron moderator, and performing other remote maintenance tasks both within and outside the Primary System. Finally, Safety includes those systems intended to protect plant investment or public safety, including the main Containment Building and the various operating cells housed within this structure.

A.4.1. Accelerator

The accelerator design is scaled from the Accelerator Production of Tritium (APT) Program[49]. A linear accelerator is used to provide an 800-MeV proton beam with a current of 100-200 mA. This proton beam is divided into four beamlets and delivered to four individual target/blanket modules. Bending magnets are then used to direct the beamlet towards the target and to spread (defocus) the beam prior to impinging onto the target window with an acceptable power density. The beamlet then impacts a liquid lead target and in the course of slowing down interacts with lead nucleons to produce the large flux of primary neutrons necessary to drive the ABC concept. The important accelerator components from the viewpoint of the Target-Blanket design are the beam tube itself, the beam splitter, the bending magnets, the beam expander, and the beam-tube isolation valves (BTIVs). The beam splitter periodically "kicks" out a portion of the main beam and directs it to the module, resulting in a pulsed beam incident on each target assembly. The period between pulses is (????MHz) compared to the thermal-response timescales of importance to the Target-Blanket design, and the beam is treated as continuous.

The beam travels through an evacuated , ~100-mm-diameter beam tube, which is sized to reduce beam-wall interactions ($<10^{-6}$/m) from the beam halo and to reduce activation of the beam tube. The split proton beam is directed towards the target assembly by means of one or two 90° bends. Each bend can be accomplished through the use of bending magnets that surround the beam tube, as is discussed quantitatively in Appendix F. The unexpanded beam (at most a few centimeters in diameter) enters the Containment Building horizontally from the side and subsequently enters the bending region, which consists of a series of copper quadupole electromagnets. The external diameter of this bending section is ~ 1 m, with an additional ~0.5 m steel radial shielding. The total diameter, therefore, is ~2.0 m in this region. The bending radius, based on the optimization described in Appendix F, is set to 3.0 m.

After being redirected vertically towards the target, the beam must be expanded prior to impacting the window and lead target. Beam expansion is accomplished under the natural effects of the space charge in this unconfined section, although a magnet may be necessary at the top of the expansion region. An expansion distance of ~10 m is necessary to spread the beam sufficiently. If magnets are not needed in the lower portion of the expansion region, the expansion distance may coincide with the shielding space (4-5 m) above the Core. Until a better understanding of the expansion process is available[21,22], it is assumed that a 10-m expansion length above the maintenance floor is sufficient. The entire expansion distance will be shielded with steel of a thickness assumed needed for the the beam-bending region (2.0-m outside diameter).

As currently envisioned, the ~1,000-m-long accelerator will be placed 10-15 m underground. The modules served by the accelerator will be aligned in a row parallel to the accelerator axis, with beamlets being split along the length of the High-Energy Beam Transport (HEBT) system (Fig. 6). The split beam travels horizontally and perpendicular to the accelerator axis. The beamlet will be conducted horizontally through the side of the containment building wall by means of a BTIV. As described above, a set of bending magnets will redirect the beam vertically downward in alignment with the vertically oriented Target-Blanket assembly. The beam tube will enter the Containment Building through the side, and will be bent downward towards the Target-Blanket assembly from the top. Isolation valves will be used to isolate the Containment Building in event of an accident in the Target-Blanket assembly. These valves have yet to be designed, although design requirements have been developed. These valves must hold containment design pressure (0.4 MPa, 60 psia) on one side with vacuum on the other. These valves cannot withstand beam impact, and it is assumed that either an accelerator trip signal or a beam-splitter trip signal will be generated as part of the containment isolation response signal.

A.4.2. Target

Several target systems have been evaluated, as is described in Appendix C. A self-cooled liquid-lead system was determined to have the optimal combination of high neutron production, low parasitic neutron capture, and good heat-removal capability. The primary difficulty associated with use of lead is associated with the containment material. The lead system must be operated at temperatures near those of the molten-salt system (????K) to limit parasitic heat loss to the lead, unless a radiation-resistant thermal insulation can be found. The target design described in the following section is based on the Los Alamos experience (Appendix C), with a number of modifications made to enhance the lifetime of the structures.

The cool lead (???K) leaves an electromagnetic(???) pump located in the reactor-vessel cell and flows to the top of the reactor vessel. At this point the lead enters an annulus formed by concentric rings of modified 9Cr-1Mo ferritic steel. The cool lead flows down into the reactor vessel and is redirected 180° for entry into the target region from the bottom, where flow continues upward towards the window. The lead impacts the Nb-1Zr target window directly and provides the only cooling for the window. The lead then flows up through another annulus, that is also formed by concentric 9Cr-1Mo steel pipes. The hot lead annulus (???K) is concentric with both the cool lead annulus and the accelerator beam tube and is located between the two. The hot lead flows upward and out of the reactor vessel to a shell-and-tube heat exchanger, where heat is given to a HITEC secondary coolant. The lead then flows back into the electromagnetic (???) pump.

The 9Cr-1Mo steel used for lead containment has adequate corrosion resistance at low temperatures. The lead temperature has been limited to 780 K(950° F) to minimize corrosion. An optimization is needed to balance the corrosion requirements with the parasitic heat loss from the fuel salt to the lead. Further reduction in the lead temperature may be possible, if necessary. The Nb-1Zr target window is the only lead-containing component not constructed from 9Cr-1Mo steel. The Nb-1Zr alloy maintains strength at high temperatures, but it is susceptible to oxidation, even in moderate (10^{-6} torr) vacuum. Oxygen may also present a problem on the lead side, as the Nb will preferentially react with any oxygen in the lead. More detailed calculations are necessary to determine whether the present lead-cooling configuration will be sufficient to for a workable window lifetime under normal as well as anticipated transient conditions.

The lead target is expected to receive ~11(???) MWt from the 17-MW beam. Because of the temperature difference between the lead system and the molten-salt system, a parasitic heat loss from the salt into the lead will amount to an additional $\lesssim$ 0.2 MW. Since this estimate is based on a simplified model, the heat load is assumed to be 0.5 MW. The lead-system heat load, therefore, is 11.5 MW; the piping, pump, and heat exchanger are designed to remove this quantity of heat.

No information is available on the lead-to-salt heat exchanger. However, the design requirements for this heat exchanger have been developed(???). The design is envisioned as a standard shell-and-tube heat exchanger with lead flow in the tubes. Compatibility problems are not envisaged for the lead-HITEC-steel system. One concern that was raised for use of HITEC in the secondary system is related to the maximum operating temperature. At temperatures of ~870 K (1,100° F) HITEC begins to decompose. At the lower temperatures proposed for the APT application, however, decomposition is not expected to be a concern. The HITEC lower temperature is only limited by freezing of lead in the target heat exchanger.

The lead system will use an electromagnetic(???) pump located on the vertical section of pipe between the bottom of the heat exchanger and the top of the reactor vessel. This electromagnetic pump will probably require active cooling of the magnets that must operate in the high-temperature environment of the reactor-vessel cell. Helium or nitrogen coolants are considered for this application. Helium is available as part of the off-gas system, and nitrogen is available from the atmosphere-control system for the reactor-vessel cell. In either case, coolant gas will be used to maintain the motor magnets below the cell operating temperature. The entire lead system may require insulation to limit the heat losses from the reactor-vessel cell atmosphere into the lead.

The lead is expected to accumulate spallation and neutron activation products. The effects of these impurities on the 9Cr-1Mo alloy corrosion are unknown. Nevertheless, a lead cleanup system is included in the design to remove these impurities as they are formed. The lead cleanup system will process the lead in a batch operation. A portion of the inventory will be transferred to the cleanup system and processed while the Target-Blanket system continues to operate. The processing rate has not been determined, but is not expected to be high; for the conditions listed in Table II, the lead is consumed at a rate of ???? mole/yr. A characteristic removal time of several days the processing time is expected to be a few days.

A.4.3. Primary System

The Primary System is defined for the purposes of the ABC Plant Layout Study to include those components that contain an appreciable quantity of fuel salt. The only exceptions are the drain and storage tanks, that are considered part of the Chemical-Processing System. The Primary System consists of the Core and any surrounding graphite moderator/reflector, the Hastelloy-N reactor vessel, the fuel-salt pumps, the intermediate heat exchangers (IHX), all and the associated piping. The reactor vessel is the reference point in the primary system, and all the other components move out radially as the system heats up. The total fuel salt volume has been estimated to be less than 16 m^3, with 4.5 m^3 (28.1%) of the fuel salt residing within the core at any one time. The effective neutron flux, therefore, is reduced by this factor of 0.28.

A.4.3.1. Core

The power production and concomitant plutonium burning will take place in the Core. At the center of the Core is the liquid-lead target. Surrounding this target is a region of graphite blocks through which the fuel-salt flows. This design is similar to the core design of the MSBR. Individual graphite assemblies are arranged to form flow channels between adjacent assemblies as well as through the center of each assembly. Each prismatic assembly is 0.1 m on a side. The assemblies are arranged into a square cylinder (*i.e.*, diameter equals height). The dimensions for the Core were derived using the following baseline assumptions. The Core power density and fuel-salt volume were assumed to be those used in the MSBR: $P_{TH}/V_{BLK}/N_{BLK} = 22.2$ MW/m^3 and $f_{FS} = 0.13$, respectively. The thermal power per module is $P_{TH}/N_{BLK} = 711$ MW.

The diameter of the Core is $D_{COR} = 3.45$ m on the basis of this average power density and computing a volume that eliminates the target volume and assuming the target top is located at the core centerline, thereby taking up the maximum Core volume. The Core diameter has been conservative taken to be 3.5 m, which provides sufficient volume to generate 711 MWt, assuming an average power density of 22.2 MW/m^3. Given these Core dimensions and the fuel-salt volume fraction, the fuel-salt volume in the Core is 4.3 m^3.

The MSBR design included both axial and radial reflectors. The reflector thickness was 0.76 m. A similarly sized relector is used in the ABC design. which is taken to be 0.75 m; a reactor-vessel inside diameter of 5.0 m results. The reactor-vessel inside height is also 5.0 m.

Shortened graphite blocks will be placed under the target assembly. The flow through and around these blocks will be used in part to cool the external Hastelloy-N thimble in which the target assembly is housed as a mechanically separate entity. Design of this region of the Core needs to balance the cooling requirements of the Hastelloy-N thimble with the parasitic heat loss into the lead system. Because the lead system must be operated at a lower temperature than the fuel-salt system, a certain amount of parasitic heat loss is expected. The heat loss into the lead may be sufficient to cool the Hastelloy-N thimble. If so, an arrangement with minimal fuel-salt flow past the thimble should result in the smallest parasitic loss; more detailed thermal analysis is needed in this area. If the parasitic heat losses are too large, the size of the lead system will increase. Preliminary estimates indicate, however, that parasitic heat loss at less than 0.5 MW.

The mixed-mean fuel-salt outlet temperature has been set at 980 K (1300° F) to match that of the MSBR design (???). It may be possible to achieve this bulk fuel-salt outlet temperature by mixing hotter salt from the flow channels with cooler salt from around the target thimble. The size of the flow channels around the Hastelloy-N thimble can be adjusted to increase the fuel-salt flow rate, thereby lowering the bulk temperature of the fuel salt. By lowering the temperature of the salt surrounding the thimble, parasitic heat losses into the lead system will be reduced. A number of unanswered questions about this potential Core arrangement have been identified. Two materials concerns are foremost. First is related to the ability of the graphite in the upper plenum to withstand the thermal cycling that is driven by mixing of salt from the hottest channels and from the cool bypass channels. The second question deals with the ability of the Hastelloy-N thimble in the region where it penetrates the upper plenum to withstand this thermal cycling. A related concern is the level of heat generation in the salt and in the Hastelloy-N thimble in this region.

The individual graphite stringers in the Core in many respects resemble conventional fuel assemblies and will be handled similarly. Failed stringers, or those that have reached a limiting radiation exposure, will be replaced during maintenance outages as individual units. The MSBR design considered two maintenance options for the graphite moderator. The reference design used a monolithic graphite assembly that was to be replaced as a unit. The backup option was based on the removal of individual stringers, which gave added operational flexibility by allowing replacement of single failed stringers as necessary. The preferred option will depend on the complexities accompanying reactor-vessel opening, the life time of the graphite, and the ability to operate with failed stringers.

A.4.3.2. Fuel-Salt Pump

The fuel salt enters an outlet plenum at the top of the Core. Two channels in the graphite reflector carry the fuel-salt out of the reactor vessel. The fuel salt flows out the side of the blanket vessel and into the fuel-salt (suction) pump inlet located at the bottom of the pump (Fig. 9). This the sump-type pump will require scale-up from pumps that have actually been tested by approximatedly a factor of ~x????, based on flow rate. Without a knowledged of the required scaling relationship, the dimensions of the ABC fuel-salt pump are taken directly from the MSBR design. The MSBR design included two identical pumps (fuel salt and coolant salt) that differed only in output with the same external dimensions. Use of these dimensions should be conservatively large (by a factor of ~x???) for the ABC design.

The pump bowl (Fig. 9) is taken to have a diameter of 2.0 m and a height of 1.5 m. A 4-m separation between the motor and the impeller is included to provide adequate shielding for the motor. The design of the pump is complicated by the required long drive shaft, which must be sealed to prevent escape of fission gases and ingress of lubricating fluids. Additionally, differential thermal expansion must be accommodated. Couplings at the top and bottom of the shaft allow the impeller end of the pump to move horizontally in relation to the motor as the Primary System piping expands upon heating.

The placement of the fuel-salt pump within the reactor-vessel cell is constrained by the free surface of the fuel salt in the pump bowl (Fig. 9). The pump bowl functions as the pressurizer in a PWR. Although the surge volume is not large, the bowl is directly connected to the fuel-salt drain tank via an overflow line (Fig. 3). The fuel-salt pump must be located with the bowl at the highest point in the fuel-salt system. The use of the pump bowl as a surge volume results in a large pump bowl. The fuel-salt height is, however, more important than the total volume (where???, clarify!!). If the fuel-salt volume held up in the pump bowl is determined to be excessive, graphite filler blocks can be added to the pump bowl. Another option is to design the pump bowl to conform better to the impeller while using the same overall expansion height. The fuel salt is assumed in the present configuration to be 1-m deep in the pump bowl, with about 30% of the pump bowl volume being occupied by fuel salt. The remaining volume is taken up by graphite blocks, the impeller, and other equipment. The fuel-salt volume in each pump bowl is approximated by $\pi(D/2)^2H$ =~1.0 m^3 for D = 2 m and H = 1 m.

A.4.3.3. Intermediate Heat Exchanger (IHX)

The IHX design for ABC closely followed that of the MSBR design (Fig. 10), with only minor changes and some scaling in size to accommodate the ABC. The IHX used in the MSBR design has a central downcomer through which the inlet (secondary) coolant salt flows. At the bottom of the IHX, the secondary coolant salt turns and flows upward along the tube array countercurrently to the fuel salt. The coolant salt is collected at the top of the IHX and exits through an annular pipe that is concentric with the inlet pipe. The fuel salt enters through the side of the shell at the top of the IHX through the inverted "L"-shaped tubes and through a vertical tubesheet, and then turns and flows down to the lower horizontal tubesheet. The collected fuel salt then exits the IHX through a central outlet pipe located at the bottom.

Only minor modifications to the MSBR design were made for application to ABC. These modifications are intended to optimize the layout and to scale the IHX to the ABC thermal power. Four IHXs were used in MSBR for a total thermal output of 2,250 MW; each IHX, therefore, had a capacity of P_{IHX} = 563 MW. Two IHXs were adopted for the ABC, each with a capacity of 711/2 = 356 MW. The ABC IHXs are ~ 30% smaller than the MSBR IHXs. This size reduction can be accomplished either through a reduction in the shell diameter or height. For the purposes of the ABC layout, the height must be roughly consistent with the reactor-vessel height. Because of the free molten-salt surface maintained in the fuel-salt pump bowl, the upper IHX tubesheet must be at or below the height of the pump impeller. On the basis of these constraints, the tube length has been set at 5.25 m. This height places the center of the upper tubesheet at the same elevation as the pump impeller, and the lowerest part of the IHX is at the same elevation as the bottom of the reactor vessel (Fig. 13).

The distance determined above is actually termed the "tube-sheet-to-tube-sheet" distance in the MSBR design. The actual tube length is somewhat longer because of the bends that are fabricated into the upper part of the tubes. This additional length is ignored, however, thereby adding conservatism to the design. Once the tube length has been determined, the shell diameter can be scaled from the MSBR design. The MSBR IHX had a shell inner diameter of D_{IHX} = 1.72 m, which gaves a tube length of L_{TUB} = 7.44 m. The diameter required for the ABC IHX design is scaled as $\sqrt{P_{IHX} L_{TUB}}$, which gives D_{IHX} = 1.63 m. This diameter is close to that of the MSBR IHX. To avoid changes in the internal dimensions and layout of the IHX, the cross section for the ABC IHX is taken directly from the MSBR design, and only the tube length is changed. This assumption gives additional conservatism into the design, because the actual capacity of the ABC IHX 110% of that required.

The final dimensions of the IHX, therefore, are as follows: shell inner diameter is 1.72 m; wall thickness of 16 mm; total height is 6.55 m, which was determined from the set tube length and from allowing for upper and lower plena spacing.

Other than shortening the tube length, the only other modification to the IHX design was the relocation of the (secondary) coolant-salt outlet pipe. In the MSBR design, the outlet pipe was concentric with the inlet pipe. The complexity of this piping configuration was eliminated in the ABC design in favor of using a more traditional exit through the side of the shell. This configuration lengthens the shell somewhat, but a simplification of the coolant-salt piping layout results. This layout has several advantages for the salt system when compared to a more traditional shell-and-tube design based on horizontal tubesheets and shell coolant penetrations in the side. After grinding down a seal weld, the tube bundle can be removed remotely from the top. Improved remote maintenance is an important element in the the design process adopted by the ABC Plant Layout Study. One of the primary disadvantages of this particular design, however, is the absence of coolant-salt drainage capability; the coolant salt must be removed using gas pressurization.

For the most part, the IHX internal components remain unchanged from the MSBR design. A 0.5-m-(outside)diameter central downcomer carries fuel salt from the upper coolant salt entrance to the lower tubesheet. This downcomer is surrounded by an array of N_{TUB} = 5,803 tubes carrying the fuel salt. Each tube has an outer diameter of D_{TUB} = 9.5 mm and a wall thickness of δ_{TUB} = 0.90 mm. The fuel-salt tubes are arranged with a constant radial and circumferential pitch of 19.1 mm. Each is shaped in the form of an inverted "L" and is knurled in a spiral pattern throughout the baffled region of the heat exchanger to increase the heat-transfer film coefficient on the coolant-salt side(???). A total of fifteen disc and torus-shaped baffles will be placed in the lower section of the IHX, with a baffle spacing of 0.3 meters. The knurling and the baffling are included to enhance the overall heat-transfer coefficient. The uppermost section of tubes contains no baffles. In this region, the tubes are bent into a sinusoidal configuration to absorb the thermal stresses.

The IHX was designed to minimize the fuel-salt volume contained therein; the IHX tubes contain $N_{TUB}\, \pi(D_{TUB} - 2\delta_{TUB})^2 L_{TUB}$ = 1.5 m^3 of fuel salt. The total fuel-salt volume in the IHX also includes the volume of the inlet and outlet plena; this volume is assumed to be less than 0.5 m^3, and the total fuel-salt volume becomes 2.0 m^3/IHX.

A.4.3.4. Fuel-Salt Piping

The fuel-salt piping connects the Core with the fuel-salt pump and the IHX. Like the reactor vessel, this piping will be fabricated from modified Hastelloy-N. This piping does not contain high pressure; the Primary System will have a maximum operating pressure less than 1.4 MPa. The piping sizes were determined from scaling MSBR piping sizes to maintain the flow velocities reasonably constant; a nominal flow velocity for both MSBR and ABC in the fuel-salt piping systems is ???? m/s, compared to ?????m/s in the Core *per se*.

The MSBR Primary System piping ranged in diameter from 0.40 m up to a maximum of 0.50 m. Piping size was limited to constrain the exo-Core fuel-salt inventory. The piping from the Core to the fuel-salt pump is the largest. To maintain the flow velocities in the ABC system close to those of the MSBR (???? m/s), 0.40-m outside-diameter piping is used for the entire fuel-salt system. The piping sizes do not scale directly with the power level for the fuel-salt system because the MSBR fuel included up to 10% thorium (???kg/m^3), which increases the density to 150% of that of pure LiF-BeF$_2$. The physical properties of the ABC fuel salt, which is very dilute (???kg/m^3) in plutonium, are similar to those of pure LiF-BeF$_2$. For piping of this size, the minimum bend radius for a 90° elbow is 0.6 m. Large radii-of-curvatures are used throughout the ABC plant layout to alleviate thermal stresses.

The piping sizes and placements for ABC were chosen to limit the fuel-salt volume contained in the pipes, while balancing this goal with the need for adequate clearance between components and acceptable flow velocities (*e.g.*, < ???m/s). The total length of primary piping is 9.75 m per fuel-salt loop, which results in a fuel salt volume of 1.1 m^3/loop for a nominal pipe inside diameter of 0.38 m (10-mm wall thickness). The total fuel-salt volume is found from adding the volume contained in the Core (4.3 m^3), the piping (1.0 m^3/loop), the fuel-salt pump (2.0 m^3/pump), and the IHX (1.1 m^3/IHX). The total volume for two loops is 12.5 m^3; a Core fraction of 4.3/12.5 = 0.34 results.

A.4.4. Balance of Plant

The Balance-of-Plant includes the secondary-coolant-salt system and all the Power-Conversion equipment. The secondary-coolant-salt system includes the shell side of both IHXs, the two coolant-salt pumps, the steam generator, the steam reheater, and the connecting piping. The Power-Conversion equipment includes the steam generator, the steam reheater, the turbines, the condensers, the feedwater-heating equipment, and all the associated piping. The BOP design is complicated by the use of the supercritical-steam (SCS) cycle[10,25]. Although the SCS cycle results in higher net plant thermal efficiency , $\eta_{TH} = 0.444$, and in a simplified feedwater heater, the SCS cycle introduces a number of difficulties that are discussed in the following sections.

A.4.4.1. Coolant-Salt System

The ABC Target-Blanket system uses a sodium fluoride - sodium fluoroborate eutectic mixture for the secondary coolant. This salt was the reference coolant salt for the MSBR design, and is commonly referred to simply as "sodium fluoroborate" or even "fluoroborate"; it is assumed that reference is made to the eutectic that contains 8% NaF.

The coolant salt carries the fission heat from the two IHXs out of the reactor-vessel cell and into the steam-generator cell to the steam generator and the steam reheater. In these two components, the fluoroborate transfers heat to the steam and flows into the coolant-salt pumps and back into the reactor-vessel cell (Figs 3 and 4). The important components of the coolant-salt system are the shell sides of the IHXs, the steam generator, the steam reheater, and the coolant-salt pumps; each are described in the following subsections.

