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**PRECONCEPTUAL ABC DESIGN
DEFINITION AND SYSTEM
CONFIGURATION LAYOUT
APPENDIX A**

March 1995

Prepared by

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for

**Los Alamos National Laboratory
under Subcontract 2254V0015-3Y as Amended**

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PRECONCEPTUAL ABC DESIGN DEFINITION AND SYSTEM CONFIGURATION LAYOUT

APPENDIX A

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A1: OPERATING CONDITIONS FOR THE STEAM-CYCLE ABC SYSTEM

INTRODUCTION

While the MSBR is the point of reference and departure for the ABC system design, it is important to recognize that the design approach for the ABC system differs significantly from that for the MSBR.

Breeding vs Destruction of Fissile Material

A careful review of the MSBR design shows that this system was a finely tuned breeder design. To achieve a high breeding gain, the carrier salt could only contain constituents with a very low neutron absorption cross section; furthermore, the carrier salt constituents had to promote neutron moderation. Beryllium was one constituent, and lithium another. However, lithium had to be enriched in ^7Li to 99.99% to be acceptable. Sodium was excluded because of its "high" neutron absorption cross section (the 2200 m/s value for sodium is 50 times higher than for beryllium). A gas sparging system to effectively remove Xe and Kr from the fuel salt had a very high priority to keep low any breeding penalty. Fission products had to be removed continuously and ^{233}Pa had to be isolated so as not to lose it due to neutron absorptions in the reactor.

All these issues are of low priority for an ABC system whose mission it is to destroy as effectively as possible the fissile material inserted into the core without producing any new fissile material.

Optimistic vs Realistic Expectations

During the 1950's and 1960's it was expected that nuclear reactors would dominate the electricity-generating capacity that had to be added to the grid. Projections for the installed nuclear capacity by the year 2000 exceeded 2000 GWe. To install such a capacity and ensure fuel supplies for the life of the plants, and to meet a rapidly increasing demand for nuclear power, the ability to breed fuel in a reactor had a very high priority.

The MSBR of the late 1960's and early 1970's was in competition with the LMFBR for breeding and cost performance. Breeding gains for oxide- and metal fueled LMFBRs were expected to be in the range of 0.4 - 0.55. Oxide- and metal-fuel LMFBRs were expected to have doubling times of 15 years and less than 10 years,

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respectively. However, during the 1980's those projected doubling times had risen to 25-30 years, and 20 years, respectively.

Equally optimistic were the operating conditions for the breeders designed during the late 1960's and early 1970's. Mixed mean outlet temperatures were expected to be in the 1100 - 1150°F range, but dropped to the 875 - 930° F range during the 1980's (the CRBRP had a mixed mean outlet temperature of 975°F).

Thermal efficiencies of 40-45% were expected during the 1960's, but the ALMR thermal efficiency was projected to be in the 35 - 36% range. Fuel cycle cost was expected to be less than 0.5 mills/kWh, and capital cost was expected to be in the \$100 - 120/kWe range (some PWR turn-key plants were sold at that time at a fixed capital cost of \$85/kWe installed). We know today that fuel cycle cost are a factor 10 higher, and capital cost are a factor 20 and more higher.

It is, therefore, not surprising that the MSBR with a somewhat marginal breeding gain of only 0.06 and a projected doubling time of over 20 years, to be a successful competitor, had to have low capital cost. This was achieved by keeping fissile inventory cost very low through frequent reprocessing and reducing the out-of-pile inventory, and aiming for a very high thermal efficiency. Duplicating the Bull Run Steam Plant conditions promised a very high thermal efficiency of 44.5%, however, at the expense of 3800 psi pressure¹.

Safety Then and Now

During the 1960's and early 1970's, being able to shut down the reactor in case of an accident was the major safety concern. Safety reports from that time explained, for example, that fuel assembly melting could occur, albeit at a low probability. The reactor would then be shut down, the damaged assembly would be removed and after a short time, upon completion of repair and decontamination work, the reactor would be restarted.

Shutdown reliability is still a major design issue for today's reactors and assurance has to be provided that failure to shut down a reactor has to be an event that is outside the design basis. The major safety concerns today, however, are with reliable decay heat removal in case of an accident and, after the reactor is shut down, the availability of service water as the ultimate heat sink (or a cooling tower), the loss of coolant, and

(1) The MSBR designers were aware that using the Bull Run Steam Plant conditions would introduce design complications and that it might be worthwhile to be less aggressive in terms of steam pressure in favor of design simplification.