A.4.4.2. Steam Generator

The steam generator is located in the steam-generator cell. The coolant-salt flows from both IHXs are combined into a single pipe. This pipe carries the coolant salt from the reactor-vessel cell to the steam-generator cell. Inside the steam-generator cell, the flow splits again into two equal-size pipes, which bend 90° in opposite directions. One of these flows enters the steam generator, the design of which is taken with few modifications from the MSBR design. Only a single steam generator is used with each 711-MW Target-Blanket module. The steam generator does not take the full load, however, because the coolant-salt flow is split between the steam generator and the steam reheater (Fig. 3 and 14) in ratio ????/????.

The MSBR steam generator was a U-tube/U-shell heat exchanger with coolant salt flowing on the shell side and supercritical steam inside the tubes. The U-shell design was used to minimize the shell diameter and, therefore, the required wall thickness. The MSBR used a total of 16 (four per loop, 140 MW/SG???) of these small steam generators. The steam generators were oriented horizontally to assist in natural circulation (???how?) and in coolant-salt drainage. Assuming the steam-generator design is not changed appreciably, the total tube length per unit of thermal power should remain constant with size. This assumption ignores the difference in edge effects as the surface-to-volume ratio changes. Given that each 140-MW MSBR steam generator contained $N_{TUB} = 393$ tubes of length $L_{TUB} = 23.3$ m, the total length of SG tubes for ABC given by $N_{TUB}\, L_{TUB} = 46{,}271$ tube m. For a SG diameter of $D_{SG} = 1.5$ m (chosen somewhat arbitrarily because a reasonable overall steam-generator size results), the radius of curvature of the U-tube shell is $R_{SG} = 4/3\ D_{SG} = 2.0$ meters. The total tube-sheet-to-tube-sheet length for the average tube, therefore, is $L_{TUB} = 2L_{SG} + (4/3)\pi D_{SG}$ in this case.

Assuming the area required per tube is constant, which for the MSBR steam-generator design was 4.29×10^{-4} m^2/tube, the number of tubes for the ABC steam generator is found to be $N_{TUB} = \pi(D_{SG}/2)^2/4.29\times10^{-4} = 4{,}115$. With $N_{TUB}\, L_{TUB} = 46{,}271$, the length per tube is $L_{TUB} = 11.24$ m. Therefore, the length of the straight portion, L, is equal to 2.48 m, which is rounded off to L = 2.5 m. The overall length of the steam generator is given by $R_{SG} + D_{SG}/2 + L$ plus the lengths of the inlet/outlet plena, where $R_{SG} = 2.0$ m, $D_{SG} = 1.5$ m, and $L_{SG} = 2.5$ m. The overall SG length, therefore, is 5.25 m plus the length of the inlet/outlet plena. A ~10% contingency has been added to this length because to account for scaling uncertainties. Furthermore, the inlet/outlet plena are assumed to be spherical, with an outer diameter of 2.0 meters. The resulting total length for the steam generator is 7.3 m and the height is 6.0 m(?????).

The required wall thickness for each plenum of radius R containing a pressure p and operated at a design stress sigma is approximated by[50]

$$t = \frac{Rp}{2\sigma} \ . \qquad \text{(A-1)}$$

For p = 25.85 MPa and σ = 53 MPa (low???), the plenum wall thickness is t = 0.24 m for R = 1.0 m. Although this is essentially a pressure vessel, it appears to be more desirable to produce the two required plena for a single steam generator than to manufacture and operate multiple, smaller steam generators. However, the MSBR use of 16 small steam generators does not support this conclusion. This issue will be addressed further in the detailed design stage. (note!: this sort of stuff cannot be sluffed off to some future study, it should be resolved here...is the estimate correct????, i.e., can σ be increased, or a more accurate stress formula use???)

A.4.4.3. Steam Reheater

The MSBR steam system design and the Bull Run Steam Plant design from which it was taken[25] both follow standard practice of including steam reheat after expansion through the high-pressure turbine (Figs. 3 and 14). Steam reheating is complicated in the molten-salt system because of the minimum feedwater-temperature requirements. To avoid the molten-salt freezing in the reheater, the reheat steam must be raised in temperature from about 560 K(550° F) (turbine exhaust) to 618 K (650° F) in a preheater. The steam reheater (SR) takes the preheated high-pressure-turbine exhaust at 618 K (650° F) and uses coolant salt to heat this intermediate-pressure steam back to 811 K (1000° F).

The MSBR reheater design used a standard tube-and-shell arrangement. To facilitate the coolant-salt and steam piping in the ABC design, the steam-generator design was scaled down to model the ABC steam reheater. This U-tube/U-shell component is significantly smaller than the steam generator and must only withstand an operating pressure of 4.0 MPa (580 psi).

The sizing calculations for the reheater are similar to those for the steam generator. Assuming that the total tube length per unit power is the same in this U-tube/U-shell design as in the MSBR shell-and-tube design, the total required tube length is found for 400-tube, 9.23-m-long, 281.3 MW/SR MSBR design after scaling to the 711-MW ABC requirements to be $N_{TUB}\,L_{TUB}$ = 9,339 tube m. For similarity in layout, the distance between the coolant-salt inlet nozzle and the bent end of the shell was taken to be the same for the reheater as for the steam generator. This distance is 5.05 m. However, the additional distance required for clearance between the coolant-salt piping and the inlet/outlet plena are not equivalent. For the reheater, the overall distance between the tubesheet and the bent end of the shell is 5.39 m. The average tube length, therefore, is L_{TUB} = 12.44 m. The number of tubes required in this case is N_{TUB} = 9339/12.44 = 751. Assuming the area required per tube is the same in this U-tube/U-shell design as in the MSBR straight-tube design, which had a shell diameter of 0.54 m and 400 tubes, the shell diameter for the ABC case is $D_{SR} = 0.54\sqrt{751/400} = 0.74$ m. As for the steam generator, additional capacity is provided to accommodate uncertainties in the heat-exchanger effectiveness. Adding a 10% margin or contingency gives a reheater shell diameter of D_{SR} = 0.8 m.

The overall dimensions for the steam reheater are as follows. The tubesheet-to-tubesheet length is L_{TUB} = 12.44 m. The inlet/outlet plena have an outer diameter of 0.9 m, which results in a total reheater length of L_{RH} = 6.0 m and a total height of H_{SR} = 3.0 m. The N_{TUB} = 751 tubes are arranged with a triangular pitch of 25 mm.

A.4.4.4. Coolant-Salt Piping

Like the fuel-salt piping, the coolant-salt piping will be constructed of modified Hastelloy-N. The coolant salt is less expensive than the fuel salt, and does not affect the fissile inventory. Minimizing the coolant-salt inventory, therefore, is not as important as is minimizing the fuel- salt volume. The coolant salt is used in the shell side of the IHXs, the steam generator, and the steam reheater. The coolant-salt piping in the MSBR has a maximum size of 0.6 m between the IHX and the flow-partitioning header that split the flow into the steam-generators and steam-reheaters streams. Scaling this piping for the ABC design by the power rating, the largest coolant-salt piping is about 0.6-m in diameter. No bends are encountered in this large diameter piping.

The smaller coolant salt piping sizes for the ABC are taken to be 0.5-m in diameter. The sizes for the smaller coolant-salt piping runs in the MSBR are as small as 0.3-m in diameter. The MSBR design utilized four steam generators and two reheaters for each 562.5 MW coolant loop, whereas the ABC design uses a single steam generator and reheater for 711 MW. The piping sizes, therefore, are somewhat larger. The minimum bend radius for this 0.5-m-diameter pipe is 0.75 m.

Because molten-salt metering valves have not been demonstrated, splitting the flow between the steam generator and steam reheater is not straightforward. The heat requirements are not equal, with the SG/SR ratio being ????/????. Control is further complicated by the requirement that the reheat-steam outlet temperature must be adjusted by the coolant-salt flow rate instead of the feedwater flow rate. An innovative solution to this problem is included in the ABC design: two coolant-salt pumps are used; one each on the exit lines from the steam generator and steam reheater. The variable-speed pump motors will be used to adjust the flow through the steam generator and steam reheater. The ability of this system to handle transients has not been addressed, but responses are expected to be similar to that of an equivalent valve-based system.

As mentioned above, the coolant salt is inexpensive, relative to the fuel salt, and limiting its volume, therefore, is not as important as limiting the fuel salt volume. The coolant-salt volume is important primarily for sizing the coolant-salt drain tanks. The coolant-salt volumes have been estimated as follows: 23 m^3 in the steam generator; 5 m^3 in the reheater; 10 m^3 in the coolant-salt piping; and 3 m^3 in the two pump bowls. The total coolant salt volume, therefore, is 41 m^3.

A.4.4.5. Power-Conversion Equipment

The design of the Power-Conversion equipment is taken from the MSBR design, which was scaled from the Bull Run Steam Plant design[25] and is illustrated in Figs. 3 and 14[10]. The only changes to the design are related to feedwater heating, since the minimum feedwater temperatures of the flows to the salt-heated steam generator and steam reheater are considerably higher than those needed in a fossil-fired plant. (*e.g.* ????K *versus* ????K).

Steam leaves the steam generator at 810 K(1000° F) and 24.5 MPa (3600 psia) (????3800 psia????). The main steam flow is then divided, with 71% of the 400 kg/s main steam going to the high pressure turbine and the remainder going to the reheat-steam preheater and ultimately to the final stage of feedwater heating. The 285 kg/s flowing to the high-pressure turbine enters the turbine throttle at 810 K (1000° F) and 23.8 MPa (3500 psia). Steam is extracted from the high-pressure turbine to drive the steam-turbine-driven main-boiler feedwater pumps and for regenerative feedwater heating. The remaining steam, 204 kg/s, leaves the high-pressure turbine at 4.1 MPa (600 psia) and 556 K (550° F). This steam is heated in the reheat-steam preheater to 617 K (650° F) before entering the reheater. The reheat steam exits the reheater at 3.7 MPa (540 psia) and 810 K (1000° F) and passes to the intermediate-pressure turbine, and then to the low-pressure turbine. The condensate from the condensers passes through full-flow demineralizers, which are necessary to maintain heat-exchanger surfaces in both the steam generator and reheater. Eight stages of regenerative feedwater heating, plus a final mixing with the steam leaving the reheat steam preheater, are included to obtain the high feedwater temperatures necessary. Most of the pressurization is accomplished through the steam-turbine-driven main-boiler feedwater pumps, with the final increase provided by electrical booster pumps added immediately upstream of the steam generator. The overall power-conversion system is complex but conventional and proven,[25] with the exception of the high feedwater-temperature requirements and the use of coolant-salt in the steam generator and reheater.

Alternate power-conversion systems were investigated as part of the MSBR design process. Because of the feedwater requirements and the high temperatures obtainable with the molten-salt system, this supercritical-steam design was determined to have the best characteristics[10,11]. Additional alternatives for the power-conversion system will be evaluated as part of the detailed design stage of ABC, however.

A.4.4.6. Steam System Piping

The steam system piping for the ABC system must withstand maximum steam conditions of nearly 25.9 MPa (3800 psia) and 810 K (1000° F). These severe service conditions limit the size of piping that can be used. The MSBR steam system was adapted from the Bull Run Steam Plant design[25], and little detail on this adapted system is available beyond the design of the the main heat exchangers (*i.e.*, preheater, reheater, and steam generator). The available information indicates that the MSBR design used the same size main steam lines as the Bull Run plant: 0.22 m inside diameter and 65-mm wall thickness. The mass flow rate (??? kg/m^2/s or m/s, but not kg/s) in the MSBR steam lines was much lower than that in the Bull Run design (????m/s for MSBR versus ????v/s for Bull Run), while the steam conditions were nearly identical. Using the mass flow rates from the MSBR design, five main steam lines and five SCS feedwater lines would be required for the single ABC steam generator. Using the more aggressive Bull Run design, a single main steam line and a single feedwater line, measuring 0.25-m ID and having a 76-mm wall thickness, is adequate.

Assuming the ABC steam system is similar in design to Bull Run such that the required main steam line flow area is only a function of unit power, where each of the four Bull Run steam loops (0.22-m ID) handles a thermal power of 950(MWe)/0.44(MWe/MWt)/4 = 539.8 MWt, the required main steam line diameter (assuming only a single line) is $D_{SL} = 0.22\sqrt{711/540} = 0.25$ m. Taking the inner diameter to be 0.25 m, the required wall thickness is estimated using the properties of 2 1/4 Cr–1 Mo ferritic steel and the following expression[50](??):

$$t_m = p\,\frac{(D_{SL} + 2t_m)}{2(\sigma E + p\,y)} = 0.182(D_{SL} + 2t_m) \quad , \qquad \text{(A-2)}$$

(???? what is the origin and basis of this expression???)

where D_{SL} is the inside diameter of of the steam line (SL), p is the internal pressure, σ is the design stress, E is ??????, and y is ????????. For a 0.25-m ID pipe, the minimum wall thickness is 73.7 mm. The main steam and feedwater lines for the ABC system, therefore, are taken to be 0.25-m ID with a 76-mm wall thickness. The maximum wall stress for this piping is determined using the following expression for the stress in a thick-wall tube[50] with po = 0 (????):

$$\sigma_t = p\frac{r_o^2 + r_i^2}{r_o^2 - r_i^2} , \qquad \text{(A-3)}$$

where r_o and r_i are the outside and inside radii of a long tube under pressure p. The stress predicted from Eq. (A-3) is slightly higher than the maximum allowable stress of 53 MPa (7,800 psi), (??? seems low???) but the predicted stress (???) is nearly equivalent to the wall stress for the Bull Run main steam lines [> 59 MPa (8,700 psi)].

Each of the steam and feedwater lines penetrating containment is equipped with a pair of main steam isolation valves (MSIVs). These valves are normally held open by air pressure and are designed to fail in the closed position in the event of a loss of air pressure. These valves complete the containment structure, and form part of the third containment barrier for the fuel salt (Fig. 4). The MSIVs valves are necessary because the power-conversion system components cannot be relied upon to contain radioactive materials. The MSIVs are large and require an attached air-storage tank to provide closing air pressure in event that the system air pressure is lost.

A.4.5. Chemical Processing

Chemical processing was an integral part of the MSBR design, from which this design was developed. The requirements of the two systems, however, are not the same. Breeding was the primary goal of the MSBR, and all other considerations were secondary to achieving an acceptable the breeding ratio and doubling time. The theoretical breeding ratio is low in a thermal neutron spectrum, and parasitic neutron capture losses are not as easily accommodated as they are in a fast breeder. For this reason, fission products, especially those with high thermal-neutron cross sections, had to be removed rapidly for the MSBR. Also, the ^{233}Pa produced as an intermediate product in the ^{232}Th-^{233}U fuel cycle has a high thermal cross section and must be removed from the high-flux region of the blanket, where decay to ^{233}U competed favorable with neutron absorption. These considerations placed a great demand on the Chemical-Processing equipment for MSBR, and drove the design towards continuous, rather than batch, chemical processing.

In the ABC design, neutrons are not as highly valued in the overall neutron balance (they are expensive to produce, however[14,15]) because of the accelerator-based supply of additional neutrons to overcome fission-product poisoning. While these effects have not been studied in detail, estimates can be made. The ultimate plutonium-burnup capability of the system is determined in part by the fission-product concentrations in the salt. The burnup requirements set by the Department of Energy[5], therefore, will have an impact on the chemical processing requirements. Economic considerations also become important because while the accelerator can provide the surplus neutrons to make up for the fission product poisons, production of these neutrons requires electric power and significant capital expenditures, and thereby lowers the overall ABC efficiency and increases the total life-cycle costs[14].

Regardless of the eventual scope of the reprocessing equipment, it is assured that a chemists' reactor like this molten-salt design will require chemical-processing equipment. Equipment will be necessary to prepare the original charge and subsequent makeup charges of fuel and coolant salt. Another capability is to cleanup contaminated salt resulting from, for example, a steam-generator tube rupture. This cleanup capability must be available for both the fuel and coolant salts. Some fission product removal capability will probably (????) be included, although some proposed MSBR fuel cycles would allow fission-product processing to be operated in a batch modes after several years of ABC operation. Finally, off-gas processing must be included for both the fuel and coolant salts. As for the MSBR, the fuel-salt drain tank is expected to play a central role in the Chemical Processing Equipment, although the ABC design is stressing chemical separations that may be simplier and generate reduced waste streams[8] than those originally pursed by the MSBR project[10,32]; the viability of these newer approaches, however, remain to be demonstrated. Each of these chemical-processing areas is discussed in the following subsections as they impact the ABC plant layout: Salt Cleanup; Fuel-Salt Drain Tank; Off-gas Processing.

A.4.5.1. Salt Cleanup Systems

The salt cleanup systems for initial salt preparation, moisture removal, and fission-product or contaminant removal may all be one system, (????elaborate this claim or drop it???) and at the very least will share some of the equipment. One of the most important pieces of equipment is the fuel-salt drain tank, which is described in the next section. Other tanks are provided for extended storage of fuel salt and drainage of coolant salt.

4.5.2 Fuel-Salt Drain Tank

The fuel-salt drain tank, as defined in Ref. 47, is nearly as complicated as the reactor vessel. The fuel-salt drain tank, however, did not receive as much design attention as did the Core during the ABC Plant Layout Study. The fuel-salt drain tank serves several important functions. This system is the initial holding tank for Primary System off-gases; nominally a two-hour hold-up tank for these gases. The drain tank also serves as an integral part of the inherent shutdown mechanism, in that fuel salt drains into this space in almost all accident situations, as well as during schedule maintenance and repair events; the drain tank provides a cooled storage tank for fuel salt that can be used during maintenance on any component of the Primary System. Finally, the drain tank provides the connection between the Primary System and the fuel-salt cleanup systems.

The drain tank is cooled by a number (???how many, how, sizes, etc.??) of individual thimbles, each of which contains naturally circulating coolant. One of the available salts, preferably NaF-$NaBF_4$ or even Na-K, can be used as a drain-tank coolant. Each of these individual cooling systems eventually gives up its heat by generating steam in the base (???) of a stack. This cooling system is always operating to remove decay heat from the off-gases contained in the drain tank. (???say something about containment boundary/envelope in this regard???)

The drain tank is sized to contain all the fuel salt contained in the Primary System (??? m^3), plus a portion of the coolant salt. This provision is intended ot assure ample storage space even in the event of a gross (???? define) failure of the IHX.

The fuel-salt drain tank will require additional investigation during the detailed design. This system must be safety grade, because of the role played in the decay-heat removal system. Improvements in the design are likely to be identified as more detail is developed. (???what are we looking for here???).

A.4.5.3. Off-gas Processing System

The off-gas processing requirements of the ABC are much simpler than those of the MSBR. The MSBR was forced by ^{233}U-breeding requirements to remove volatile neutron absorbers shortly after formation. The ABC is expected to include a simpler system that is more akin that demonstrated on the MSRE[11]. Helium will be bubbled through the fuel salt in the bowls of fuel-salt pump. This helium will strip volatile fission products from the fuel salt and carry them into the off-gas processing system for eventual store on zeolite beds (activated charcoal was originally proposed for MSBR). Although the details of the off-gas processing system have not been developed, a simple system is envisioned. It may even be possible to reintroduce some of the volatiles (primarily noble gases) back into the fuel salt along with the helium. Such a recycle will greatly reduce the volume of zeolite or carbon hold-up beds required.

A.4.6. Instrumentation and Control

Development of instrumentation and controls for the fuel-salt (including the neutronics), the coolant-salt, the steam-conversion, the chemical-processing, and the safety systems will require a major effort. At the time of the MSBR program cancellation, much of the needed development work had been identified and begun, technological advances and regulatory changes have occurred, and the entire I&C development program plan needs to be reviewed and modernized. In this section, the I&C requirements for the ABC system are identified, to the extent possible at this stage of preconceptual design. Some of the development needs this capability are discussed.

All of the standard detection devices to measure level, pressure, and temperature in molten-salt environments are needed. In addition, more advanced diagnostics are needed to perform on-line measurement of plutonium concentration, fission-product concentration, REDOX potential of the salt, and moisture content of the salt. The two most important capabilities are plutonium concentration and REDOX potential measurements; both are necessary to prevent plutonium precipitation and/or plateout, as well as controlling fission-product concentration and corrosion.

Many of the control elements needed are readily adaptable from other applications. One item that may require additional development is the neutronic monitoring and SCRAM system. The control system must be made to respond quickly to off-normal situations while providing operational flexibility. The large population of accelerator-produced neutrons may complicate the monitoring and control systems (???why???). The increase demands of improved control in the ABC in large part stem from the need to protect capital investment and maintain high plant availability, rather that to deal with increased safety concerns (???).

A.4.7. Operation and Maintenance

Operations and maintenance has become an important part of commercial power reactor costs over the last two decades.(???reference???) The O&M costs have risen much faster than expected (really????, reference)???). The primary lesson learned from this experience is that the design should include O&M considerations from the outset to prevent high costs in the future. It is economically justified to increase capital costs to provide room for routine operations and infrequent replacement and maintenance operations. These considerations are amplified in a fluid-fuel designs.

The general principles, adopted from the breeder program, guided the ABC Plant and are listed as follows[47]:

- Each system is composed of manageable units joined by suitable disconnects and lines which can be cut and rewelded remotely.
- Each unit is accessible and replaceable from directly above through removable shielding.
- Failed units are removed and replaced. (???is this worthy of of a canonic status????)

These design requirements increased the size of the containment building and the associated cost, but it is expected that adherence to these canons will greatly simplify both planned and unplanned maintenance. Components requiring remote replacement are expected include the graphite moderator stringers, the target, the Hastelloy-N target thimble, fuel-salt pumps, control rods and shutdown rods, the bending and focusing magnets in the beam tube, the beam expanders, the IHX tubesheets, fuel-salt piping, lead target system components (pumps, heat exchangers, piping), salt-processing equipment, off-gas processing equipment, and various I&C components. Replacement of most of the other components with longevities less than ~40 yr can be performed using direct maintenance, with a suitable cooling period. For example, the activation product concentration in the coolant salt is expected to decay within a few days (???really???) to a level allowing contact maintenance on the steam generator and steam reheater.

A.4.8. Safety Systems

Safety systems have always been included on nuclear reactors, but modern regulatory requirements continue to place increasing reliance on these systems to protect both public safety and plant investment. The safety systems for the ABC design have been identified at a level of detail commensurate with the available plant detail. Because of the additional regulatory requirements imposed on safety systems, an attempt has been made in this section to identify the main safety systems anticipate as needed for ABC.

(???paragraph needed to describe the three-tiered containment systems of Fig. 4, and how it is similar to and different from regular fission reactors????)

A SCRAM system for the accelerator and the blanket will be included. Based on a number of trip signals, the proton beam will be diverted to a beam stop (or, more likely, the entire accelerator will be shut down) and both the control and shutdown rods will be inserted into the core. Components included in the SCRAM system include the neutron monitors, the digital control system, the shutdown and control rods, and the diverter magnet control system or power supply.

A related system is the containment isolation system that closes a number of isolation valves (*e.g.*, BTIVs and MSIVs) in response to certain off-normal situations. The isolation system is to prevent or mitigate the release of radioactivity to the environment. The containment-isolation system forms an integral part of the third barrier Fig. 4) that is dictated as a design requirement for this ABC system. Components included in the containment-isolation system are primarily valves: main steam isolation valves; isolation valves on the tertiary cooling loop for the target; isolation valves on the atmosphere-control system exhaust; and isolation valves on the accelerator beam line. The other components are detectors and controllers for these valves.