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overpower transients resulting from accidents. The early reactor designs showed little in terms of passive and inherent safety features which, today, are mandated for advanced reactors by the NRC. Economics is viewed in the context of safety and reactor availability.

Therefore, for the ABC system design, safety came first, and ABC systems economics has to be achieved within the safety constraints.

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A1.0 Operating Conditions for the ABC Steam Cycle

1.1 ABC primary system operating conditions

The operating conditions for the ABC system are constrained by the carrier salt selections for the fuel and coolant.

1.1.1 fuel salt compositions

For molten salt reactors two chemical fuel forms have been considered, fluorides and chlorides. Fluoride salts have the ability to dissolve large amounts of thorium and uranium fluorides as well as fission product fluorides, show low viscosities and a low vapor pressure at high temperatures. Uranium can easily be separated from the salt in a fluoride volatility process that stems from the high volatility of UF_6 (boiling point of less than $100^\circ C$) and the low volatility of UF_4 (boiling temperature over $1400^\circ C$).

Chlorides are less well known than fluorides. Because chlorine is less reactive than fluorine, it is expected that not all fission products would be available in the melt in chloride form. Furthermore, as shown in Table 1, chlorides would have to use enriched chlorine because the neutron absorption cross section for natural chlorine is very high.

Therefore, only the fluoride form of salts has been pursued as a suitable form for reactor applications.

The following constituents for the fuel salt have to be considered:

LiF
BeF₂
ZrF₄
NaF
PuF₃

Because the PuF₃ concentration will be initially less than 0.05 mole %, it is assumed that, at least initially, the material properties of the fuel salt are determined by the carrier salt. At one time bismuth fluoride had also been considered. However, because of its reduction by structural materials, it was removed from the list of contenders.

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The preferred configuration for molten salt fuel is a thermal neutron spectrum. This will result in a much lower fissile inventory than in a fast configuration. Thermal spectrum configurations can be achieved by either using an internal graphite moderator, or by surrounding an unmoderated core of small dimensions with a thick graphite reflector.

Table 1: Elements/Isotopes Suitable for High-Temperature Fuels

MATERIAL	σ_s (barns at 2200 m/sec)
Nitrogen-15	0.000024
Oxygen	0.0002
Deuterium	0.00057
Carbon	0.0033
Fluorine	0.009
Beryllium	0.010
Bismuth	0.032
Lithium-7	0.033
Boron-11	0.05
Magnesium	0.063
Silicon	0.13
Lead	0.17
Zirconium	0.18
Phosphorus	0.21
Aluminum	0.23
Hydrogen	0.33
Calcium	0.43
Sulfur	0.49
Sodium	0.53
Chlorine-37	0.56
Tin	0.6
Cerium	0.7
Rubidium	0.7
Chlorine (nat.)	33.8
Lithium (nat.)	71

A fast spectrum is difficult to achieve with a molten salt based on fluorides. The German MOSEL project was based on uranium, sodium and beryllium fluorides and achieved only an epithermal spectrum. The MOSEL core was surrounded by a blanket fluid consisting of the same carrier salt but with thorium fluoride as the fertile material.

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For plutonium destruction a fast spectrum might not be optimal because of the solubility limits for PuF_3 . As fission products build up the plutonium inventory has to increase to maintain the same eigenvalue. The fissile inventory in a fast molten salt reactor would start out very high but then increase less steeply as in a thermal system because of the lower reactivity worth of fission products in such a spectrum. However, it is expected that the solubility limit for PuF_3 (and the rare earth trifluorides) is reached earlier than in a thermal system.

The following mixtures of fluorides are candidates for fuel carrier salt:

LiF-BeF_2
 LiF-NaF-BeF_2
 NaF-BeF_2

Additions of ZrF_4 have been considered to avoid fuel precipitation in the presence of oxidants.

Table 2 summarizes the material properties of these salts and two additional salts:

LiF-BeF_2
 LiF-NaF-BeF_2
 NaF-BeF_2
 $\text{LiF-BeF}_2\text{-ZrF}_4\text{-UF}_4$
 $\text{LiF-BeF}_2\text{-ThF}_4\text{-UF}_4$

The second to last salt is the MSRE fuel salt that was shown for comparison purposes and to illustrate the effect of a ZrF_4 addition. The last salt is the MSBR salt shown for comparison only.

Option 1: LiF-BeF_2

This was the fuel carrier salt and coolant salt for the MSRE. The MSRE operation with partially enriched uranium produced about 500 g of plutonium and analyses of samples showed that all had stayed with the salt. This salt is well characterized. It was also the carrier salt for the MSBR fuel. This salt contains lithium that needs to be enriched in Li-7 which is costly. Both Li-6 and Li-7 are the main sources for tritium production.

Option 2: LiF-NaF-BeF_2

This salt was not considered for the MSBR because it contains sodium which is a strong absorber. However, its liquidus temperature is 130°C lower than for the option

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1 carrier salt. This salt contains lithium that needs to be enriched which is costly. Lithium is the main source for tritium production.

Option 3: NaF-BeF₂

This salt was not considered for the MSBR because it contains sodium which is a strong neutron absorber. Its liquidus temperature is very close to that of the option 2 carrier salt. Its viscosity is higher but still lower than that of water at 600°F which is 86 centipoise. There are no significant differences in the other physical properties.

Option 4: Either of the previous options with a ZrF₄ addition

Table 2 shows that the addition of a small amount of ZrF₄ does not change properties significantly. The change in viscosity might very well reflect the reduction in BeF₂ and to a lesser extent the addition of ZrF₄.

1.1.2 coolant salt compositions

Commonly used in other reactors are either water, sodium or helium coolants. However, they are not suited for molten salt reactor applications.

Water leakage into the fuel would lead to the precipitation of fuel and the production of high-pressure steam, hydrogen and oxygen.

Sodium reacts with the fuel in that it would reduce all cations except lithium. Beryllium metal could alloy with structural metals. Pu could precipitate in the primary system and could later lead to an unacceptably high increase in reactivity. Sodium also reacts vigorously with water and steam and produces hydrogen, sodium hydroxide, and sodium oxide.

The use of helium would require a high operating pressure that would negate one of the advantages of the molten salt reactor, namely its low operating pressure characteristic. In case helium would leak into the fuel salt, unacceptably high power surges could result due to the positive reactivity coefficient for voids or bubbles.

Therefore, different types of coolants have to be investigated.

The following candidate compositions for coolant salt are shown in Table 3 and briefly discussed below:

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NaBF₃-NaF
LiF-BeF₂
LiF-NaF-BeF₂
NaF-BeF₂
KNO₃-NaNO₂·2-NaNO₃ (Hitec)

Option 1: NaBF₃-NaF

The main reasons for selecting sodium fluoroborate instead of LiF-BeF₂ as the MSBR coolant were:

1. avoiding the high cost for using enriched Li-7
2. avoiding the high freezing point of the MSRE coolant

The principal advantages of sodium fluoroborate are the low cost, low viscosity, and its capability to trap tritium. There are claims that tritium would be completely trapped in this coolant, however, the experimental data available for review do not support this claim. In its pure form, it is not corrosive to all major or minor constituents of structural metal alloys.

The chief disadvantage of this coolant arises from its mixing with fuel salt during an accident. This leads to the generation of BF₃ gas and probably a redistribution of fissile material between immiscible phases. Other disadvantages of this coolant salt are its thermal decomposition that produces BF₃, and its reaction with steam. In case of steam inleakage, metal corrosion can be greatly increased. In case the coolant leaks into the steam system, the pH of water will be changed due to the formation of HF with fluoride-containing salts and the water will also contain ppm concentrations of fluorides. There is the potential for stress-corrosion cracking.

The decomposition of the coolant leaves most of the BF₃ in the salt, but some of it can be swept out of the pump bowl by the required sweep gas. If BF₃ is not added, the coolant will slowly change its composition and increase in freezing point. The He-BF₃ mixture cannot be continuously discharged because of the toxicity and chemical reactivity of BF₃ and the need to recover helium. If BF₃ is released into the off-gas system, it may react with the charcoal beds.