A related system is the fuel-salt drain system that under certain conditions opens freeze valves that are installed between the fuel-salt system and a drain tank. By draining the fuel salt, the loss of fuel-salt (*via* a pipe break) can be mitigated, ensure shutdown of the fission reaction, and ensure continued decay-heat removal (*via* the passive heat-removal system in the tank). Included in the fuel-salt drain system are the necessary detectors, the freeze valves, the piping, and the drain tank itself. The passive heat removal system for the drain tank might also be placed in this category.

The entire primary fuel system, the molten-salt processing system, and the off-gas system may all be considered safety systems because they contain radioactive materials. This broad interpretation of the definition may be correct, but all these systems are not described here because of the details about them provided elsewhere in this report.

A.5. Flow Calculations

The following sections describe the calculations used to determine flow rates and flow velocities in the fuel-salt, coolant-salt, and steam systems. Although some scaling was possible for the coolant-salt and steam systems, the fuel salt proposed for the ABC design is sufficiently different from that of the MSBR that scaling was not possible. The MSBR fuel-salt contained an appreciable quantity of thorium, which changed the physical properties of the salt. While physical properties are not available for the plutonium-containing salt to be used in the ABC, the plutonium concentration is sufficiently low that the values for pure LiF-BeF_2 can be used without introducing significant error.

A.5.1. Fuel Salt

The ABC flow calculations rely in part on those performed for the MSBR design. However, the MSBR was designed to breed ^{233}U and, therefore, included a substantial thorium fraction (???%, ???kg/m^3) in the fuel. The ABC fuel is basically LiF-BeF_2 carrier, with plutonium present almost at impurity levels (*i.e.*, less than 1%). The properties of the ABC fuel are not the same as those of the MSBR fuel. Physical property data for the plutonium fuel is not available. However, information is available for pure LiF-BeF_2 and, therefore, is used to describe the ABC fuel salt.

The fuel-salt enters the Core at 839 K (1050° F) and leaves at 978 K (1300° F). The physical property values are as follows. The density of the fuel salt is 1,976 kg/m^3 at 830 K (1050° F) and 1,918 kg/m3 at 978 K (1300° F). These values are taken from page 28 of Ref. 46. The specific heat is 2,385 J/kg/K (page 22 of Ref. 46) The fuel-salt velocity in the Core can be found by assuming that 711 MWt are produced in the molten-salt blanket, and the flow area is equivalent to the Core cross section, $A_{COR} = \pi R_{COR}^2$, times the fuel-salt fraction, f_{MS}. The mass flow rate through the Core, the fuel-salt velocity, and the volumetric flow rate are are given by

$$G_{FS} \bullet (m^3/s) = \frac{\dot{M}_{FS}}{\rho_{FS}} \tag{A-4a}$$

$$M_{FS}\,(kg/s) = \rho_{FS}\, v_{FS}\, A_{BLK} \tag{A-4b}$$

$$v_{FS}(m/s) = \frac{P_{TH}/N_{BLK}}{\rho_{FS}\, A_{BLK}\, f_{FS}\, c_p\, \Delta T} \; . \tag{A-4c}$$

Using the parameters listed above with R_{BLK} = 1.75 m and f_{FS} = 0.13 (A_{BLK} = 9.61 m^2) gives v_{FS} = 0.88 m/s, $\dot{M}_{FS}$ = 2,146 kg/s, and G_{FS}= 1.1 m^3/s.

The fuel salt leaves the Core through two 0.40-m diameter pipes. The pipe wall thickness, t_w, is determined from the maximum allowable stress. Assuming the maximum fuel-salt pressure is 1.4 MPa (200 psi), the minimum wall thickness is given by[50]

$$t_W(m) = \frac{pR}{\sigma} \; , \tag{A-5}$$

which for σ = 23.8 MPa (???seems low????) gives t_w = 12.7 mm. The inner diameter of this pipe, therefore, is 0.38 m, for which the velocity in this pipe is 4.9 m/s. The mass flow rate and volumetric flow rates are half the corresponding Core flow rates. All the other fuel-salt piping is of the same diameter and have similar velocities. The fuel salt in the IHX passes through N_{TUB} = 7,793 tubes, each having a 7.7-mm ID; the fuel-salt velocity of in these tubes is 2.0 m/s.

A.5.2. Coolant Salt

The coolant salt, NaF-$NaBF_4$, enters the IHX at 894 K (1150°) and leaves at 728 K (850° F); the respective densities are 1,811 and 1,930 kg/m^3. To remove the 711/2 = 355.5 MWt from the shell side of the IHX with this temperature difference, the mass flow rate must $\dot{M}_{CS}$ = 1,415 kg/s for c_p = 1,507 J/Kg/K. The average fluid velocity is 0.45 m/s based on a shell-side flow area for the coolant salt of $[(\pi/4)\times(1.718^2 - 0.505^2 - 5803\times0.00953^2) =]$ 1.68 m^2. (???review origin of this diameters??). In the smaller-diameter piping (0.51-m OD, 0.47-m ID, 0.1734-m^2 flow area), the average flow velocity is 1,415(kg/s)/1,811(kg/m^2)/0.1734 (m^2) = 4.5 m/s. In the larger (0.63-m OD, 0.57-m

ID, 0.2565-m^2 flow area) piping that passes through the reactor-vessel cell wall, the the average coolant salt velocity is $2\times1415(\text{kg/s})/1811(\text{kg/m}^3)/0.2565(\text{m}^2) = 6.09$ m/s. The 711-MW heat load is split unevenly between the steam generator and the steam reheater, with most of the load going to the steam generator according to the ratio 617/94. The coolant salt flows through the shell side of the steam generator with a flow velocity of $(617/711)\times1415(\text{MW})/1870(\text{kg/m}^3)/0.657(\text{m}^2) = 0.53$ m/s, where the flow area is $(\pi/4)(1.5^2 - 5115\times0.0127^2) = 0.657\ \text{m}^2$. The flow area in the steam reheater is $(\pi/4)\times(0.80^2 - 751\times(0.0191)^2) = 0.2846\ \text{m}^2$, and the flow velocity through the reheater shell is $(94/711)\times1415(\text{kg/s})/1870(\text{kg/m}^3)/0.2846(\text{m}^2) = 0.35$ m/s.

A.6. Outstanding Issues for the Reference Steam Design

The ABC Plant Layout Study is based to the greatest extent possible on the MSBR conceptual engineering design[10,11]. The Ref.-10 report devotes an entire chapter to the uncertainties and outstanding issues associated with the design. Because little additional work has been performed since the completion of the MSBR project in 1971, the outstanding issues have changed little. This section is based closely on Chapter 16 from Ref-10, with changes only where appropriate to the ABC application or where new insights have developed. In addition to the MSBR issues, additional and generally MSBR-unrelated issues associated with the ABC design, particularly with respect to the accelerator and the target system; these issues are also included in this section. A topical division according to Materials, Engineering Design, and "Other" is used to express these issues from an MSBR-related engineering perspective and to provide material for that component of the ABC R&D plan[9].

A.6.1. Materials

A.6.1.1. Fuel Salt

The fuel salt for the MSBR was a mixture of $LiF\text{-}BeF_2\text{-}ThF_4\text{-}UF_4$; the thorium was included for breeding purposes. The $LiF\text{-}BeF_2$ mixture was used as the base salt because the primary focus of the program was fissile-fuel breeding using thermal neutrons. The $LiF\text{-}BeF_2$ salt, once 99% of the ^6Li isotope is removed, has the lowest parasitic absorption of the available salt mixtures and, therefore, allows the highest possible breeding ratio. For the plutonium-burning ABC mission, the $LiF\text{-}BeF_2$ salt may not be an optimum choice. This salt is the best characterized of all the potential salt mixtures; most of the testing done in conjunction with the MSRR program was with a $LiF\text{-}Be_2$-based salt. Difficulties associated with this salt include a relatively high melting point [732 K (858° F)], limited solubility of plutonium (<???? molar or ???? kg/m^3), and production of tritium through neutron interactions with the lithium, even if most of the ^6Li isotope is removed.

An alternate salt that was considered for use in the MSBR is $NaF\text{-}ZrF_4$. This salt reduces the tritium production by a factor of ????? to the fission yield. The increased parasitic absorption with this salt is not nearly as important for the plutonium-burning accelerator-driven ABC as it is in a breeder system. A primary drawbacks of this alternate salt is the lack of specific information on its behavior and compatibility with construction materials. Also, the solubility of plutonium in this salt and the melting point of this salt are unknown. They may be better than the reference salt. (???? hard to believe, this knowledge deficit needs to be corrected????)

A.6.1.2. Secondary Coolant

The NaF-$NaBF_4$ coolant chosen as the reference secondary coolant for both the ABC design and the MSBR has some associated uncertainties related to materials compatibility in the operating system, maintenance of the cover gas, and a high melting point (???? K). The complex feedwater and reheat systems are driven by the high melting point of this coolant. Some potential design improvements could be made if a better secondary coolant can be found. However, the NaF-$NaBF_4$ has been shown to trap tritium under certain conditions (????elaborate???) and holds promise as a means to trap and prevent tritium migration into the steam system.

Alternative coolants have been examined, including LiF-BeF_2, HITEC, KF-ZrF_4, He, and NaF-LiF-BeF_2. Each alternative has advantages and disadvantages, and none is a clear better choice than NaF-$NaBF_4$.(???can a table of advantages/disadvantages be included so all this comparative assessment is not lost???) As long as a lithium-based fuel salt is used, tritium trapping in the secondary coolant salt is of prime importance in limiting the off-site releases. One option that may simplify the entire design, at the expense of some thermal efficiency, is use of helium as the secondary coolant driving a gas turbine directly without use of an intermediate loop. Preliminary calculations using realistic pressure drops, equipment efficiencies, and recuperator effectiveness indicate that a gross plant thermal efficiency of 32.5% is possible using helium; an efficiency of 35% was estimated if a nitrogen secondary coolant and direct-cycle (????) conversion was used. Use of either gas will allow almost complete trapping of tritium in the secondary coolant.(????really?, how to remove tritium from N_2????).

A.6.1.3. Hastelloy-N

Use of modified (???modified how???) Hastelloy-N has been assumed for all primary and secondary systems in contact with molten salts. One of the outstanding issues associated with use of Hastelloy-N is the ability to survive the neutron irradiation and temperatures in the reactor vessel and the target thimble. Graphite shielding of the reactor vessel in the reference fully moderated design mitigate that concern somewhat, but the longevity of Hastelloy-N component in the target thimble remains an unknown. Neutronics and thermal calculations are needed to characterize better the neutron flux and energy spectrum to estimate the component lifetime. Designs using mechanical joints or easily accessible welds have been considered that promise rapid replacement of the target thimble during maintenance shutdowns. If such a design is found, a thimble lifetime of only a few years will be adequate. A lifetime of less than one year will require re-design of the target/blanket assembly or consideration of alternate alloys. The Hastelloy-N was chosen for reasons of chemical compatibility with the fuel salt. Other alloys limited by corrosion lifetimes may provide the best combination of chemical and irradiation compatibility, at least for the thimble. All other areas in contact with molten-salts will be fabricated from (modified) Hastelloy-N. Additional work will be required to prove the suitability of modified Hastelloy-N for use in pressure vessels (*e.g.*, steam generator???) and for stress-corrosion compatibility with steam.

A.6.1.4. Graphite

Manufacture of the graphite blocks prescribed for the MSBR design was considered an outstanding issue. Extrusion of 6-m blocks was considered problematic, although extrusion of shorter blocks was considered readily obtainable. The ABC system as currently envisioned uses blocks less than 3-m in length. It is not known whether these lengths are obtainable. (????why include this last statement???)

The high-quality (???define???) graphite needed for the ABC system may also not be readily obtainable.(???really???) However, the quality of graphite needed is not as high as that required for the MSBR design because the ABC system is not concerned with breeding and the associated high tolerances on parasitic neutron absorption. Sealing of the graphite to prevent diffusion of fission gases and fission-product-ladened molten salt into the graphite will not be required. A related issue of great import to the ABC system is the quantity of contaminated graphite that must be disposed as High-Level Waste. The magnitude of this potential waste stream is a function of the quantity of graphite in the core (an undermoderated system *versus* a fully moderated system, Appendix E), the lifetime of this graphite, and the difficulty of post-irradiation cleanup of graphite to achieve a more benign disposal classification. It has been suggested (???by whom???) that the production of any amount of contaminated graphite in the high-level-waste category is unacceptable. If that is the case, the reactor-vessel lifetime will be reduced, (???how does this follow???) perhaps to an unmanageably short lifetime. Incineration of the graphite is probably not an option because of the ^{14}C production in the graphite.

A.6.2. Engineering Design

A.6.2.1. Blanket Core

The blanket core design is largely based on the MSBR core design. Neither concept has undergone a detailed design at a level necessary for construction. An overriding concern is the temperature and stress distribution in the Core, especially in the region that surrounds the Target thimble and in the upper plenum. As presently envisioned, a bypass flow will be used to maintain the thimble at a lower temperature than the bulk fuel-salt outlet temperature. This thermal-hydraulic will minimize the parasitic heat loss to the Pb system, and will minimize the materials concerns for the both inner and outer thimbles. Whether the graphite and the Hastelloy-N thimble can withstand the thermal transients that are expected in the upper plenum is unclear. Mixing of hot and cold streams must occur in the upper plenum to minimize stresses in the fuel-salt outlet piping running from the Core to the fuel-salt pumps.

Another potential Core issue is the need to assurance against flow instabilities, especially if an open (undermoderated) Core design is chosen. Aside from the aforementioned problems of graphite waste and a generally more complex and difficult-to-maintain configuration, the fully moderated Core is preferable to the undermoderated design.

No control (startup and shutdown) rods have been included in the design at this stage, although the need for these systems is recognized as being necessary from a safety/licensing standpoint. Graphite displacement rods are considered adequate to satisfy these requirements.

A.6.2.2. Intermediate Heat Exchangers

The IHX design was copied from the MSBR design, with the exception of the coolant-salt outlet nozzle, which was part of a concentric pipe arrangement in the MSBR design. The heat-transfer data used in designing the MSBR heat exchanger was incomplete. Also, if an alternate fuel salt or coolant is used, the IHXs must be re-sized and possibly redesigned to maximize the efficiency.

A.6.2.3. Molte-Salt Circulation Pumps

Although no salt circulation pumps of the size prescribed for the MSBR were ever operated, at the time it was believed that scale-up of the pumps used in the MSRE and in numerous pump loops was readily obtainable. The pumps prescribed for the ABC design are of a smaller capacity than those required by the MSBR. Re-designing these pumps will not be problematic, however, this technology is becoming more difficult to resurrect.

A.6.2.4. Drain Tank

The drain-tank design used for the ABC system was adopted to the extent practical from the MSBR design. The MSBR drain-tank design had never been optimized, this area deserves additional attention to identify potential improvements in the design of this critical component. For example, the question of whether or not a passive heat-removal system for the drain tank can function as the ultimate heat-removal system in the event of a loss of power must be addressed, or a separate passive heat-removal capability might have to be provided for the reactor vessel *per se*. Many other issues associated with the drain tank and its cooling systems can be identified. The design is dictated to some extent by the requirement that three barriers (Fig. 4) be provided for the fuel salt at all times, including while the fuel salt is stored passively in the drain tank. Also, the heat extracted from the drain tank is wasted in the present design; it may be possible to capture this heat for beneficial purposes, if passive heat removal is not required.

A.6.2.5. Fuel-Salt Drain Valve

It has been assumed that the fuel-salt drain valve can be designed as a freeze valve, although the details of the design have not been addressed. This area remained unresolved at the conclusion of the MSBR program. A valve development program will be necessary to choose the optimum design. Implementation of a number of parallel freeze valves of the kind used in the MSRE represents an alternative for ABC. (???say something about this valve???) The lifetime of that type of freeze valve is uncertain, and and means for periodic replacement would have to be incorporated into the design.

A.6.2.6. Bubble Generator and Separator

The fission-product gases in the MSBR were to be swept from the salt in a bleed-stream processing system that introduced helium purge gas into the salt, allowed circulation of this helium through the core, and separated the resulting mixture of helium and fission gases. This procedure was necessary in the fissile-fuel breeder designed for neutron economy. It has not been determined what, if any (????), separation of fission gases from the fuel salt would be required for ABC. Because neutron economy is reduce in importance, a higher concentration of parasitic fission products can be accommodated. However, it would be desirable to maintain low concentrations to prevent unanticipated reactions such as the tellurium-Hastelloy problem encountered in MSRE. It therefore, may, be advisable to include a gas stripping system in the ABC design. No space provision in included in the design at this stage, although this equipment is not expected to be large. The same bleed stream may be processed for noble-metal removal if this is deemed necessary. The entire issue of necessary on-line fuel salt processing is unclear and warrants attention at this point.

A.6.2.7. Off-gas System

The off-gas system for the MSBR conceptual design was not completed in detail. It was acknowledged at the time that additional effort would be required to complete and optimize the design of this system. No major problems were anticipated. As discussed in the last section, the entire issue of fission-product removal and off-gas processing needs to be re-addressed for ABC. Some degree of off-gas processing will be necessary, since helium purge gas is required for a number of components (*i.e.*, salt pumps) and this purge gas will have to be cleaned prior to release or reuse.

A.6.2.8. Steam Generators

The steam generators described for ABC are scaled from the MSBR design, with the exception of the inlet/outlet nozzle placement. The once-through U-tube/U-shell design that is heated by molten-salt and cooled supercritical steam has never been built, and a development program for this component will be required. It is expected, however, that this effort will be straightforward. Two of the outstanding issues are the required feedwater temperature and the stress-corrosion resistance of Hastelloy-N on the steam side.

A.6.2.9. Instrumentation and Control Systems

The I&C systems have received no attention at this stage of the design. It is recognized that cooling will be required for a number of these I&C components in the reactor-vessel and steam-generator cells because of the high ambient temperatures (???? K). Hardening for radiation resistance may also be required. This is an important area that needs more attention, particularly when the complexities of an accelerator-driven multiplexed system that is aiming at high availability is considered.

A.6.2.10. Cell Wall Construction

The reactor-vessel and steam-generator cell walls must be cooled to prevent dryout of the concrete. It has been assumed that a steel liner with embedded cooling water channels will be used to maintain the concrete at a low temperature. The design of the cell wall will require more detailed thermal calculations to assure that adequate concrete protection can be provided while simultaneously minimizing the parasitic heat loss from the cell atmosphere to the cell-wall cooling system. A number of related issues must be resolved before the cell-wall design can be completed. The first issue deals with whether cooling of the primary system (and/or the secondary system) is required, or whether the passive system provided in the drain tank adequate. A choice must be made between an oven concept for heating the Primary System and Secondary Cooling-Salt System prior to startup, as has been assumed in this design, or a system based on component heating. The temperature that is best for operating the reactor vessel cell to minimize parasitic heat losses from the fuel salt and to allow those components that require cooling (I&C) to be cooled must be determined. Another issues is whether the cell walls should be insulated; the cell walls cannot be insulated if they are to be used as an integral part of a passive heat-removal system for the fuel salt. Finally, the thicknesses of the walls used in the design so far have been arbitrarily selected and may need to be increased in the future to support the overlying equipment.

A.6.3. Other Outstanding Issues

A.6.3.1. Tritium

As long as a lithium-based salt is used for fuel salt, tritium production will be large. Both the MSBR and the ABC designs call for the use of 99.99% ^7Li. For the ABC design, tritium is produced at a rate of 27 g/yr, or 750 Ci/d, for the 711-MWt module, as scaled from the MSBR design. The tritium produced easily passes through most materials at the high temperatures used in the MSBR or ABC designs. The tritium is expected to be a component in the off-gas system, the reactor-vessel-cell atmosphere- processing system, the coolant salt, the steam-generator-cell atmosphere system, and the steam system, unless confined/removed before reaching the SCS system. From the steam system, the tritium is expected to pass through the main condenser to the environment. The magnitude of the expected tritium release from the condenser under uncontrolled conditions cannot be licensed; it is estimated that this release rate (750 Ci/d) must be reduced by over a factor of ~100.(????) Additional work is needed to address where this tritium migrates and the means by which it can be confined.

Various methods of controlling tritium were considered., including: a) increasing the sparging rate in the fuel salt, b) increasing the U^{4+} to U^{3+} ratio, increasing corrosion concerns, c) sparging with HF, d) trapping in the NaF-$NaBF_4$, and e) using an alternate coolant that would allow even better trapping. Other methods of controlling tritium that are available to the ABC program include use of an alternate fuel salt that does not contain lithium (NaF-ZrF_4) and use of a gas turbine system that allows trapping of tritium in the gas coolant and separation from the coolant after oxidation.

A.6.3.2. Chemical Processing

The Chemical-Processing system envisioned for the MSBR was a complex system that removed most of the fission products and protactinium. This level of complexity may not be necessary for the ABC design, but processing will nevertheless be required. The reactivity decrease resulting from fission-product buildup can be accommodated by increasing the plutonium concentration or by increasing the beam current, but strong economic, safety, and operational constraints intervene to limit these "fixes" to otherwise unresolved chemical-processing issues. Nonetheless, increased parasitic absorption is otherwise not as critical as for a fissile-fuel breeding system based on thermal neutrons. Chemical processing to remove oxide impurities, secondary coolant (in-leakage??), and certain problematic elements (the noble metals perhaps) will be required. Processing to remove these elements is an open issue and needs to be resolved. The amount of processing required and the processing rates need to be determined. One approach would operate the fuel salt on an eight-year cycle; at the end of the cycle, the fuel salt would be drained into a tank and processed for the soluble fission products, which makeup ??????% of the total fission products. An attractive feature of this arrangement is the low processing rate that is conducted more or less "off-line". For a 711-MWt ABC module, the fuel- salt volume is less than 16 m^3, so the required fuel-salt processing rate (for complete regeneration of clean salt) is in the range 2-3 m^3/yr. An important goal is to develop a a self-contained processing module that would process at this rate. In the event of a failure, the entire module would be replaced as a unit. This processing module is not the same as the one which would be used to clean up contaminated salt (*i.e.*, from a Primary System leak or during startup). Also, as indicated above, this self-contained unit deals only with ?????% of the fission products generated.

A.6.3.3. Fission-Product Distribution

It is not known how the fission products will distribute during reactor operation. Approximately ????% chemically "noble" with respect to fluorine and, therefore, are expected to plate out onto cooler surfaces within the fuel-salt system or otherwise form insoluble precipitates in the fuel salt. Cooling must be provided for the affected components to remove local disposition of decay afterheat. Given that approximately ??????% of the fission products are in gaseous form, the remaining ????% will build up in the fuel salt and, depending on the above-mentioned cost/safety/inventory issues, can be processed batchwise and outside the Core environment, as suggested above. Additional experimentation will be required to advance understand fission product-behavior. It is possible that separation of fission products from the fuel salt may be promoted (*i.e.*, on the heat exchanger surfaces) beneficially.

A.6.3.4. Steam Conditions

The supercritical-steam system proposed for the ABC design is scaled from the MSBR design to the extent practical. The MSBR steam system was itself taken from the Bull Run Steam Plant (coal-fired) design[25]. The efficiency of this supercritical cycle is high (η_{TH} = 0.444)), but this high efficiency is obtained through increased system complexity. A simpler system with lower thermal efficiency may possibly be more advantageous in terms of cost, reliability, and R&D needs for the ABC application. mission. The ABC burner is not intended to be an economical power producer, *per se*; power production is to offset the capital and operating costs, especially the accelerator capital cost and annual charges for power consumption. While high η_{TH} values help with the ABC total life-cycle cost, the overall question of these η_{TH} trade offs is an open issue and is closely related to considerations of alternate secondary coolants. For example, the helium (or nitrogen) gas turbine system would eliminate the steam system altogether, while use of HITEC or another secondary coolant likely will require re-design of the steam conditions.