Option 2: ⁷LiF-BeF₂

This salt is fully compatible with the fuel salt as demonstrated in the MSRE. Lithium is present in form of Li-7 to prevent discarding all fuel salt in case of a leak of the

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coolant into the fuel salt.

In the MSRE, heat was transferred from the coolant at 546°C (1015°F) to a low-efficiency, air-cooled radiator. In the MSBR, the coolant has to transport the heat to portions of a steam system at much lower temperatures than 546°C. For coolants with LiF contents of 60 - 67 mole %, the freezing points are between 440 and 460 °C. To prevent this coolant salt from freezing in the steam-producing equipment would require an abnormally high feedwater temperature, resulting in a decrease in thermal efficiency. It had been estimated that about a 10% drop in thermal efficiency would occur in a supercritical steam cycle when the feedwater temperature would have to be increased from 700°F (371°C) to 800°F (426°C).

Furthermore, this coolant is ineffective in trapping tritium.

Option 3: ${}^7\text{LiF-NaF-BeF}_2$

This coolant is compatible with the fuel salt. It has a low freezing point (that is even below that for sodium fluoroborate), and a low vapor pressure. In its pure form, it is not corrosive to all major or minor constituents of structural metal alloys.

Its disadvantages are its high cost due to the enrichment of Li, its increased corrosivity when mixed with steam (similar to that for sodium fluoroborate), and its inefficiency in trapping tritium.

Option 4: NaF-BeF_2

Based on material properties, this seems to be a very desirable coolant salt. It has not been pursued for MSBR applications, probably because (1) in case of a leak into the fuel, a complex cleanup would be required, and (2) NaF could not be present in a marginal breeder like the MSBR. Furthermore, its ability to trap tritium has not been investigated.

Option 5: $\text{KNO}_3\text{-NaNO}_2\text{-NaNO}_3$ (Hitec)

The main advantages for this coolant are (1) its low melting point, (2) it is inexpensive and (3) it allows the use of carbon steel up to 850°F, and above this temperature stainless steel. It is also expected to trap tritium.

Its disadvantages are its violent reaction with graphite if a leak occurred in the IHX, its stability in a radiation field at high temperatures and the need for salt processing in case of a leak.

1.1.3 primary system options

Because of its violent reaction with moderator or reflector graphite, Hitec is eliminated from any consideration in the primary system. The other fuel-coolant combinations are briefly characterized in Table 4.

For the characterization of the fuel-coolant combination the following issues have to be addressed:

- do fuel and coolant react when mixed?
- is tritium produced in the fuel?
- if tritium is produced, does the coolant trap the tritium?
- if coolant leaks into fuel, is fuel processing required?
- melting point of coolant
- cost of coolant and fuel carrier salts
- is the coolant salt stable?
- is the fuel salt a strong neutron absorber?

These topics are briefly summarized in Table. 4.

1.1.4 proposed primary system operating conditions

Based on the evaluation of the different fuel-coolant salt combinations, at this time, it would be prudent to use:

- $\text{LiF-BeF}_2\text{-PuF}_3\text{-(ZrF}_4\text{)}$ as fuel
- $\text{NaBF}_3\text{-NaF}$ as coolant

These two salts have been investigated extensively under the MSBR Program. They are, by and large, well characterized. However, these choices are less than optimal for a Pu destroyer.

It is recommended to investigate also:

- $\text{NaF-BeF}_2\text{-PuF}_3$ as fuel
- NaF-BeF_2 as coolant.

This combination has the following advantages:

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1. for all practical purposes, no tritium is produced
2. the melting point for this fuel is 220°F lower than for the LiF-BeF₂-based fuel
3. the melting point of this coolant is slightly lower than that for the sodium fluoroborate coolant
4. fuel and coolant are fully compatible
5. the coolant is thermally stable
6. the cost for the carrier salt is low; no enrichment services are required

Potential disadvantages are:

1. sodium is a much stronger neutron absorber than lithium
2. the neutron spectrum is slightly harder than for the LiF-BeF₂ carrier salt which requires a slightly higher Pu inventory, but results in slightly lower negative reactivity worth of the fission products

However, it is conceivable that the advantages of the NaF-BeF₂-based salts can far outweigh the disadvantages. In regard to inventory requirements, later in life, the fissile inventory requirements are governed by the accumulated fission products in the core. I.e., the hardening of the spectrum due to the substitution of Li by Na has little effect.