A.6.3.5. Maintenance

It was recognized early in the ABC Plant Layout Study that remote maintenance was a necessity for large portions of the ABC plant. Space for remote maintenance equipment was included in the plant layout, although it is not certain that the allowed space is sufficient. Design of many of the components (IHX, target, *etc.*) as well as the plant layout was dictated by the remote maintenance considerations. Although laydown area was considered, the available space may be insufficient. This area will require additional consideration as the design progresses. Other important issues relate to the degree of "on-hands" maintenance that will be possible on the accelerator; the degree to which the three-containment-barrier system and philosophy can be laid aside during a module maintenance operation; a wide range of minor and major maintenance needs that are unique to a fluid-fuel system; and special maintenance issues related to operations on a highly multiplexed system like ABC.

A.6.3.6. Safety Analyses

This ABC Plant Layout Study has not conducted a concerted safety study, although some preliminary safety issues have been analyzed. Because of the low vapor pressure of the fuel salt, the energy contained in the primary system is much less than that of a similarly sized PWR. Nevertheless, a number of safety issues are known to exist that are both generic to a driven nuclear reactor and unique to the fluid-fuel nuclear power plant. These issues must be analyzed by future conceptual design studies of the ABC and include: positive reactivity insertion (fueling error) with and without accelerator trip; failure of drain-tank freeze valve; loss of target and fuel-salt flooding; loss of Primary System pump; (???others more related to the Accelerator Equipment/Primary System area ????); steam-generator tube rupture; a steam system break inside containment; an Anticipated Transient Without SCRAM (???) (ATWS); an inadvertent drain of the Primary System; etc. Without a detailed conceptual design, all safety analyses are qualitative and superficial compared to future needs in this important area.

Appendix B. Accelerator Scaling Relationships for ABC

Minimization of the "wall-plug" power, $P_{EA} = P_B/(\eta_{DC}\,\eta_{RF}\,\eta_{WG}\,\eta_B)$, required to create a proton beam of power, $P_B = I_B E_B$, per unit neutron source, $I_n(n/s) = YI_B/e$, is one of a number of important considerations in optimizing the ABC accelerator for minimum capital (power supplies, higher-average-current accelerator structure) and operating (electrical-energy) costs. In the above expression for P_{EA}, the four efficiency factors, η_j, are: AC → DC conversion, η_{DC}; DC → RF conversion, η_{RF}; waveguide transport losses from RF power supplies → RF accelerator cavity, η_{WG}; and RF cavity → beam coupling. Typical values for η_j are $\eta_{WG} = 0.95$, $\eta_{DC} = 0.95$, $\eta_{RF} = 0.65$, and the ratio of (time-averaged) beam power to total power delivered to the accelerating RF cavity (average beam power + average RF power) is given by,

$$\eta_B = \frac{1}{1+I^*/I_B} , \tag{B-1}$$

where $I^* = G\,f_D/R_s/\cos\phi$ is an accelerator parameter comprised of: the accelerating field gradient averaged over the accelerator structure, G(MV/m); a nominal value for the cavity shunt resistance, R_s(MΩ/m); cosine of the phase angle between the accelerating voltage and the proton packet or bunch, $\cos\phi$; and the fraction, f_D, of the time that the accelerating RF voltage is applied (6% for LAMPF, 100% for ABC). This expression is applied here only to describe the CCL, which is the major power consumer, although similar efficiency scalings can be applied to the less-power intensive front-end accelerating elements[14] (Fig. 6). For values typical of ATW/ABC[14,20] $G \sim 1$ MV/m, $R_s \sim 36.$ MΩ/m, $\cos\phi = 0.87$, and $f_D = 1.0$, it follows that $I^* = 0.032$ A. From the viewpoint of RF-cavity → beam coupling efficiency, high-current and reduced-energy accelerator parameters are favored, although issues related to injector (ion-source) technology and to beam stability and scrape-off losses establish limits on the degree to which high currents can be pushed to reduce η_B; for ATW/ABC parameters (I_B = 200-250 mA), η_B approaches ~0.85, which projects to an overall ("wall-plug") efficiency of $\eta_A = P_B/P_{EA}$ ~0.48-0.50. Comparable values can be achieved for the lower-I_B ABCs through comparable reductions in the RF-duty factor, f_D, although such reductions have cost impacts.

Minimization of the "wall-plug" energy invested in each neutron is an important objective for either "neutron factory". Defining $E_n(MeV/n) = P_{EA}/eI_n$ and using the off-set linear representation for the target neutron yield per proton, $Y(n/p) = (E_B - E_B^o)/y$, where for fitting parameters E_B^o(= 200 MeV) and y(= 35 MeV/n) typical of ABC target, the following expression for E_n results:

$$E_n(Mev/n) = \frac{y}{\eta_{DC}\ \eta_{RF}\ \eta_{WG}} \ \frac{E_B/E_B^o + (P_B/E_B^o)^2}{E_B/E_B^o - 1} , \tag{B-2}$$

where $P^*(MW) = I^*\ E_B^o$ is a design parameter that characterizes both the accelerator and target. Equation (B-2) is plotted on Fig. B-1 and illustrates the optimum energy cost to

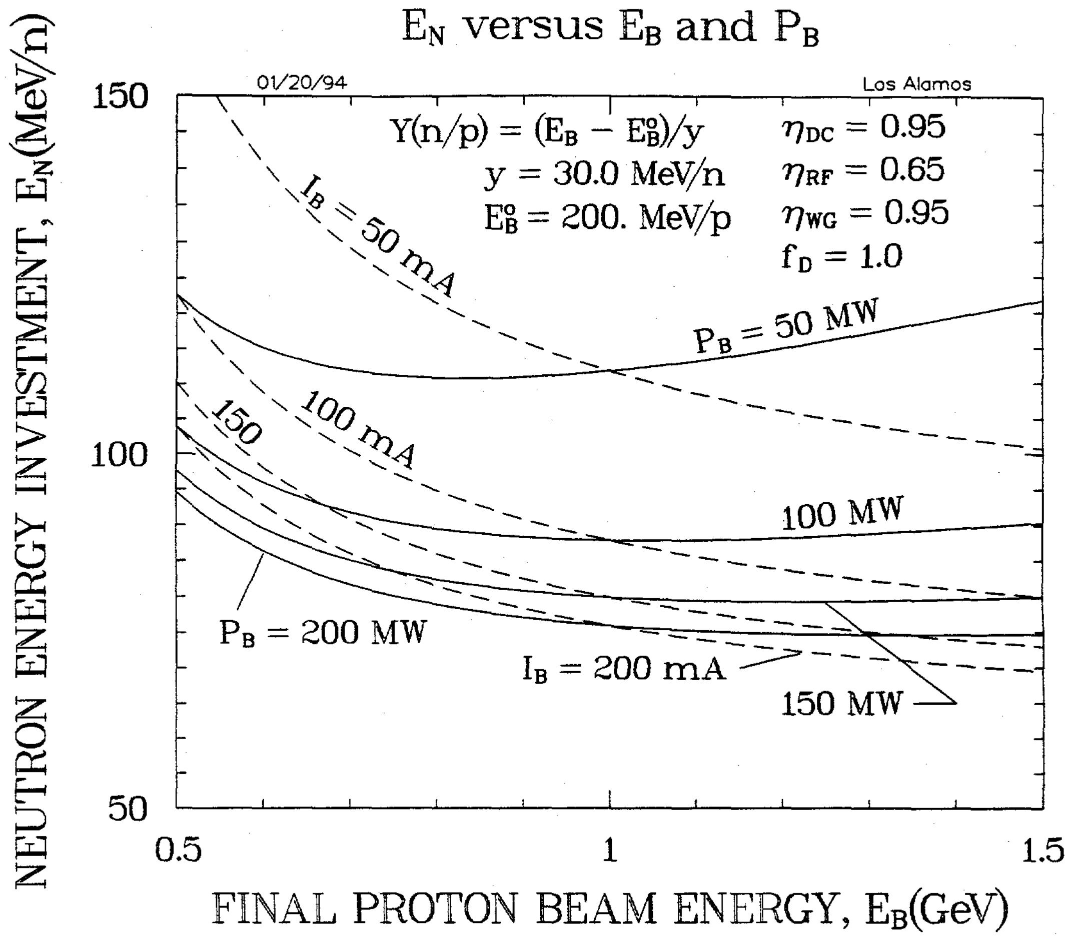

Figure B-1. Dependence of "wall-plug" energy required to generate a spallation neutron in beam energy and/or (peak) beam current, showing the optimum beam energy resulting from a trade off between accelerator efficiency (higher beam current and lower beam energy for a given beam power is favored) and target neutron yield per proton (higher beam energy is favored)

produce a neutron resulting from the balance for a given capacity, P_B, between increased E_B (increased neutron yield per proton) and reduced E_B (increased I_B and increased η_B). For a given beam power, accelerator (CCL) structure, and target-yield characteristics, the minimum "wall-plug" energy investiture per source neutron occurs at the following beam energy and has the following values, respectively:

$$(E_B / E_B^o)_{min} \; 1 + \sqrt{1 + P_B / P^*} \qquad \text{(B-3)}$$

$$\frac{(E_n)_{min}}{(E_n)_{\infty}} = 1 + \frac{2(1 + \sqrt{1 + P_B / P^*})}{P_B / P^*} \; . \qquad \text{(B-4)}$$

For the accelerator efficiencies and target parameters suggested above, $(E_n)_\infty = y/(\eta_{DC}\ \eta_{RF}\ \eta_{WG}) = 94$ Mev/n and $P^* = 5.7f_D$ MW. The decrease in the minimum E_n(MeV/n) with increasing beam energy is accompanied by a decrease in the peak-to-average current, $f_D = I_B/I_B^{max}$, and, hence, increased demands on the injector(s) and the local accelerating structure. The constraint on I_B^{max} is expressed on Fig. B-1 in terms of the parameter $P_B/P^* = E_B\ I_B^{max}/P^*$ as lines of constant I_B^{max} by plotting Eq. (B-1) accordingly.

Appendix C. Accelerator-Based Conversion (ABC) Spallation Target: Function and Design Issues

The technical function, design bases, options and issues of an efficient neutron spallation target are summarized in this Appendix. The basis of the ABC target design reported in Sec. III.B.2. rests primarily with the generic material reported herein.

C.1. Target Function

The function of the Accelerator-Based Conversion spallation target is to convert high-energy ($\geq$ 800 MeV) protons generated by the accelerator to lower-energy (< 20 MeV) neutrons, and to deliver these spallation neutrons to the plutonium-bearing blanket. A functional diagnal shown for the target is in Fig. C-1. In performing this function, the

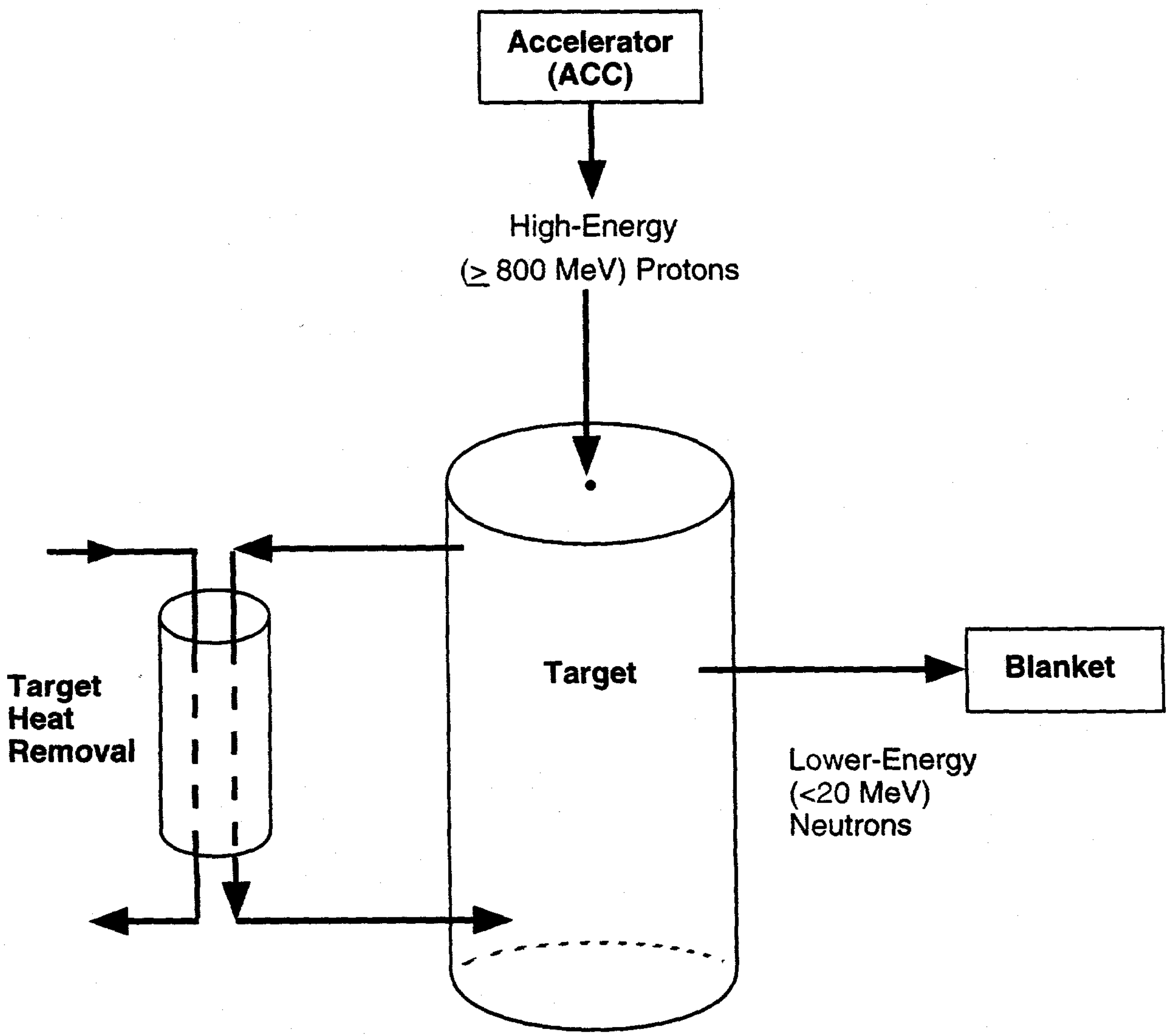

Figure C-1. Schematic representation of target functional performance.

target must be cooled sufficiently for steady-state operation and designed to reduce risk related to radiological release or other detrimental consequences resulting from off-normal operating conditions (*e.g.*, loss of target coolant, maladjusted beam distribution, *etc.*).

The conversion of high-energy protons to neutrons relies on intranuclear reactions between the incident protons and the nucleons in the target material. Consequently, to maximize neutron production, the target material should have a large number of nucleons per individual nucleus; a high-Z material is preferred. While the protons can interact directly with bound neutrons to eject them from the nucleus, the emitted neutrons tend to be peaked in the forward direction and to have very high energy (from >20 MeV up to the proton energy). These neutrons are inefficiently used in the blanket, since the slowing-down length is large, even for efficient neutron-moderating materials. Neutrons of more use to a moderating, thermal-neutron blanket are produced through the interaction of a proton with the nucleons in a nucleus in general, thereby leaving the nucleus in an excited state after interacting with the proton. Release of excess energy in the excited nucleus occurs by "boiling off" nucleons, and a substantial portion of these are neutrons. These neutrons are emitted isotropically with an average energy in the range 1-2 MeV. The number of neutrons generated by a given proton energy depends on the target material, as is illustrated in Fig. C-2.

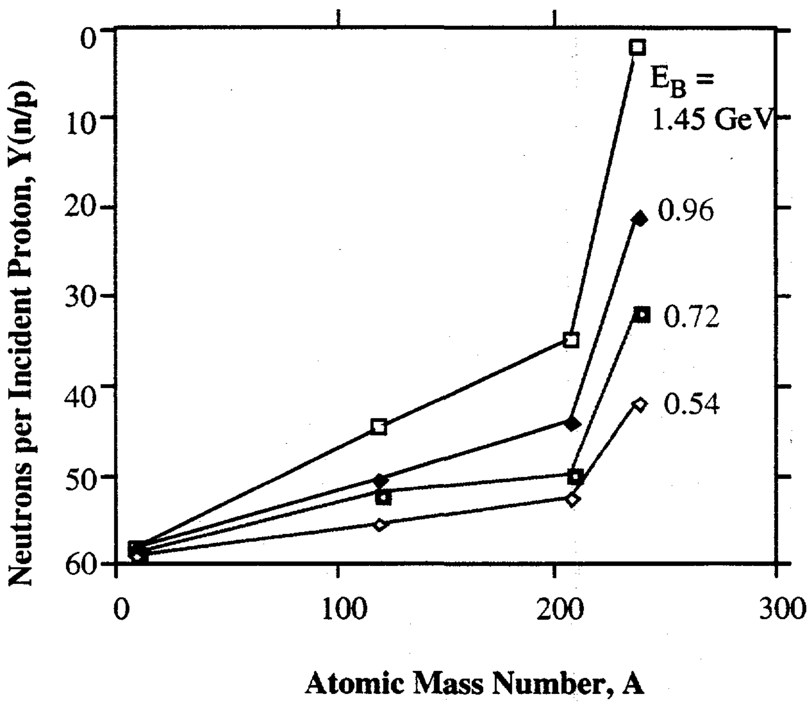

Figure C-2. Experimental neutron yields as a function of target atomic number.[31]

While maximizing the generation of low-energy neutrons is important to the overall efficiency of the ABC system, the ultimate target performance depends on the efficient transfer of these neutrons to the blanket. This efficiency depends on the neutron absorption characteristics of the target material(s) and the volumetric distribution over which the neutrons are generated (*i.e.*, whether the target produces a highly-peaked, intense neutron distribution; or whether the distribution is more evenly distributed; as well as the volume required to achieve maximum neutron production for a given neutron-source distribution). The target absorption characteristics depend not only on intrinsic nuclear parameters, but also on the amount of thermalization that occurs in the target, which is strongly dependent on the type and quantity of coolant used, as well as the target geometry and configuration. Similarly, the neutron-source distribution also depends on the target material (density), coolant fraction (*i.e.*, effective target density), and geometry. The ABC target design with the overall goal of achieving the required neutron fluxes in the blanket volume with a minimum accelerator beam power, therefore, will have to optimize neutron production, minimize neutron absorption in the target, and maximize neutron leakage to a blanket. Achieving these three goals also requires that the coolant fraction be minimized.

The ability of the ABC spallation target to achieve the functional goals described above, therefore, depends on three specifications: a) target material; b) target geometry; and c) target heat-removal system. All three specifications are interdependent, however, and this interrelationship must be fully understood before an effective and optimal ABC target design can be realized. Each general target specification is discussed below.

C.2. Target Design Issues

C.2.1. Target Material

Table C-I lists the materials that are the most likely candidates for use in high-energy spallation targets. The materials choice affects the ultimate system performance in four ways: a) the mass of the material affects the spallation neutron yield; b) the density affects the volume over which the neutrons are generated and the proton energy is deposited; c) the absorption cross section affects the amount of parasitic neutron absorption in the target; d) and thermal properties like the conductivity and the melting point affect the heat-removal requirements, which also impacts the obtainable neutron yield and the amount of parasitic absorption related thereto.

Table C-I. Potential Accelerator Target Materials[29,30]

Material	Z/A	σ_a[b] (barn)	Thermal Conductivity (W/m/K)	Melting Point (K)	Density (kg/m^3)	Range[c] (mm)
Ta	73/181	21.	54.	3,270.	16,600.	260
W	74/184	19.2	180.	3,380.	19,300.	225.
Pb	82/207	0.17	35.	600.	11,400.	389.
Bi	83/209	0.034	8.5	644.	9,800.	454.
Pb-Bi[b]	82.5/208	0.094	9.3[b]	398.	10,500.	423.
Th	90/232	7.4	41.	1,968.	11,700.	387.
U	92/238	7.59	25.	1,406.	18,900.	241.

[a] 55 w/o Bi
[b] 423 K
[c] E_B = 800 MeV

The spallation neutron yield is a linear function of the target-material atomic mass number until fission becomes significant, at which point the yield dramatically increases, as is shown in Fig. C-2. The data shown in Fig. C-2 were generated for pencil-shaped beams that impinged non-optimal target geometries, and, hence, an ABC target design will generate different yields than those shown. While boosting neutron production, the use of fissionable material in the target also dramatically increases the target power density and makes the target heat removal more difficult. The increase in coolant required to deal with the increased heat load limits the increase in neutron yield associated with the fission boost through increased parasitic absorption, depending upon the beam power density to which the target is exposed. In addition, fissioning and the attendant fission products increases the radiological hazards associated with target maintenance and disposal, as well as the hazards encountered under accident conditions

The remaining candidate target materials can be divided into two classes: a) high-temperature materials, Ta and W; and b) low-temperature materials, Pb, Bi, Pb-Bi alloy. The high-temperature materials have the advantage of being easier to cool, although they are large neutron absorbers compared to the low-temperature materials. Alternatively, if the low-temperature materials are used in solid forms, stringent cooling requirements and power-density limitations must be imposed. However, the low-temperature materials also can be used in a self-cooled liquid form and operate at higher target power densities and with improved performance because the elimination of external coolant and the potential for reduced (coolant) structure. The final material property that affects target performance is mass density. The density of the target material determines the proton range and, hence, the volume over which the neutron source is generated and the proton energy is deposited. Table C-I also summarizes the density of target materials, as well as the range of 800-MeV protons in each material.

C.2.2. Target Geometry

The geometrical configuration of the ABC target will affect the neutron-source distribution, the amount of parasitic absorption which occurs in the target, and the target cooling requirements. The target can either be designed as a monolith (with coolant channels if a solid material) or in a heterogeneous configuration, with the neutron producing materials separated into different regions within the target.

The neutron source in a solid target will be axially peaked at the front (beam side), as is shown in Fig. C-3; the degree of source peaking is dependent on the target material density. This peaking will maximize the peak neutron flux generated in the blanket, but it will also produce large flux gradients. Alternatively, if the target is heterogeneous in the axial direction, with the neutron-producing material spaced at appropriate intervals, the axialpeaking can be reduced. This approach, however, will create a more distributed source and a lower peak flux in the blanket, although the flux averaged over the entire blanket volume (assuming the source is not distributed too much) should remain approximately the same.

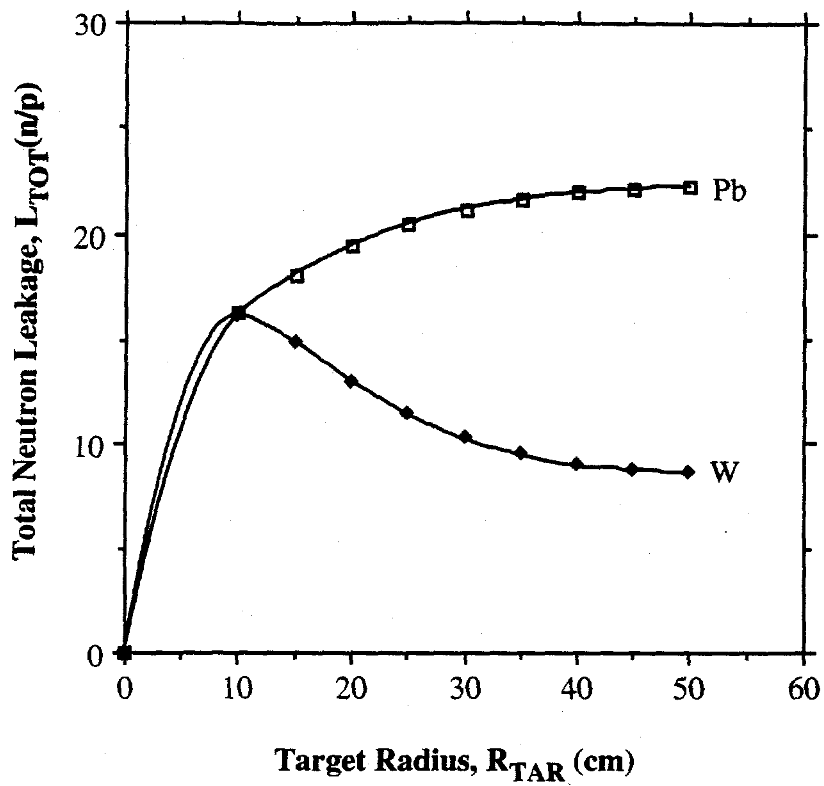

Figure C-3. Total neutron leakage from lead and tungsten targets as a function of radius.

A solid target will optimize at a smaller radius to offset absorption, although for low absorbing materials the change in optimal radius will be small. This effect is shown in Fig. C-4. The maximum source delivered from a monolithic tungsten target is less than a lead target even though the total spallation yield for each material is comparable. This is a result of the additional parasitic absorption that occurs in the tungsten. For a cylindrical blanket configuration larger peak fluxes near the inner blanket radius will result because of the smaller volume over which the neutron source is distributed. For highly absorbing materials, however, the overall neutron source delivered to the blanket will be lower than can be achieved in a heterogeneous system, and a solid target will have large adverse effects on the multiplication of the system compared to a heterogeneous design. The reduction in parasitic absorption for heterogeneous designs results from the creation of neutron leakage paths that allow neutrons to escape out of the target while traversing a minimum amount of absorbing material. Careful design is required, however, to assure that high-energy protons are not lost through the same leakage paths.

Finally, an axially heterogeneous target design allows greater flexibility with regard to the target cooling system because it allows the beam power to be distributed over a larger volume of material. A heterogeneous configuration, however, will be more complex that a solid target design, and for ABC this added complexity could impact target maintenance and replacement schemes.

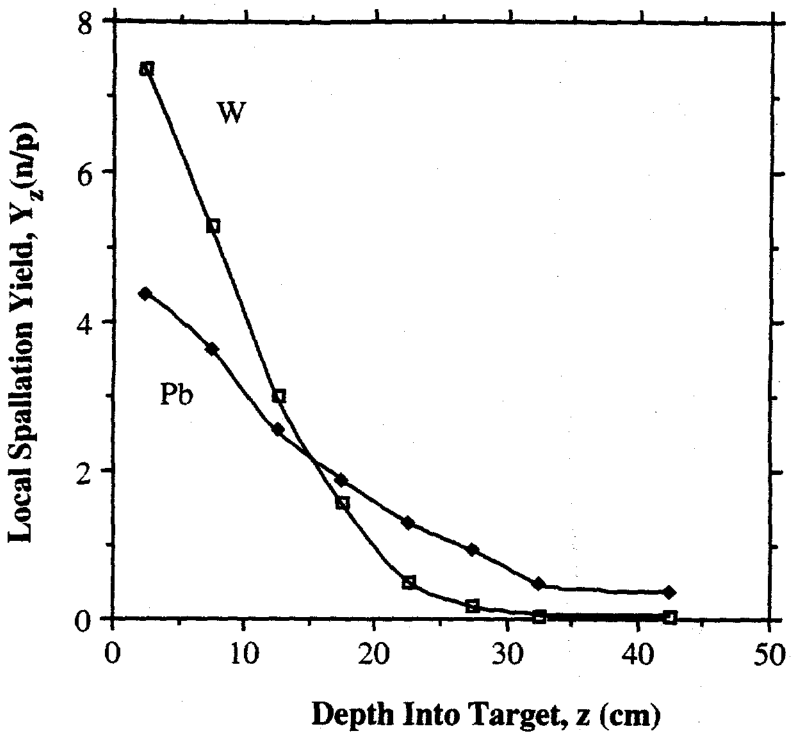

Figure C-4. Axial distribution of neutron in solid tungsten targets.

C.2.3. Target Heat Removal

Significant quantities of power will be generated in the target material, and, consequently, a target cooling system must be incorporated. The target heat-removal system transfers thermal energy from the target material to the ultimate heat sink. The target heat-removal system can be based on either the flow of an external coolant through the target material, or the flow of the target material itself through a heat exchanger.

In achieving the above-stated goal, the target heat-removal system must meet the following requirements: selection of the coolant material must be based on such factors as overall neutron economy, material compatibilities, pumping power demands, generation of both chemical and nuclear waste, and overall system safety. Potential target coolants include liquid metals, molten salt, D_2O, and inert gases. The final design of the integrated target system must take into consideration conventional engineering constraints related to thermal-hydraulics, erosion, corrosion, materials, structural, and other design, fabrication, and maintenance requirements. Peak target internal volumetric heating will be limited based on the integrated target system design using commonly accepted industrial practices. Sufficient margins and reliability levels must be incorporated into the design to assure safe operation throughout all appropriate circumstances, including normal operating conditions, anticipated operational occurrences, and design basis events; passive and/or active residual heat removal will be incorporated into the target cooling system.

C.3. Selection of Target Material

Two ABC spallation target design concepts have been selected for further investigation. The primary ABC spallation target concept utilizes flowing liquid lead as the target material and the primary coolant. This concept was selected because of high neutron production, source, capability of operating over a wide range of power densities (and, hence, is applicable to a wide range of blanket and accelerator designs), minimal effect on blanket multiplication because of its small parasitic absorption, and potential for a simple, monolithic configuration. Maintenance on this design is expected to be minimal, since the only material that must be replaced is the structural container; the lead may be drained during material replacement and reused throughout the lifetime of the plant.

The secondary ABC spallation target concept uses an array of solid thorium rods, clad with Hastelloy-N, and cooled by flowing molten salt. This concept has the advantages of high neutron source strength (although not as large as for the lead target) in a compact monolithic configuration, eliminating the need for a separate target cooling system by using the plutonium-bearing molten salt as the target coolant, and providing a target material disposal method, since ultimately the spent-thorium rods could be introduced into the blanket and transmuted, thereby reducing the ABC waste stream. The thorium target, however, has the disadvantages of being limited to ABC blanket systems of ~500 MW or less because of power-density constraints; a more complex fabrication process, is required because heterogeneity must be introduced into the target through tailored void regions within the rods; a mechanism for removal of the fission-product decay heat produced within the target is also required. The thorium target will not be discussed further in this document. A more detailed discussion of its design and associated issues can be found in Ref. 34.

C.4. Physics Optimization

A number of issues must be resolved in the design of the ABC flowing-lead spallation target, including: the size, configuration, and location of the target with respect to the blanket; selection of the structural material to contain the lead and its associated lifetime; removal of heat deposited both in the lead and the structural material containing it; as well as any issues associated with resolving the three listed (*e.g.*, pumping the lead, secondary coolant system, tailoring the flow to adequately cool the structure, *etc.*). An initial estimate of the required size of the target can be obtained through a simple parameterization of the neutron source emitted based on target size. Figure C-5 shows the neutron leakage from an infinitely-long (no bottom leakage) lead target as a function of target radius, and Fig. C-6 shows the neutron leakage of a 45-cm radius target as a function of target length. The total leakage curves from Figs. C-5 and C-6 would indicate an optimal target would be approximately 45 cm in radius and 60 cm in length. However, simply maximizing the number of neutrons emitted from the target is not adequate, because they only become useful if they enter the blanket region and are absorbed in the fuel. A requirement for bombarding the spallation target with protons is that the protons have an unobstructed pathway to the target surface; this unobstructed pathway also provides a leakage path for generated neutrons to escape without ever entering the blanket.

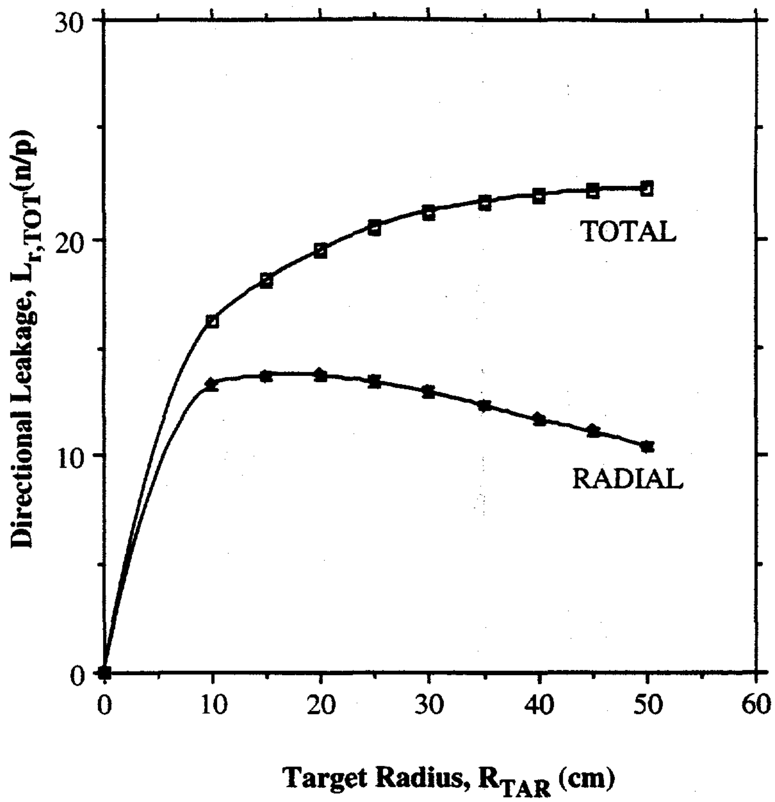

Figure C-5. Leakage from an infinitely long lead target (no leakage from end).

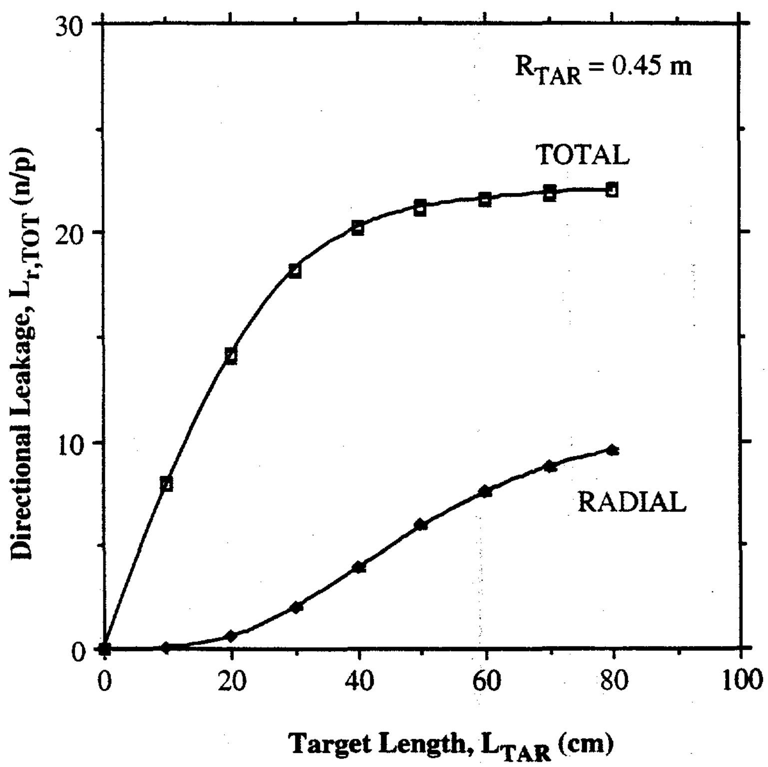

Figure C-6. Variation of neutron leakage with target length.

If the target were to be placed at the top of the blanket so that any neutrons leaking from the front (beam-side) face were lost, the target dimensions would optimize solely on the basis of radial leakage. The variation of radial leakage with target radius is also shown in Fig. C-5. Figure C-5 shows that optimizing solely on radial leakage results in a much smaller target radius. Figure C-6 also gives the radial leakage as a function of target length. The majority of leakage from a target of this radius (45 cm) is always in the vertical direction, and for targets of greater than ~15 cm in length, the main leakage path is through the front face of the target.

Conversely, if the target is placed at the bottom of the blanket so that any neutrons that leak from the back of the target are lost, as well as those which leak through the beam pipe, the optimization becomes more complex, since it is not just a function of target radius, but is dependent on the proton entrance length (distance from top of the blanket to the top of the target) as well. The optimization shown in Fig. C-7 indicates an optimal target radius of somewhere between 20 cm and 40 cm, with an entrance length of at least 75 cm. Finally, if the target is moved away from the bottom of the blanket (or made longer), so that neutrons emitted through the rear (of the 60-cm length modeled here) can be reflected back into the useful part of the blanket (or if transmutable material is located directly behind the target), the optimization changes, as is shown in Fig. C-8. In this case, the useful leakage increases with increasing target radius as long as a substantial entrance length is provided and the size is not increased to the point where absorption in the lead becomes significant. The optimal dimensions obtained from Fig. C-8 are approximately the same as those obtained from the total leakage values in Figs. C-5 and C-6. Finally, not only does changing target dimensions change the total neutron source emitted, but the distribution of the source as well. Figure C-9 shows the axial neutron source distributions for 25-cm and 45-cm radius targets. The target is located at an axial position of $0 < z < 60$ cm, and the radial leakage is determined through the cylinder encompassing the outer radius of each target. Fig. C-9 shows that a smaller target radius produces a much more axially peaked neutron source. For the fluid-fueled ABC system, neutron-flux peaking is much less of a problem than in solid-fueled systems, but peaking is still a concern with regard to possible materials damage issues.

Because the lead contained in the target will be a mixed (radioactive and toxic) waste stream, its volume should be minimized. Reduction in lead volume by changing the internal target geometry, however, causes a change in the total source, as is illustrated above. This change in source will then translate into change in accelerator current requirements, which ultimately translates into cost of operation. Figure. C-10 shows an optimization calculation for lead mass *versus* operating (power) costs of the accelerator. It should be noted that the optimization shown in Fig. C-10 is strongly design dependent. In this case, the lead mass is being altered by changing the overall target radius with a set entrance length. The changing beam radius effects to total neutron yield due to changes in proton leakage from the target. This optimization will change, depending on the variables being altered. The main point to be made, however, is that this type of optimization will be required for any specific target design, and arbitrary target design changes can not be made without first determining the ultimate effects they will have on the system operation.

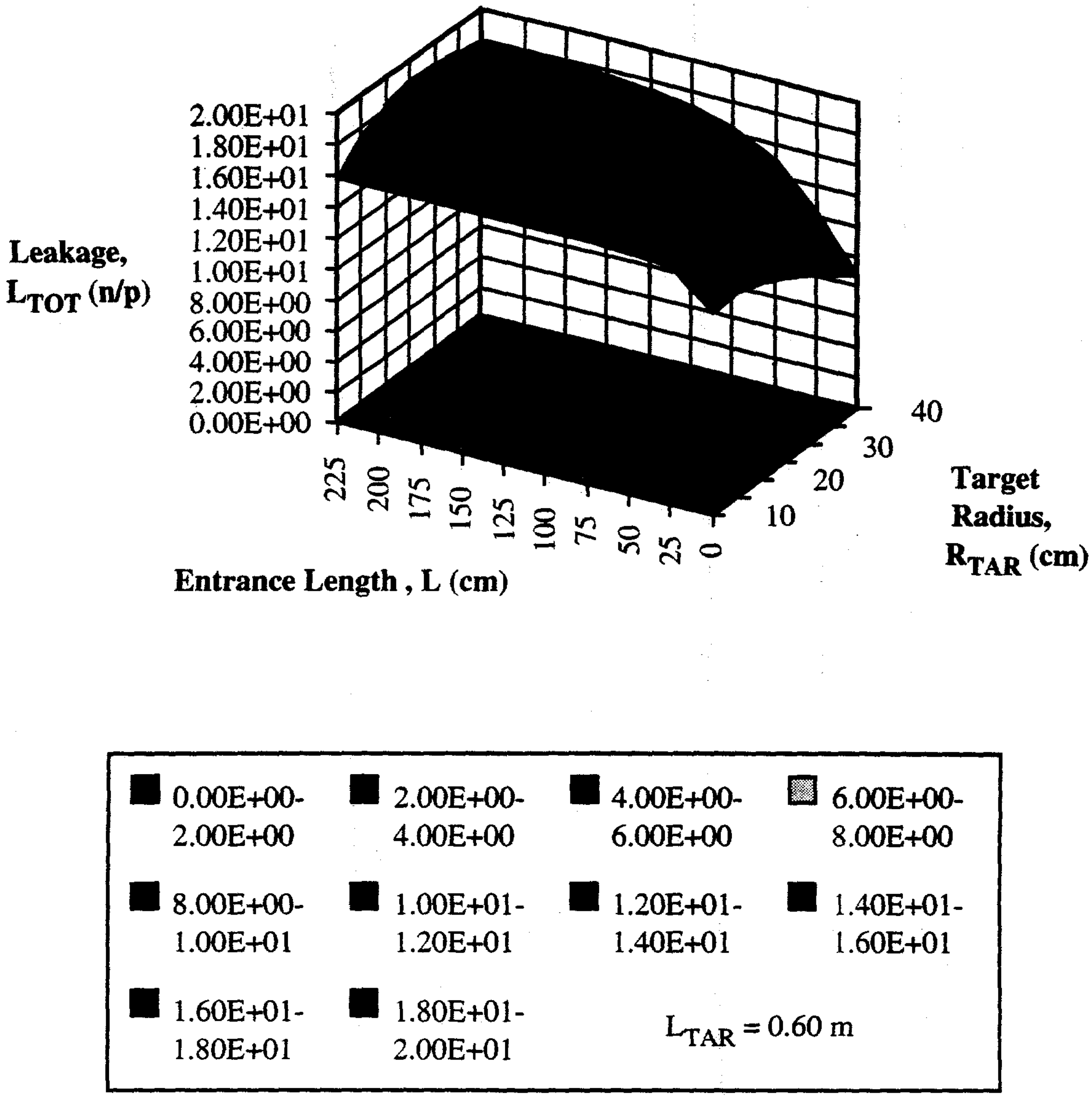

Figure C-7. Parametric variation in radial leakage for a 0.60-m-long lead target.

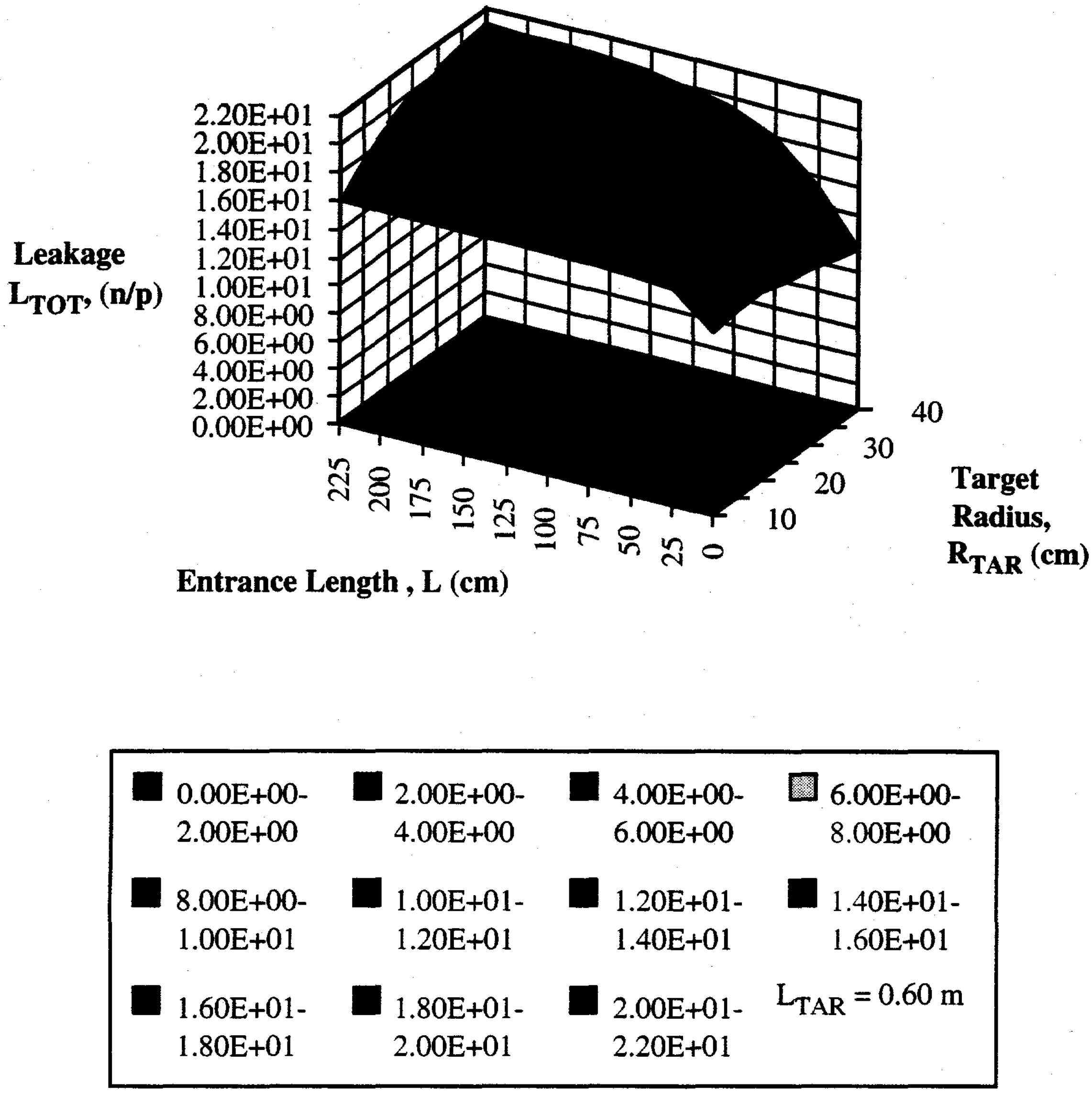

Figure C-8. Parametric variation in radial plus end leakage for a 0.60-m-long lead target.