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Table 2: Properties of Different Fuel Carrier Salts

Composition (mole %)	LiF* (65) BeF2 (29.1) ZrF4 (5) UF4 (0.9)	LiF (66) BeF2 (34)	LiF (23) NaF (41) BeF2 (36)	NaF (57) BeF2 (43)	LiF** (71.7) BeF2 (16) ThF4 (12) UF4 (0.3)
Liquidus Temperature, C (F)	434 (813)	458 (856)	328 (622)	340 (634)	500 (932)
<u>Physical Properties at 850 F (454 C)</u>					
Density (lb/ft3)	141	125.5	136	139	206
Heat Capacity, (Btu-lb-1- F-1)	0.47	0.57	0.47	0.44	0.33
Viscosity (centipoise)	10	40.4	40	65	12
Vapor Pressure at 1125 F (607 C) in (mm)	negligible	negligible	negligible	negligible	negligible
Thermal Conductivity, (watts/ C-cm)	0.01	0.01	0.01	0.01	0.008

* MSRE fuel salt; cited because it shows the effect of the ZrF4 addition on properties

** MSBR fuel salt composition is shown for comparison only

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Table 3: Properties of Different Coolant Salts

Composition (mole %)	NaBF ₃ (92) NaF (8)	LiF (66) BeF ₂ (34)	LiF (23) NaF (41) BeF ₂ (36)	NaF (57) BeF ₂ (43)	KNO ₃ (53) NaNO ₂ (40) NaNO ₃ (7)
Liquidus Temperature, C (F)	385 (725)	458 (856)	328 (622)	340 (634)	109 (228)
<u>Physical Properties at 850 F (454 C)</u>					
Density (lb/ft ³)	121	125.5	136	139	108
Heat Capacity, (Btu-lb-1- F-1)	0.36	0.57	0.47	0.44	0.37
Viscosity (centipoise)	2	40.4	40	65	3.1
Vapor Pressure at 1125 F (607 C) in (mm)	200*	negligible	negligible	negligible	negligible
Thermal Conductivity, (watts/ C-cm)	0.005	0.01	0.01	0.01	0.33

* pressure of BF₃ in equilibrium with this melt composition

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Table 4: Primary System Options Characteristics

FUEL	COOLANT	ADVANTAGES	DISADVANTAGES
LiF-BeF ₂ -PuF ₃	LiF-BeF ₂	extensive MSRE experience compatibility no processing required in case of leaks	high cost fuel and coolant high melting point tritium production no tritium trapping
LiF-BeF ₂ -PuF ₃	NaBF ₃ -NaF	extensive fuel experience low coolant melting point tritium trapping in coolant low cost coolant	high cost fuel and coolant fuel-coolant reaction thermal decomposition of coolant handling of BF ₃ tritium production fuel processing in case of leaks
LiF-BeF ₂ -PuF ₃	NaF-BeF ₂ -LiF	extensive fuel experience fuel/coolant compatibility low coolant melting point high cost for ⁷ Li	high cost fuel and coolant tritium production no tritium trapping fuel processing in case of leaks
LiF-BeF ₂ -PuF ₃	NaF-BeF ₂	extensive fuel experience fuel/coolant compatibility low coolant melting point low coolant cost	high cost fuel and coolant tritium production no tritium trapping fuel processing in case of leaks
NaF-BeF ₂ -LiF-PuF ₃	LiF-BeF ₂	part of LiF is replaced by NaF (lower cost, less tritium production) fuel/coolant compatible no processing required in case of a leak	high melting point coolant high cost coolant tritium production no tritium trapping