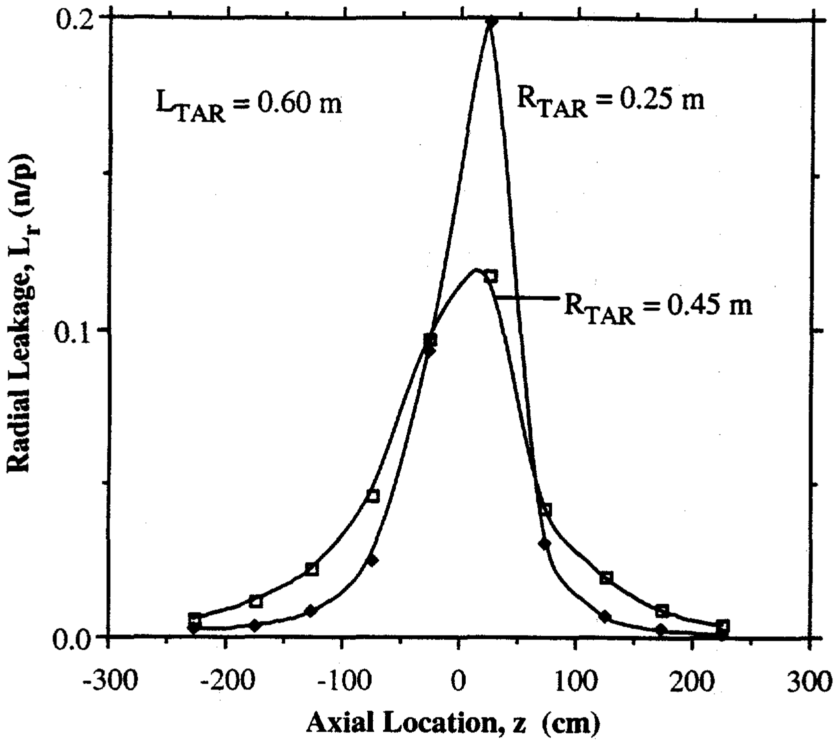

Figure C-9. Variation of radial leakage distribution with target radius.

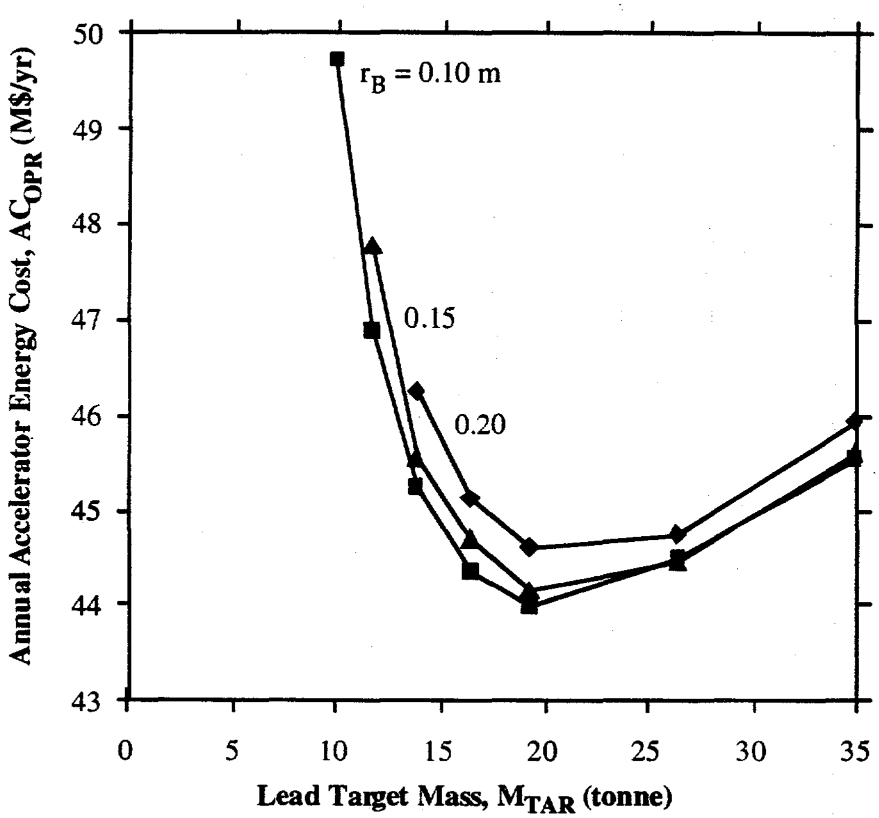

Figure C-10. Design-specific optimization of lead mass.

In the present preconceptual target design for ABC, molten-lead is circulated through an active spallation region, where it picks up sensible heat that is later transferred to a secondary coolant. This design is unique in that the beam window is an integral part of the target containment structure and is convectively cooled by the flowing lead. Consequently, the structural container for the molten-lead will be exposed to a significant flux of high-energy protons and neutrons as well as a corrosive environment. Selection of the container material will greatly effect the lifetime and safety of the target subsystem; the material should have good compatibility with molten-lead, sufficient mechanical strength at operating temperatures, a low neutron absorption cross section, and good performance under an intense radiation environment.

In selecting a structural material for a liquid-metal application, a number of important factors must be considered, including: compatibility of the structural material with the liquid metal; the thermal and radiation environments; and the physical geometry of the container. In the ABC system, operational conditions are similar to those encountered in nuclear reactor systems. The exception is the spallation target region, where a high-energy proton and neutron flux exists. Consequently, the large corporate knowledge base of liquid-metal reactor systems is applicable, but not conclusive. For the particular application of a circulating molten-lead spallation target, three factors have been used to rank the applicability of a wide range of materials for structural containment.

One of the major factors limiting the design and lifetime of most high-temperature liquid-metal engineering systems is liquid metal corrosion. A number of corrosion mechanisms is liquid metals can be identified: dissolution attack; temperature gradient mass transfer; concentration gradient mass transfer; impurity reactions; erosion-corrosion; intergranular attack; alloying between the liquid metal and the solid containment metal; self-welding of solid metals; and liquid-metal embrittlement. Of these mechanisms, temperature-gradient mass transfer is usually the most damaging in applications where large temperature gradients exist. The solubility of the container material in the liquid metal is a function of temperature. After a period of operation, solubility limits are reached and the temperature dependent solubility results in the transfer of the container material from the hottest location to the coldest location in the flowing loop. The effect in the circulating molten-lead target is to dissolve container material from the window and deposit it in the heat exchanger. Both the thinning of the window and potential fouling of the heat exchanger are undesirable.

The relative resistance to mass transfer in liquid lead of 24 metals and alloys were measured by Cathcart and Manly[33], and the results are summarized in Table C-II. The materials tested were divided into three groups based on their relative resistance to mass transfer in a thermal convection loop test; the "heavy mass transfer" group included nickel, titanium, cobalt, chromium, iron, beryllium, Inconel, 304 Stainless Steel, and 310 Stainless Steel, the "usually little mass transfer" group included Hastelloy-N, 410 Stainless Steel, and 446 Stainless Steel, and the "no mass transfer" group included niobium and molybdenum. In an ORNL thermal convection loop test, Croloy 2-1/4 (Fe-2.25Cr-1Mo) exhibited 25-200 μm of attack at 593-654°C, and Nb-1Zr showed no attack after 5,280 hours exposure to lead at 760°C[28]. In conjunction with the LMFR program, BNL also ran a Croloy 2-1/4 loop containing Zr-inhibited lead for over 27,000 hr at a 550°C hot leg temperature and a 118 K temperature difference; no significant corrosion was observed.

Table C-II. Properties of Selected Materials.

Material.	Density (kg/m^3)	T_m (°C)	ACS barn	Mass Transfer	Attack Resistence	UTS (RT) (MPa)	UTS(500) (MPa)	YS(500) (MPa)
Be	1.85	1,283	0.009	heavy	good	228-690	145-170	
Mg	1.74	650	0.063			90-220		
Al	2.70	660	0.215			40-140		
Ti	4.51	1,668	5.6	heavy	limited	220	100	
V	6.09	1,735	5.1			472-911		
Cr	7.19	1,890	2.99	heavy		413	225-242	
Mn	7.44	1,245	13.2		poor	496		
Fe	7.87	1,539	2.53	heavy	good			
Co	8.89	1,495	37	heavy		234-945		
Ni	8.91	1,455	4.6	heavy	poor	317		
Cu	8.92	1,083	3.69		poor	209-344		
Zr	6.51	1,845	0.18		limited			
Nb	8.57	2,415	1.1	no	good	210-334	20-290	90-120
Mo	10.22	2,610	2.5	no	good	400-600	250-350	
Ru	12.48	2,334	2.46					
Rh	12.44	1,966	150			951-2,068		
Hf	13.09	2,222	115					
Ta	16.68	2,996	21.3		good	250-400	200-300	
W	19.26	3,410	19.2		good	560-3,922		196-1,667
Os	22.6	2,700	14.7					
Ir	22.5	2,454	430			990-2,480		530
Pt	21.45	1,769	8.1		poor			
Th	11.66	1,750	7			150-250	50-80	
304 SS	8.0	1,425		heavy	poor	515-2,240		
304LSS	8.0	1,425			poor	480-620		
310SS				heavy				190
316SS	8.0	1,390			poor	515-1,690		
316LSS	8.0	1,390			poor	480-620		
347SS	8.0	1,412		heavy				
410SS				little		780	450	380
446SS				little			280	
Inconel	8.5	1,410		heavy		697	650	
Inconel	8.19	718				995-1,448	990-1,200	
Hastelloy-B	9.24	1,335			little		583	531
Hastelloy-N	8.89						724	586
Zircalloy-2		1,817	0.193				570	
Zircalloy-4		1,817	0.194					
Nb-1Zr	8.4	2,467				280	200	130
PWC-11	8.4					345	262	150
T-111	16.7	2,977				690	379	
Mo-13Re	10.9	2,537				550	530	
TZM	10.2					552-883		
Croloy 2-1/4		1,530				680	470	220
HT-9		1,520				550-1,500		500-1,100

Table C-II. Properties of Selected Materials. (Cont.-1)

Density = density at 25°C in kg/m3
T_m = melting point in °C
ACS = absorption cross section for 2,200 m/s neutrons in barns/atom
Mass transfer = mass transfer in liquid lead from J. V. Cathcart and W. D. Manly[33]
Resistance to attack = resistance to attack by liquid lead at 600°C from Liquid-Metals Handbook[36]
Good = rate of attack is less than 25 µm/yr; limited = rate of attack is 25-50 µm/yr, poor = rate of attack is greater than 250 µm/yr
UTS(RT) = ultimate tensile strength at room temperature
UTS(500) = ultimate tensile strength at 500°C
YS(500) = 0.2% yield strength at 500°C

304SS: Fe-(18-20)Cr-(8-12)Ni-2Mn-1Si-0.08C-0.045P-0.03S
304LSS: Fe-(18-20)Cr-(8-12)Ni-2Mn-1Si-0.03C-0.045P-0.03S
310SS: Fe-(24-26)Cr-(19-22)Ni-2Mn-1.5Si-0.25C-0.045P-0.03S
316SS: Fe-(16-18)Cr-(10-14)Ni-2Mn-1Si-0.08C-0.045P-0.03S-(2-3)Mo
316LSS: Fe-(16-18)Cr-(10-14)Ni-2Mn-1Si-0.03C-0.045P-0.03S-(2-3)Mo
347SS: Fe-(17-19)Cr-(9-13)Ni-2Mn-1Si-0.08C-0.045P-0.03S-10 x: % C min Nb + Ta
410SS: Fe-(11.5-13)Cr-1Mn-1Si-0.15C-0.04P-0.03S
446SS: Fe-(23-27)Cr-1.5Mn-1Si-0.2C-0.04P-0.03S
Inconel : 72Ni-15.5Cr-8Fe-1Mn-0.5Si-0.15C
Inconel-718 : 52.5Ni-19Cr-18Fe-5.2Nb-3.0Mo-1.0Co
Hastelloy B : 67Ni-28Mo-5Fe
Hastelloy N : 68Ni-17Mo-7Cr-5Fe
Zircalloy-2 : Zr-1.5Sn-0.14Fe-0.1Cr-0.06Ni
Zircalloy-3 : Zr-0.25Sn-0.25Fe-0.05Cr-0.05Ni
Zircalloy-4 : Zr-1.5Sn-0.17Fe-0.12Cr
T-111 : Ta-8W-2Hf
PWC-11 : Nb-1Zr-0.06C
TZM : Mo-0.5Ti-0.1Zr
Croloy 2-1/4 : Fe-2.25Cr-1Mo-0.15C
HT-9 : Fe-12Cr-1Mo-0.5Ni-0.5W-0.3V-0.2C

Dissolution attack is another important consideration for liquid metal corrosion. From the results of a static corrosion test, the evaluation of materials according to their resistance to attack by liquid lead has been studied,[35] and is summarized in Table C-II. At 600°C, the materials with good resistance included niobium, molybdenum, tantalum, tungsten, beryllium, iron, mild carbon steel, low-chromium steel, and ferritic stainless steels. Aluminum, titanium, and zirconium exhibited limited resistance. Austenitic stainless steels, copper-base alloys, nickel, and nickel-base alloys showed poor resistance.

A sufficient mechanical strength at service temperatures is another factor in the selection of structural materials for the circulating molten-lead target. Preliminary design inlet and outlet temperatures of the flowing lead are 500 and 400°C, respectively. The ultimate tensile strength at room temperature, ultimate tensile strength at 500°C, and 0.2% yield strength at 500°C of selected candidate materials are shown in Table C-II. The strength of a material, however, depends on many factors, with processing history and impurity levels being particularly important. The mechanical strengths shown in Table C-II should not be regarded as absolute. In the preliminary target design, the maximum hoop stress in the container has been calculated to be approximately 28 MPa at a lower temperature region (400°C). At this high stress location, aluminum-base alloys will lose most of their strength because of their low melting point (around 660°C). As is shown in Table C-II, other common structural materials such as stainless steels, nickel-base alloys, zirconium-base alloys, refractory alloys, and iron-base heat resistance alloys (for example Croloy 2-1/4 and HT-9) will have enough strength at this location.

Also of importance is the strength of the material at the highest temperature location. In the present ABC design, it is estimated that the maximum temperature of the system will be approximately 900-1000°C at the point where the proton beam impinges the window. At this temperature, iron-base alloys are inadequate, because they have melting points in the range 1,400-1,500°C and creep becomes significant at 600-650°C [half of the (absolute) melting point of the material]. Iron-base heat resistant alloys, such as Croloy 2-1/4 and HT-9, are usually limited up to 650-700°C. Similarly, nickel-base and cobalt-base superalloys are marginally acceptable because they are limited up to 900-1000°C. Another consideration for the use of these superalloys are that nickel is incompatible with liquid lead and cobalt has a high neutron absorption cross section (37 barn). Refractory metals (Nb, Ta, Mo, W) are usually considered at service temperatures above 900°C.

The candidate structural material should also have a low thermal neutron absorption cross section to offer maximum neutron yield. The neutron absorption cross section of selected materials are tabulated in Table C-II. Beryllium has the lowest absorption cross section (0.009 barns), however, it exhibited heavy mass transfer in liquid lead.[33] Aluminum has a good absorption cross section (0.215 barns), however, it has a low melting point of 660°C. Zirconium has a low absorption cross section (0.18 barns), as does niobium (1.1 barns), iron (2.53 barns), and molybdenum (2.5 barns). Tantalum (21.3 barns), rhenium (84 barns), and tungsten (19.2 barns) have relatively high absorption cross sections.

In summary, Nb-1Zr has been selected for use as the structural container for the molten-lead target. It has been shown to be compatible with molten lead, it has sufficient mechanical strength at operating temperatures up to 1000°C, and it has a low neutron absorption cross section. Because of these same attributes, Nb-1Zr was chosen as the primary structural material for the SP-100 space nuclear power system. It is readily available in all product forms from several commercial sources and has a significant data base for fission irradiated performance. Molybdenum has a little higher mechanical strength and neutron absorption cross section than niobium, and has good compatibility with liquid lead. Consequently, Mo is a good backup material. Iron showed heavy mass transfer in liquid lead, and iron-base alloys do not have sufficient strength for beam entry window at 1000°C. Although Croloy 2-1/4 and HT-9 are compatible with molten lead, their low strength at 1000°C make them unattractive as a window material. Zirconium has some attractive properties such as a very low neutron absorption cross section and relatively high melting point (1845°C), however, its mass transfer behavior in lead is not known. Beryllium has very good absorption cross section, but it exhibited heavy mass transfer. Nickel and nickel-base alloys can not be used because of their heavy mass transfer and poor resistance to attack.

C.5. Target Waste Generation

The use of flowing lead as the primary target material minimizes the waste stream from the target because only the structural material will require replacement on a regular basis as a result of radiation damage; the lead can be drained an reused throughout the lifetime of the plant. The level of activation of the structural material as well as the required frequency of replacement will depend upon the final material selected as well as the radiation environment to which it is exposed. Hence, little can be said of this waste stream at this time; a full blanket design will be required before this issue can be resolved with any confidence. More information is available for the spallation products which will be produced in the lead, however. The spallation process produces an extensive number of nuclides which are not normally produced in fission reactors. In addition, the distribution of nuclides produced is substantially different than that produced through fission. The spallation product distribution produced by bombardment of lead with 800-MeV protons in shown in Fig. C-11. The spallation process produces nuclides which span the entire range of elements up to the mass (and even slightly greater) of the target material. The greatest production occurs near the target-material mass, because of only a few nucleons are ejected in any single reaction. Hence, the nuclides produced in this peak will typically be dominant contributors to the waste stream, and these nuclides depend upon the target material. Unfortunately, in lead a number of long-lived isotopes (^{194}Hg, ^{202}Pb, ^{205}Pb) are produced in this peak production region. A second peak occurs in the intermediate-mass nuclides due to the generation of high-energy fission. The waste characteristics of these nuclides are similar to fission products; however, the yields of these materials are relatively low. The final waste stream will depend upon the accelerator characteristics and the lifetime of the plant, but a highly radioactive material with a long-lived component can be expected.

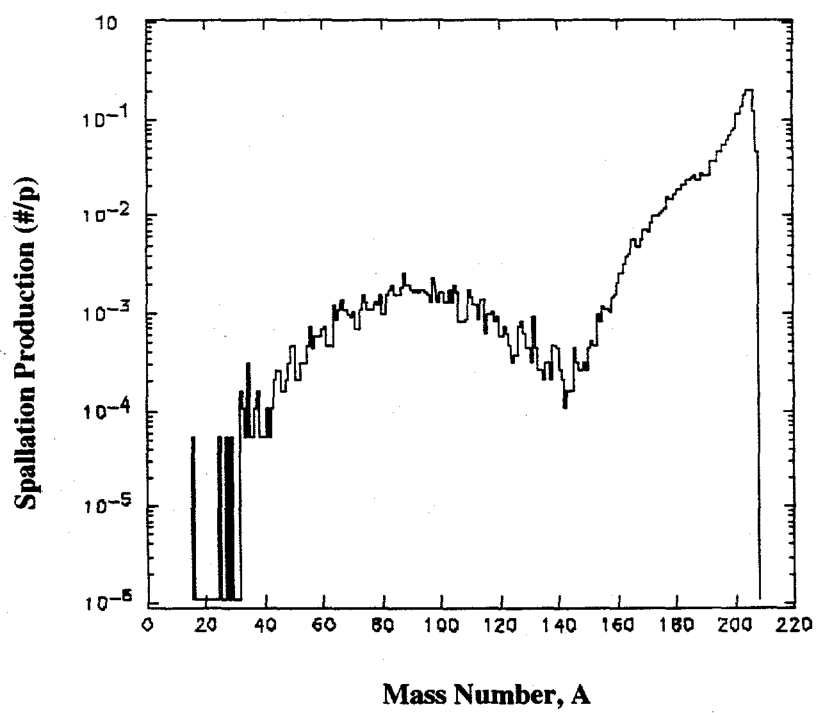

Figure C-11. Spallation product distribution produced by 800-MeV protons impinging on a lead target.

C.6. Conclusions

A flowing-lead target has been selected as the primary focus for the spallation target design effort for the ABC system. This material offers the advantages of producing an intense neutron source with little parasitic absorption, minimum waste generation through reuse of the primary target material, and the ability to operate over a wide range of system and accelerator power levels. Careful design is required, though, to effectively couple this target with both the accelerator and highly-multiplying ABC blanket. Efforts are ongoing in this regard in addition to dealing with the engineering issues associated with operating a liquid-metal-based system.

Appendix D. Directly Cooled Target *versus* Separately Cooled Window

Whether the window interface between target and blanket should be directly cooled by the flowing lead, or whether a separate coolant should be used is uncertain at this stage in the ABC target design. The addition of a separate coolant loop means that additional structure will be required in the target region, as well as introducing additional failure modes because of the added complexity. The peak window temperature, however, can be reduced, which in turn reduces the requirements placed on the selection of material. This approach, however, does not reduce the structural temperature in other (non-window) target materials, where temperatures are expected to remain in the 500-600°C range; these temperatures are above limits that are allowable for most common structural materials, such as stainless steels.

Selection of the window coolant becomes an important consideration for this option. Preferably, the coolant should be liquid at room temperature (to avoid the requirement for an additional thaw/freeze system, as is required for the lead); require minimal pressurization to avoid boiling; and be chemically compatible with the lead in case of inadvertent mixing. Whatever liquid is used as the secondary coolant for the lead would most likely be used also as the window coolant (*e.g.*, NaK, industrial salt, or molten salt), although temperature limits become a considerations. This design requires that the corrosion behavior of this liquid on the structural material be acceptable also.

Additional suggestions have been made for configurations that separate the window from the lead with an inert gas region, or for eliminating the window altogether and relying on the low vapor pressure of lead (along with strategically placed "cold fingers") to minimize contamination of the accelerator tube and vacuum system. Introduction of a free surface into a flowing system, as is required for either of these alternatives, invariably leads to flow instabilities and increases the technical difficulty in producing an acceptable design. While not theoretically infeasible, it does not appear that the added difficulty of design and potential complexity of operation are worth the benefits gained by removing the window.

Appendix E. Constant-k_{eff} Fuel-Cycle and dpa Analyses for the Plutonium-Burning ABC Concept Based on the Graphite-Moderated Molten-Salt Blanket

E.1. Introduction

A calculational methodology that couples the transport code MCNP[37] with the isotope depletion code ORIGEN2[38] using a modified Euler method is described in Ref. 7, and the results of applying this methodology to the ABC Target-Blanket system are reported in this Appendix. The method allows for depletion calculations for very high burnup systems where large spectral changes occur over time; the method used is more accurate than the simple Euler method[7]. The rate of increase in plutonium inventory required of a constant-k_{eff} ABC operating scenario is estimated. This kind of operating mode for ABC would vary the plutonium loading to assure a constant power output for a constant accelerator capacity over the life of the plant. Also reported in this appendix are the results of using MCNP to estimate the impact of (graphite) moderator configuration on the graphite neutron damage rate and, hence, lifetime.

E.2. Model

To quantify the burnup of weapons plutonium in ABC, Accelerator Transmutation of Waste (ATW)[2,19,20] and Accelerator Driven Energy Production (ADEP)[2,15] systems, it is necessary to perform depletion calculations under conditions where the neutron energy spectrum in the system is strongly time-dependent. This situation is in contrast with a hypothetical system where, without spectral changes with time, the equations governing the buildup, consumption and decay of radionuclides would be linear.

The depletion analysis code ORIGEN2[38] is exact for such a linear system. In general, the code solves N coupled equations governing the rate at which the amount of nuclei changes as a function of time, where N is the total number of species present in the system. For each species the time-dependence of the atom density is determined by sources from radioactive decay, neutron absorption, and input from an external feed; likewise, losses are modeled in terms of radioactive decay, neutron absorption, and chemical processing. If all the coefficients in this rate equation for a given species are independent of the concentration for that species, the rate equation is linear. Also, if a system operates at a constant specific power, and all cross section *ratios* were constant, the system would remain linear

Cross section ratios in general, however, change with time. The dependence of an average cross section on concentration is unique to each system and must be found through some explicit neutron transport calculation. One code performs a fully-coupled depletion/transport calculation, but this code has only one spatial dimension and uses only a fixed 26-group structure[39]. Presently, three-dimensional transport codes that also perform fully coupled depletion calculations are not available.

In order to solve this problem in approximate fashion, the codes MCNP4a[37] and ORIGEN2 have been self-consistently coupled in a scheme called the Modified Euler Depletion Analysis Loop (MEDAL).[7] The code MCNP4a is a three-dimensional Monte Carlo neutron and photon transport code that uses continuous energy-dependent cross section sets derived from the Evaluated Nuclear Data File (ENDF/B-V)[40] or uses the

Evaluated Nuclear Data Library (ENDL)[41]. The "Unmodified Euler method" refers to a method for solving differential equations by evaluating any non-linear coefficients at the beginning of a time step only, with the subsequent use of an explicit forward time step[42]. In the present application[7], MCNP and ORIGEN are coupled in a simple 1-2-1-2... repetitive sequence, as is done in the code package MOCUP[43]. Two auxiliary Fortran codes used in the MEDAL scheme are called KOLAPS and TDI. Under the Modified Euler method, the one-group cross sections are first evaluated at the beginning of a given time step. The code KOLAPS writes these cross sections to a library that is used by ORIGEN to perform a temporary forward time step calculation. The inventory is extracted with the code TDI from the ORIGEN output file. This input is then written to an MCNP input file. The neutron spectrum is calculated again to create another set of one-group cross sections. At this point in the calculation, the code KOLAPS reads both these MCNP results and the results at the beginning of the time step and then performs an average. These average values are used to perform the forward time step that is actually saved and used by the routine. The temporary forward-time-step results are discarded. It is believed that this procedure is more reliable under more circumstances the Unmodified Euler Method per unit of computer time used.

As noted above, the coupling between MCNP and ORIGEN2 uses two main auxiliary codes, KOLAPS and TDI. First MCNP computes effective one-group cross section evaluations for the 34 actinide species. These cross sections are stored as tallies in ae file that is is read in by the code KOLAPS. This code also has its own 124-group library of fission product cross sections. A separate tally in MCNP creates a 124-group energy spectrum, which is collapsed against the library to produce individual one-group cross sections. The fission product and actinide cross sections are then written to an ORIGEN library.

As the code ORIGEN2 is run, plutonium feed is added at a rate that holds k_{inf} fixed at a predetermined value. The concentration updating occurs once every internal time step, which is fixed at 10 days. After a certain number of these internal time steps, called "recycles", the calculations are stopped and the concentrations are fed into MCNP for another crossection evaluation The MCNP input represents as accurately as possible the composition computed by ORIGEN. This matching is done by the code TDI, which reads the 34 actinide species from the ORIGEN output. In addition, 62 fission product species are read from the ORIGEN output file and is similarly treated. These species, which are those for which MCNP has continuous cross section evaluations, account typically for 85% of all fission product captures in the system. The remaining 15% are accounted by an increase in the concentration of the other species. For the first MCNP input file the concentrations are read in from an auxiliary input file.

The code TDI has routines to compute material compositions and some geometry models, so that a complete MCNP input file is written by simply reading a few lines of input. The materials and geometries are those commonly found in ABC designs. An adjustable hexagonal unit cell geometry is provided in the code TDI. The code can also simulate a full target/blanket assembly using a variety of materials, as is described in the following subsection.

In Ref. 7, unmodified and modified Euler approximations to differential equations that have known solutions are evaluated. Computational comparisons are then given for ABC using the methods described above; only the ABC results that use graphite moderation in the MSBR-like configuration[10] are reported in this appendix.

E.3. Results

E.3.1. Case Without Graphite Moderation

In the computational example reported here the cell volume is composed of 100% fuel salt, is composed in turn of 1/3 LiF and 2/3 BeF_2. The initial loading of weapon Pu is 0.032 mole/liter along with and initial loading of ^{151}Eu neutron poison at $3x10^{-5}$ atom fraction. Figure E-1 shows the results of the ^{239}Pu calculations for 1- and 2-yr time steps for the modified and unmodified computational methods[7]. Because an exact solution is not available, the results of a modified Euler calculation with 1/4-yr between spectral updates is used as a benchmark. When this model is used, discontinuities are observed in the concentration of the feed species with time. The concentration versus time in Fig E-1 is concave downward because of the time-dependence of the fission-product composition. When the spectrum is updated, ORIGEN may determine that the system is above the specified k_{eff} and will temporarily halt the plutonium feed. A negative slope results.

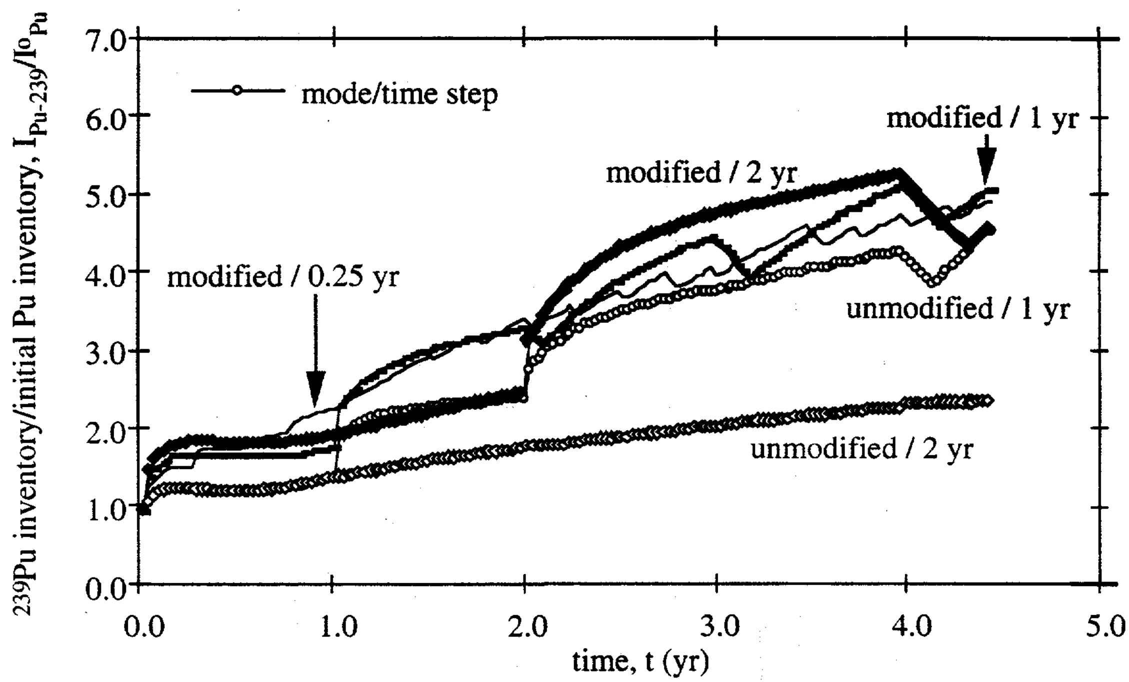

Figure E-1. Time dependence of ^{239}Pu inventory in a homogeneous unit cell calculation using the modified and unmodified[7] Euler methods to couple MCNP4 and ORIGEN2.1. Because an exact solution is not available for this case, a "modified" calculation with a three-month period between (neutron) spectral updates was taken as a benchmark comparison. Similar behavior to the analytical model[7] is seen, where the unmodified solution lags the benchmark and, the modified methods oscillates about the benchmark.

This behavior is similar to that reported for idealized models in Ref 7: the modified method produces solutions that oscillate about the benchmark, while the unmodified method tends to lag monotonically. The departure from the true solution, which is presumably most closely approximated by the modified method with a 1/4-yr time step, is minimized by the modified method with the 2-yr time step compared to the unmodified method with the 1 year time step. Because the case considered here has no graphite, the system is always somewhat epithermal. The system moves farther into the resonance region after a short time. Of the ABC systems examined, this case exhibits the greatest departure from linearity. The early time behavior of this system has been shown because this is where the greatest departures occur. For times greater than 5-yr, the modified Euler scheme using 1 or 2- yrs between spectrum updates or the unmodified scheme with 1-yr or less between updates produced adequate results.

E.3.2. Target/Blanket Depletion Calculations for ABC

As noted above, the code TDI is capable of writing full target-blanket MCNP input files with a few input lines. This capability is useful for performing scoping calculations. In this subsection are presented the results from three full target-blanket simulations. In all cases the actinide burn rate is chosen to generate a total fission power of 711 MW. The volume of fuel salt in the blanket is 7.9 m^3, and an equal amount of fuel salt is held external to the blanket in heat exchangers, pumps, and related piping. The in-blanket power density in the salt, therefore, is 90 MW/m^3, which is near that in the MSBR and is adopted for the ABC base case.

Two designs employing heterogeneous lattices and a third, homogeneous design were investigated. These cases are illustrated in Figure E-2. The heterogeneous lattice refers to a configuration where a portion of the blanket consists of blocks of graphite with circular coolant holes drilled in a regular pattern. The fuel salt flows through these holes between an upper and lower plenum. The holes are 2.4 cm in diameter in one design and 5.2 cm in the other. The pitch between holes is 6.5 cm for both cases. The fuel-salt volume fractions in the lattice are 12.5% and 51% for the two cases, respectively. The homogeneous design did not use any graphite internal to where the fuel salt is held in the blanket. All three systems are heavily reflected by graphite. The target in the models consists of a cylinder of lead 60 cm in diameter and 80 cm in length. The initial concentration of weapon-return plutonium is 0.01 molar (~3.0 kg/m^3 actinide fluoride) in all three cases. A small amount of ^{151}Eu is present initially as a burnable poison to fine-tune the initial k_{eff} to 0.95. The k_{eff} is held constant at this value throughout the life of the system by adjusting the plutonium feed.

One goal in this investigation is to estimate system performance in terms of burnup of weapon-return plutonium, with burnup being defined as the ratio of the number of moles fissioned to the sum of the moles fissioned and held internal to the system. Another goal is to estimate the rate of graphite consumption from radiation damage. A tally has been introduced into the MCNP input file that computes the carbon displacements per atom (dpa) in the graphite moderator per year of operation. On the basis of computational results discussed earlier, a time step of one year between spectral updates, using the MEDAL scheme, was judged as sufficiently accurate.

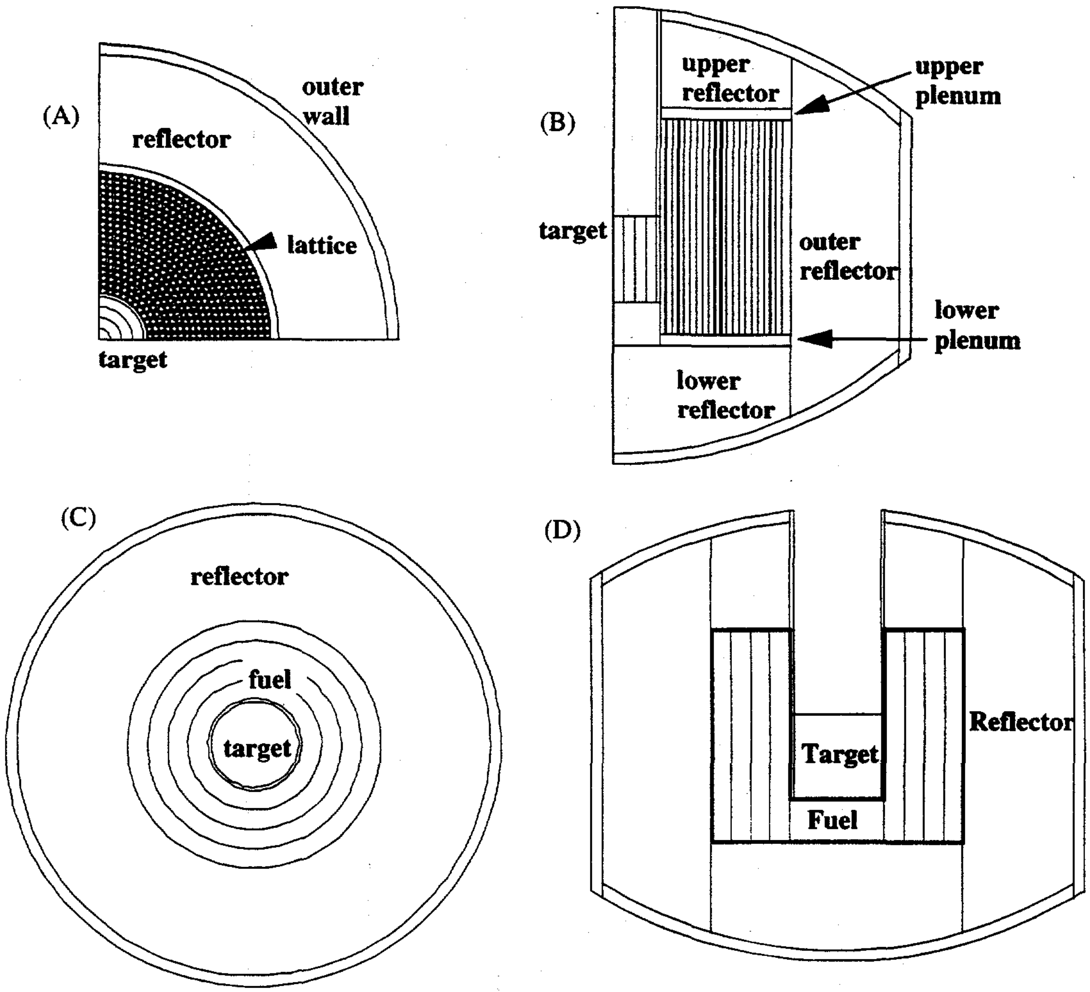

Figure E-2. Target-Blanket models used in coupled MCNP/ORIGEN2 calculations of constant-k_∞ depletion and dpa results; the fuel region is divided into five radial zones.

A. top view of heterogeneous case
B. side view of heterogeneous case
C. top view of homogeneous case
D. side view of homogeneous case

Figures E-3 and E-4 show , respectively, the time-dependent ^{239}Pu inventory and the capture and fission cross sections for this species in the homogeneous design. The inventory in the homogeneous system is the highest, followed by the heterogeneous design having a 51% fuel-salt volume fraction. The one-group fission cross section for ^{239}Pu begins at ~170 barns, and decreases rapidly to ~20 barns after 12 years. These values are to be compared to values in a PWR and CANDU spectrum of typically 110 and 240 barns, respectively. The capture-to-fission ratio, α, increases by 12% over the 12-yr simulated in these computations.

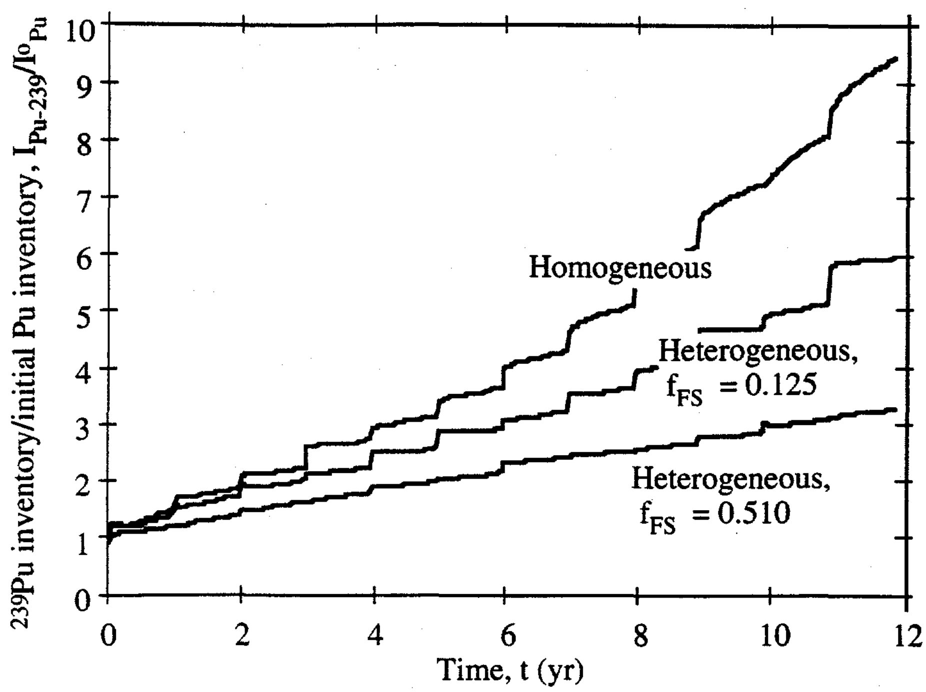

Figure E-3. Time-dependence of ^{239}Pu inventory for three difference blanket models; plutonium feed is adjusted to maintain $k_\infty = 0.95$.

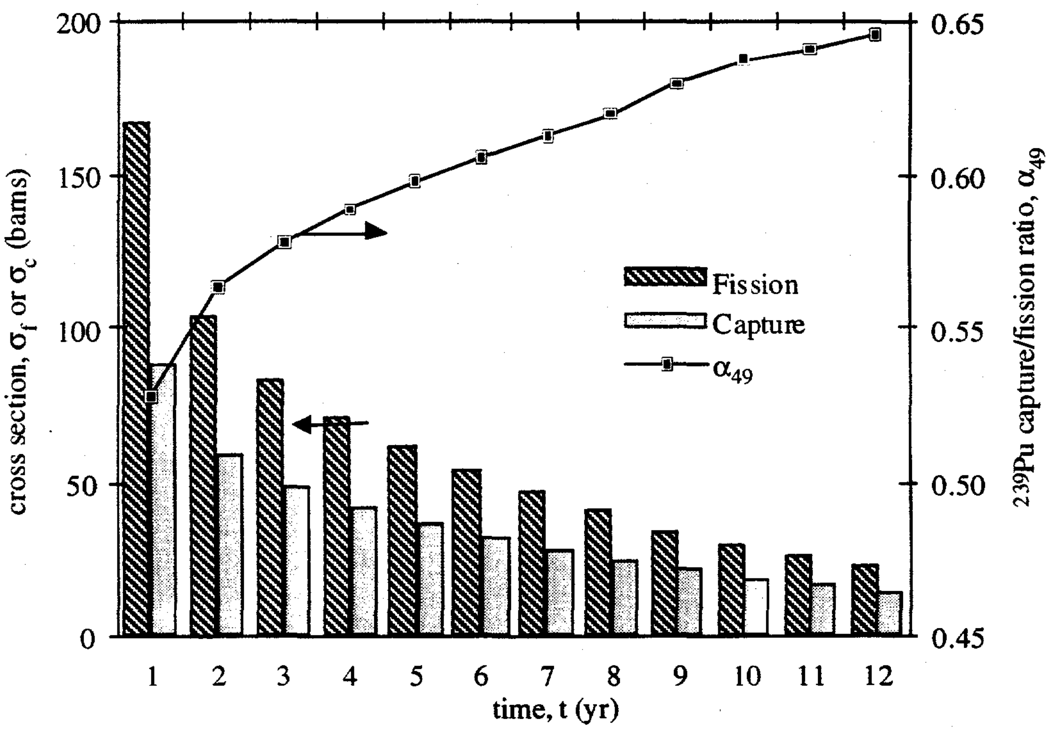

Figure E-4. Time-dependence of ^{239}Pu fission and capture cross sections as reported by MCNP4 in a homogeneous blanket; the capture- to-fission ratio varies 12% over the 12-yr period modeled.

Figure E-5 gives the plutonium burnup results for the three systems considered. In terms of plutonium burnup, counting all isotopes, the heterogeneous designs perform about as well as each other, and both out-perform the homogeneous design. It appears that the harder spectrum in the homogeneous system produces more capture reactions per fission in the fissile Pu isotopes ^{239}Pu and ^{241}Pu, respectively. The effect is to produce the non-fissile isotopes ^{240}Pu and ^{242}Pu at a more rapid rate.

An interesting feature of the homogeneous case in Fig. E-5 is that it reaches a maximum after about 5 years and decrease thereafter. It is suggested that fuel-salt cleanup would occur once every ~5 yr. in this system. Otherwise, fuel-salt cleanup would not be required until actinide solubility limits are reached (??? molar), which could be as long as 20-25 yrs.

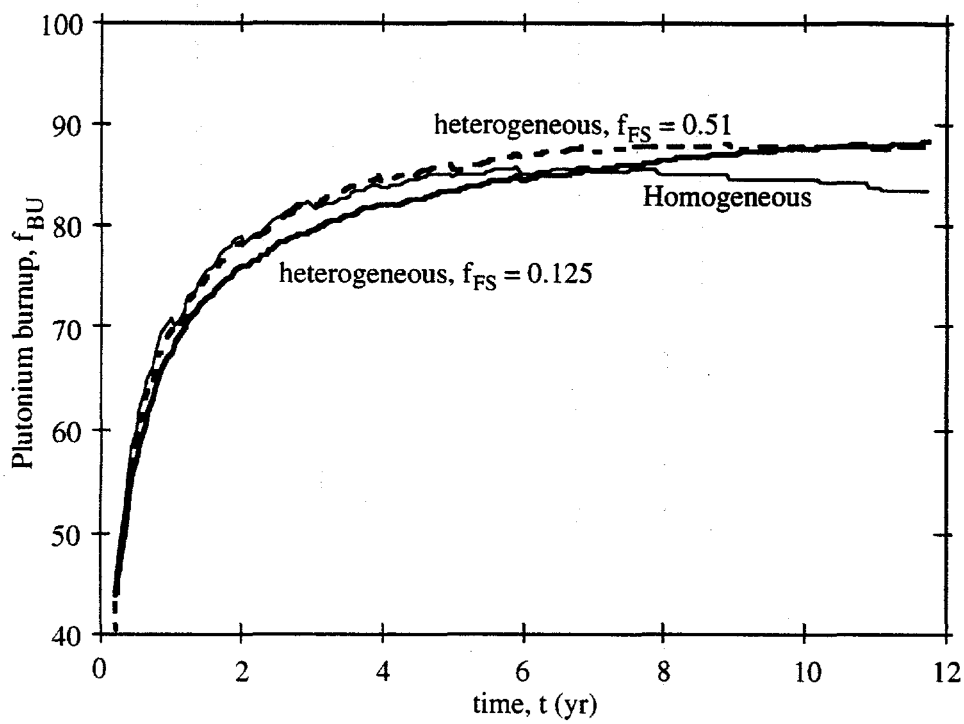

Figure E-5. Cumulative burnup of plutonium in the three blanket models indicated; burnup is defined as the moles fissioned divided by the moles fissioned plus the inventory. This computation indicates over 80% burnup; higher burnup is possible if the plutonium feed is stopped and the k_{eff} is allowed to decrease.