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NaF-BeF2-LiF-PuF3	NaBF3-NaF	part of LiF is replaced by NaF (lower cost, less tritium production) low melting point coolant tritium trapping in coolant low cost coolant	fuel-coolant reaction thermal decomposition of coolant handling of BF3 tritium production fuel processing in case of leak
NaF-BeF2-LiF-PuF3	NaF-BeF2-LiF	part of LiF is replaced by NaF (lower cost, less tritium production) low melting point coolant low cost coolant fuel/coolant compatible	high cost coolant tritium production no tritium trapping
NaF-BeF2-LiF-PuF3	NaF-BeF2	part of LiF is replaced by NaF (lower cost, less tritium production) low melting point coolant fuel/coolant compatible	tritium production no tritium trapping
NaF-BeF2-PuF3	LiF-BeF2	low melting point fuel no tritium production fuel/coolant compatible low cost fuel	high cost coolant high melting point coolant sodium absorptions fuel processing in case of leaks
NaF-BeF2-PuF3	NaBF3-NaF	low melting point fuel no tritium production low cost fuel low cost coolant	fuel/coolant reaction thermal decomposition of coolant handling of BF3 fuel processing in case of leaks
NaF-BeF2-PuF3	NaF-BeF2-LiF	low melting point fuel no tritium production low cost fuel fuel/coolant compatible	high cost coolant sodium absorptions

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NaF-BeF2-PuF3	NaF-BeF2	low melting point fuel low melting point coolant no tritium production low cost fuel low cost coolant fuel/coolant compatible	sodium absorptions
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1.2 evaluation of steam cycle conditions

At the time different breeder options were considered (i.e., the late 1960's), the LMFBR fuel cycle required a large centrally located reprocessing and refabrication plant that served between 25-50 GWe of installed capacity. This lowered the unit cost for reprocessing and refabrication and was one of the reasons why an EBR-II-type breeder with metallic fuel and an integrated fuel cycle was eliminated. Not unlike the EBR-II-type breeder, the MSBR was proposed as a breeder with its own reprocessing plant. To offset any cost penalty coming from the larger capital investment, the MSBR designers aimed for as high a thermal efficiency as possible to reduce power cost.

1.2.1 MSBR supercritical steam cycle

The MSBR copied the steam conditions for the Bull Run Steam Plant, a very large coal-fired electricity generating plant just outside of Oak Ridge, TN². Table 5 shows the steam conditions and steam generator design for the MSBR.

When applied to the MSBR this supercritical steam cycle yielded a net plant thermal efficiency of 44.4%.

However, there were particular requirements for the MSBR. The feedwater supplied to the steam generator had to be at a temperature high enough to avoid coolant-salt freezing. Tentatively, this temperature had been set at 700°F (371°C). It was also assumed that the cold reheat steam must be preheated to 650°F before it enters the reheaters. These constraints led to the selection of a system in which the final stage of feedwater heating is by direct mixing with high-pressure steam. This process established the need for pressure-booster pumps in the feedwater line, because after the feedwater is heated to about 700°F in the mixing chamber, its pressure must be raised to that of the steam generator inlet pressure of about 3800 psia.

The MSBR required that about 5.1×10^6 lb/hr of 551°F steam leaving the high-pressure turbine exhaust be preheated to about 650°F before it enters the reheaters. The arrangement that had been proposed is to reheat the steam by heat exchange with steam at steam generator exit conditions of 3600 psia and 1000°F. The flowsheet for the supercritical-pressure cycle with feedwater heated by mixing is shown in Fig. 1.

² This 950 MWe plant supplies steam at 3500 psia and 1000 F to the turbine throttle, with a reheat to 1000 F.

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It is not clear whether the use of reheat and its improvement in thermal efficiency is sufficiently cost-effective to offset the added complexity and cost for this system. If reheat is not used, external moisture separators are required to prevent excessive moisture in the last stages of the low-pressure turbines. For the MSBR it was tentatively concluded that reheat provides for a somewhat better turbine performance than moisture separation. However, ORNL-4541 makes reference to an LMFBR study of a 2400 psig, 900°F/900°F steam cycle, comparing moisture separation and steam reheat that concluded that the economic gain for reheat was not sufficient to offset the added complexities and reduction in plant availability.

It is recommended to investigate the reheat cycle in more detail because its elimination would lead to a substantial system simplification through the elimination of the reheaters, the reheat steam preheaters, and the flow proportioners that divide the coolant salt flow between the steam generators and the reheaters, the latter ones being large heat exchangers with coolant salt on the shell side and steam on the tube side.