The dpa rate in the regions of the highest flux in the blankets are summarized in Fig. E-6. If an upper limit of 33 dpa is taken before the graphite must be replaced, for the blanket with 12.5% fuel-salt volume fraction, no graphite replacement would be required for at least 12 years. Replacement would have to begin in about 6 years for the heterogeneous system with 51.% fuel-salt volume. Because of no graphite in the fuel-salt region for the homogeneous system, these calculations are taken to apply to the graphite reflector material located immediately adjacent to the highest flux region in the blanket. Some of this material would have to be replaced every few years. It appears that the graphite lifetime and associated waste and downtime problems would be less for the heterogeneous system operated with a 12% fuel-salt volume fraction.

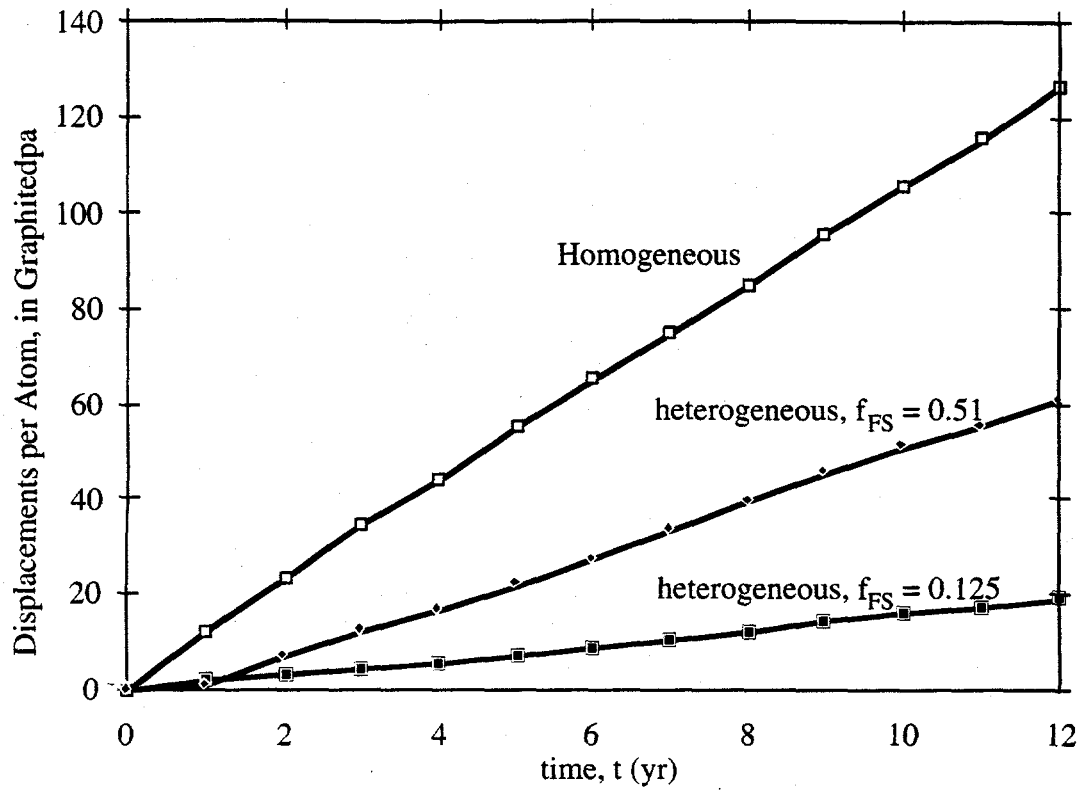

Figure E-6. Cumulative graphite dpa reported by MCNP4 in the three blanket models indicated as a function of irradiation time under constant-power conditions; these results correspond to the highest-flux regions for each system.

Table E-I summarizes the performance of the three Target-Blanket cases considered. The solubility limit of plutonium in the salt is taken to be 0.01 atom fraction and may present a problem in some systems. The final plutonium loading is not sufficient in any of the three cases considered to cause solubility problems, even if all the fission products having similar have similar chemistry (valence) are included. The number of moles of plutonium fissioned is large compared to the initial or final inventory. The dpa rates listed in this table are averaged over the system fuel-salt volume and over the 12-year period of operation.

Table E-I Blanket Performance Measures for the Three Cases Shown on Fig. E-2.[(a)]

Configuration	Fuel-Salt Fraction f_{FS}	Final Plutonium Concentration (molar)	Final Plutonium Concentration kg/m^3	Trivalent Atom Fraction	Total Plutonium Fissions (moles)	Plutonium Burnup Fraction	Volume-Average Damage Rate (dpa/yr)
heterogeneous	0.125	0.106	45.6	0.004	13.076	0.89	1.3
heterogeneous	0.51	0.112	44.9	0.004	13.076	0.88	4.7
homogeneous	–	0.160	57.9	0.0046	13.076	0.84	13.0

(a) Power density in the active fuel volume is 90 MW/m^3. The ratio of active fuel-salt blanket volume to external fuel-salt volume is unity; total fuel-salt volume is 15.8 m^3; the initial plutonium salt loading is 0.03 molar; a total of 13,076 kmoles of actinide is fissioned in 12 full-power years; total fission power is 711 MW and the duty factor is $p_f = 1.0$.

E.4. Conclusions

At beginning of life an ABC fuel salt would be dilute in plutonium, and the majority of captures and fissions occur in the ^{239}Pu and the resonance absorber present at the beginning of life. As the fission products and higher actinides build-in with time, a higher plutonium inventory is required to maintain a constant k_{eff} (*i.e.*, constant thermal power output and accelerator power input). The spectrum becomes more epithermal, and individual one-group cross sections for fissile species can vary significantly. These changes can occur rapidly (???? yrs), and the MEDAL method will give greater confidence in the results under these circumstances. The calculations showed that the modified method with a 1-yr time step between spectral updates is probably an adequate procedure for heterogeneous or homogeneous ABC systems. The calculations that modeled the full Target-Blanket showed that a heterogeneous design using a 12.5% fuel-salt volume fraction predicted a higher plutonium burnup and graphite replacement at a reduced rate than either a harder-spectrum heterogeneous design or a homogeneous design.

Appendix F. Sizing of Beam Bending and Expansion Configuration

The bending and spreading of the full-energy proton beam onto the target window can impact strongly the size of the primary containment building and the Target/Blanket maintenance scheme. Early into the ABC Plant Layout Study it was decided to first bend the (split) horizontal beam 90° downward in the direction of the target, and then to spread the beam from its original radius $r_B \sim 0.01$ m to a value $R_{TAR} \sim 0.5$ m at the target (window). The beam spreading is to be achieved by an unfocused drift along a vertical drift tube of expansion length $\ell_{EXP} \simeq 10$ m, as determined by unpublished calculations[21,22]. Figure F-1 illustrates this geometry. The purpose of this Appendix is to give a parametric basis for the choice of beam bending radius, bending-magnet size, and associated power requirement.

F.1. Model

As indicated on Fig. F-1 an idealized quadrupole magnet is assumed to provide the horizontal magnetic field, B(T), needed to bend a proton beam of momentum p(GeV/c) and kinetic energy E_B(Gev), where $c = 3(10)^8$ m/s is the speed of light in vacuum. Using the cyclotron formula, p = eBR, where $e = 1.602(10)^{-19}$ J/eV and R(m) is the radius of curvature (Fig.F-1), the following expression results,

$$B(T)R(m) = 3.33p'(GeV), \tag{F-1}$$

where p′(GeV) = pc/e; in these units $p' = E_B\sqrt{1 + 2E_o/E_B}$, where $E_o = 0.938$ GeV is the proton rest mass. For a idealized N-pole bending magnet (N = 4, Fig. F-1), B and the current per conductor, I, are related as follows:

$$B(T) = N\frac{\mu_o I}{2\pi(a + \delta)} , \tag{F-2}$$

where $\mu_o = 4\pi(10)^{-7}$ h/m is the permeativity of free space and the distances a and δ are the beam-tube and conductor radii, respectively. For the purposes of this analysis, the quadrupole conductors are assumed to located close to the beam tube, with imposition of the $<10^{-6}$/m beam scrape-off fraction[45] not requiring the need for intervening shielding.

An approximate costing algorithm for the bending magnets is used to give guidance of the two main component costs: the magnet structure and the power supplies for the driven copper coils assumed to provide the field B. If η is a nominal coil resistivity that reflects area requirements for coolant and structure, the ohmic power dissipated in and the mass of the simple quadrupole coils illustrated in Fig. F-1 are given by

$$P_\omega(W) = \eta\frac{R}{\delta^2} I^2 \tag{F-3}$$

$$M(kg) = \rho\, \pi^2\, R\, \delta^2 , \tag{F-4}$$

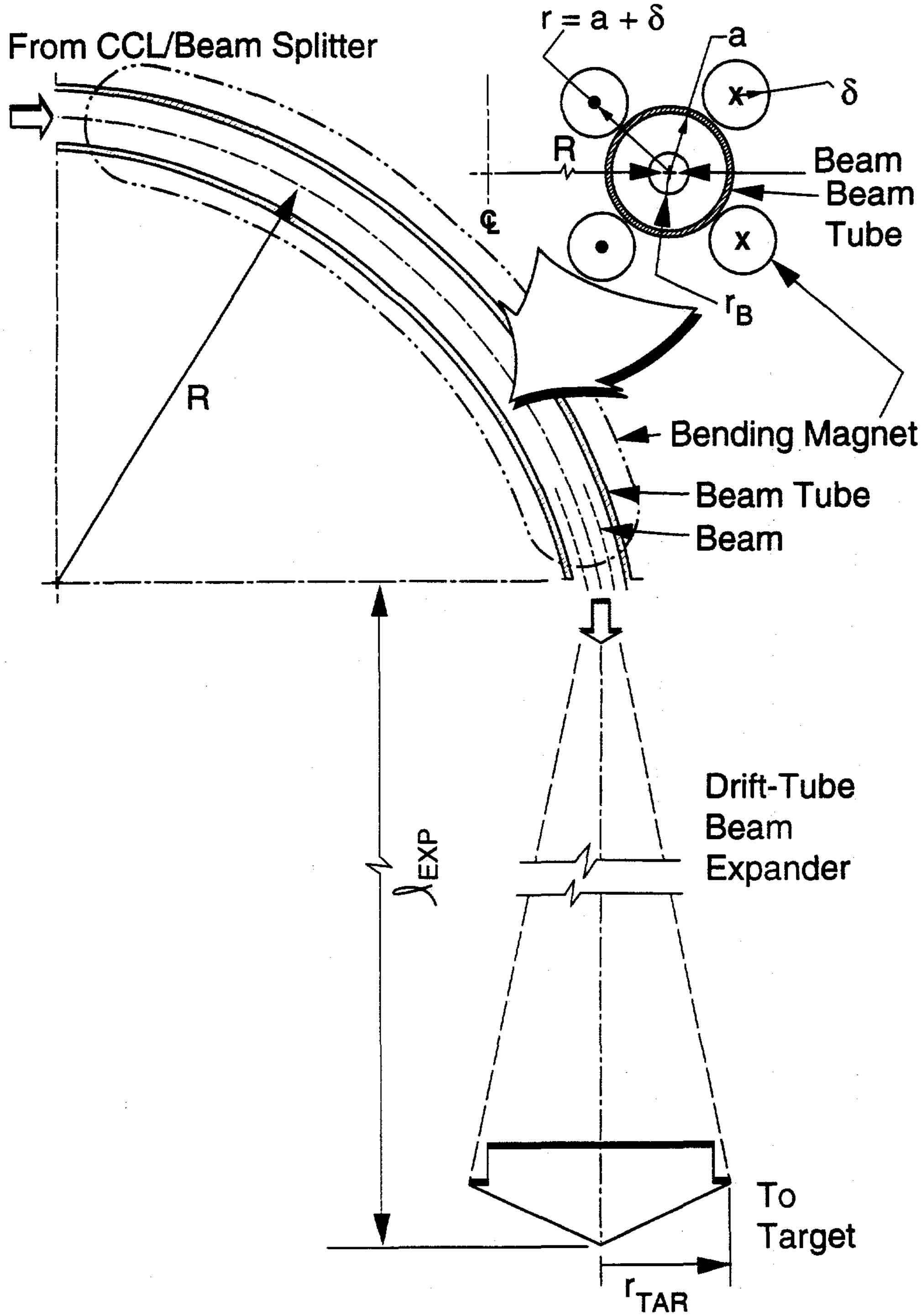

Figure F-1. Schematic diagram of model used to size the ABC beam-bending magnets and associated systems; dimensions and associated notation are used in the analyses leading to the design recommendations of R and δ, as well as associated power requirements.

where ρ is a nominal density of the quadrupole conductors. If the unit cost of quadrupole and power supplies are c_M($/kg) and c_P($/W), respectively, the cost of the bending magnet system can be expressed as follows:

$$C_{BM}(\$) = \alpha_1 \frac{(a + \delta)^2}{\delta^2 \ R} + \alpha_2 \ \delta^2 \ R \ , \tag{F-5}$$

where,

$$\alpha_1(\$m) = \frac{c_P}{(N \ \mu_o / 2 / \pi^2)} \ \eta \ (BR)^2 \ ,$$

$$\alpha_2(\$/m^3) = \pi^2 \ c_M \ \rho \tag{F-6}$$

The unit cost c_M is chosen to include provisions for beam tube, refocusing magnets (if any), support structure, *etc*. While this expression shows C_{MP} is minimized for a given δ or R, the following conductor power-density (cooling) constraint eliminates consideration of a "local" cost minimum as a basis for making design choices of R and δ:

$$a + \delta = \alpha_3 \ \delta^2 \ R \tag{F-7}$$

$$\alpha_3(1/m^2) = \pi \frac{N \ \mu_o}{2\pi} \ \frac{j}{BR} \ , \tag{F-8}$$

where $j(A/m^2)$ is the nominal current density in the quadrupole conductor. Under these conditions, the total cost, C_{BM}($) and the unit cost, $c_{BM}(\$/m) = C_{BM}/(\pi \ R/2)$, are given by

$$C_{BM}(\$) = \alpha_4(a + \delta) \tag{F-9}$$

$$c_{BM}(\$/m) = \alpha_5 \ \delta^2, \tag{F-10}$$

where,

$$\alpha_4(\$/m) = (\alpha_1 \alpha_3^2 + \alpha_2)/\alpha_3 \tag{F-11}$$

$$\alpha_5(\$/m^3) = (2/\pi) \ (\alpha_1 \alpha_3^2 + \alpha_2) \ . \tag{F-12}$$

The local minimum cost is obtained by differentiating Eq. (F-5) with respect to R and δ and setting the respective derivatives to zero, which gives the following results:

$$R^2 = (\alpha_2 / \alpha_2) \ \frac{(a + \delta)^2}{\delta^4} \tag{F-13}$$

$$R^2 = \left(1 - \frac{\delta}{a + \delta}\right) \left(\frac{\alpha_1}{\alpha_2}\right) \frac{(a + \delta)^2}{\delta^4} \ . \tag{F-14}$$

It is seen that these two minima are satisfied only for $\delta = 0$ (*i.e.*, infinite radius of curvature), which is the minimum-cost design suggested by Eq. (F-9). This minimization of the bending-magnet costs by a design that minimizes δ and maximizes R will have serious cost impacts on other parts of the ABC system (*e.g.*, the containment building).

As seen from Eq. (F-9), the total cost for a given (nominal) current density scales linearly with δ, with the (minimum-cost) intercept being α_4 a and the slope being α_4. Minimization of α_4 *vis-a-vis* minimization of α_3 [refer to Eq. (F-11)] leads to minimized bending-magnet cost. The α_3 value that minimizes α_4 and the associated current density are given by,

$$\alpha_3 = \sqrt{(\alpha_2 / \alpha_1)} \qquad \text{(F-15)}$$

$$\hat{j}(\text{A/m}^2) = \sqrt{\frac{\rho \; c_M}{\eta \; c_P}} \qquad \text{(F-16)}$$

For this minimum-cost constraint, α_4 equals $2\sqrt{\alpha_1 \; \alpha_2}$. The conductor (nominal) power density corresponding to these minimum-cost conditions is

$$\eta \; \hat{j}(\text{W/m}^2) = \rho \; c_M / c_P \; . \qquad \text{(F-17)}$$

It is easily shown that these minimum-cost conditions result in equal costs for the bending-magnet structure and power supplies.

F.2. Results

The procedure used here to determine R and δ for the purposes of scoping the ABC layout in a way to assure that the bending magnet has a reduced cost impact on other parts of the ABC system. This procedure, which is not unique, first fixes the unit cost of the bending magnet, c_{BM}(\$/m), at a value that is comparable to the rest of the accelerator structure [$c_{CCL} \sim 200{,}000$ \$/m)][2]. Secondly, the cost-minimizing current density [Eq. (F-16)] is used, while monitoring the conductor power density to assure an easily coolable configuration. Table F-I lists the parameters that evolve from this assumption and the specification of a $E_B = 0.80$ GeV beam. Additionally the beam-tube radius is constrained at the value given in Table F-I (a = 0.10 m). The parametric dependence of C_{BM}, c_{BM}, R, and B on δ is illustrated on Fig. F-2.

Table F-I Summary of Assumed and Computed Parameters for the ABC Bending Magnets

Input	
Beam energy, E_B(GeV)	0.80
Beam momentum, p'(Gev) = pc/e	1.46
Field-radius product, BR(Tm)	4.88
Number of conductors, N[(a)]	4
Nominal conductor resistivity, $\eta(10^{-8}$ ohm m)[(b)]	2.00
Unit cost of magnet power supplies, c_P(\$/W)	0 .30
Unit cost of beam-bending structure, c_M(\$/kg)[(c)]	400.
Nominal density of beam-bending structure, ρ(kg/m^3)	7,800.
Optimum current density, j(MA/m2)[(d)]	22.8
Conductor power density, η j(MW/m3)	10.4
Computed constants:	
• α_1(M\$ m)	0.22
• α_2(M\$/m^3)	30.79
• α_3(1/m^2)	11.75
• α_4(M\$/m)	5.24
• α_5(M\$/m^3)	39.21
Unit cost of bending magnets, c_{BM}(M\$/m)[(e)]	0.20
Output	
Conductor radius, δ(m)	0.072
Beam radius of curvature, R(m)	2.83
Magnetic field, B(T)	1.58
Conductor current, I(MA/conductor)	0.37
Resistive power losses, P_Ω(MW)	1.50
Mass of conductor, M(tonne)	1.10
Costs(M\$)	0.90
• magnets, beam tube, structure, *etc.*	0.45
• power supplies	0.45

(a) simple (continuous) quadrupole.
(b) includes provisions for structure and coolant.
(c) prorated upward to include costs of beam tube, structure, shielding, *etc.*
(d) based on minimum bending magnet cost constraint, Eq. (F-16).
(e) chosen to be comparable to unit cost of main accelerator structure[15,18].

On the basis of this simplied engineering and costing model of the ABC beam-bending magnets, the assumption of $c_{BM} \sim c_{CCL}$, and a = 0.10, a design is suggested where R = 2.8 m and δ = 0.07 m; the overall envelope of this system is $2(a + 2\delta) \sim 0.5$ m without provisions for shielding. As note above, this minimum-cost condition corresponds to equal structure and power-supply costs. The conductor power density for these conditions is an acceptable ~10.4 MW/m^3. In consideration of the model used and the purposes of its application, further refinement is not warranted. Key dimensions of R = 3.0 m and δ = 0.1 m are suggested for the ABC plant layout, along with L_{EXP} = 10. m for the drift-tube beam expander; control and (continual) active feed back of the final beam "foot print" will probably necessitate a large quadrupole magnet located at the exit of the beam-bending magnets; and added cubic meter is suggested to represent this system in the ABC plant layout.

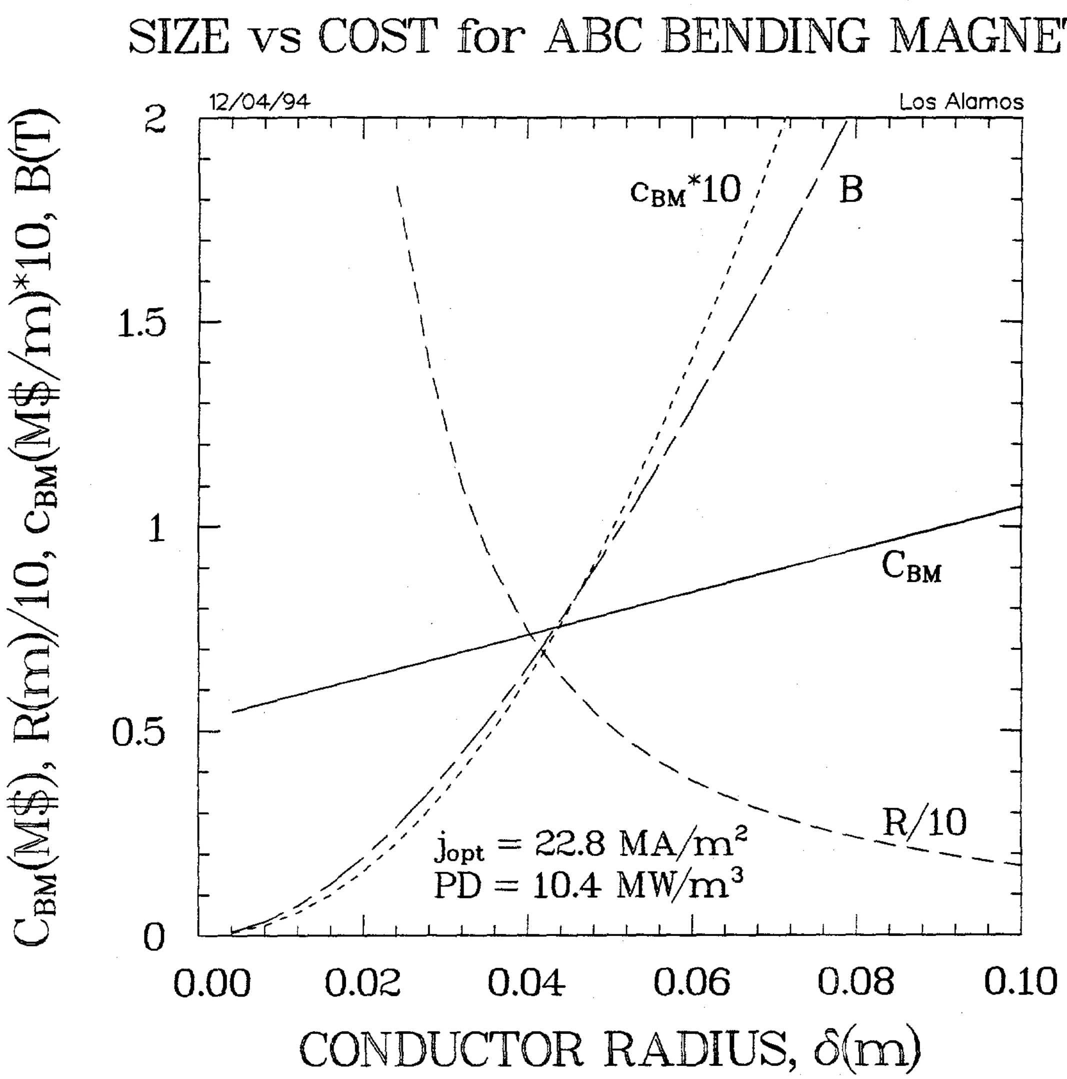

Figure F-2. Parametric dependence of bending-magnet total cost, C_{BM}, radius of curvature, R, unit cost, c_{BM}, and magnetic field, B, for the indicated fixed parameters.