1.2.2 MSBR subcritical steam cycle

During the design of the MSBR it was recognized that the supercritical steam cycle with 3800 psig pressure might be too high for a nuclear power plant, especially for a low-pressure system like the MSBR. A subcritical steam system was investigated for the MSBR with the following results³:

	Supercritical Cycle	Subcritical Cycle
Nominal feed temperature to steam generator, °F	700	700
Mixing pressure, nominal	3500	2600
Booster pump or steam compressor discharge pressure, psia	3800	2900
Booster pump or steam compressor power requirements, MWe	7.4	52
Steam flow through steam generator, lb/hr	9.5x10E + 6	19x10E + 6
Overall thermal efficiency, %	44.5	41.1

³ Table 5.2 taken from ORNL-4541 (June 1971)

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The subcritical steam cycle is a modified Loeffler cycle shown in Fig. 2. Both cycles use preheating of the cold steam and subsequent reheat. The difference comes from the heating of the feedwater in a subcritical-pressure cycle where the water is being converted to superheated steam in the drum and then compressed and blown into the steam generator that, in essence, is used as a superheater. Instead of the booster pumps used in the supercritical cycle, steam compressors are required to raise the steam pressure.

For both the supercritical and subcritical cycles, other ways to heat the feedwater need to be explored. Among them the use of a high-pressure heat exchanger the heat the feedwater to 700°F.

The effect of feedwater temperature on performance of the supercritical steam cycle for the MSBR has also been investigated with the following results⁴:

Nominal Feed Temperature, °F	Booster Pump Work, MWe	Steam Generator Flow Rate, lb/hr	Net Plant Efficiency, %
580	none required	$7.4 \times 10E + 6$	44.9
700	7.4	$9.5 \times 10E + 6$	44.5
800	87	$28 \times 10E + 6$	41.3

The case with a feedwater temperature of 580°F assumes that this is about the highest feedwater temperature that can be attained by regenerative feedwater heating. In this case, no special mixer or booster pump would be required, and it was assumed that the reheat steam would not require preheating. For the 700 and 800°F feedwater cases, the feedwater is heated by mixing with steam from reheat steam preheaters. The higher the feedwater temperature has to be raised, the higher are the power requirements for the booster pumps and the steam generator flow rates, but the lower are the net plant thermal efficiencies.

⁴ Table 5.4 taken from ORNL 4541 (June 1971)

1.2.3 other steam cycle conditions

Table 5 summarizes the steam cycle conditions for the CRBRP, the ALMR, CANDU 3 and CANDU Pickering. Among those listed, only CANDU Pickering has been in operation. It is noteworthy that the advanced reactors have by far the largest steam generator capacities.

The size of the MSBR steam generator was prudently selected. It was realized that about 3750 psia fluid pressure on the inside of the tubes imposes very severe environments on the heads and tube sheets. This led to the decision to keep the shell diameter at 18 inches to keep stresses within more tolerable limits. For the otherwise very advanced steam cycle (and because of it), the steam generator size was kept at 121 MW, which is the smallest of all the sizes listed.

1.2.4 proposed steam cycle conditions for the ABC

Instead of a horizontal U-tube steam generator design, a vertical U-tube design is proposed similar to that used in CANDU (see Fig. 3)

The following steam cycle conditions are proposed:

intermediate coolant	
inlet temperature, °F	1150
outlet temperature, °F	850
feedwater/steam	
inlet temperature, °F	700
outlet temperature, °F	1000
steam pressure, psia	1000

These are tentative conditions mostly dictated by safety considerations. Unless it can be justified, no reheat should be employed.

The steam pressure most likely has to be raised, perhaps even significantly (like to 2400 psia) to achieve a high plant efficiency. However, the safety implications should be assessed carefully. The molten salt system is attractive because of the low vapor pressure of the salt even at very high temperatures that allows for a low operating pressure. This leads to a simplified primary and secondary system design. While the selection of 1000 psia as steam pressure is arbitrary, analysis should support increases to ensure the low-pressure characteristics of the molten salt system is maintained.

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TABLE 5: CHARACTERISTICS OF SOME STEAM GENERATOR DESIGNS

FEATURE	MSBR	ALMR-REF	ALMR-MODB	CANDU 3	CANDU Pickering
Type	horizontal U-tube	helical coil	helical coil	vertical U-tube	vertical U-tube
capacity, MW	121	477	850	689.7	138
intermediate coolant					
inlet temperature, F	1150	864	890	587	560
outlet temperature, F	850	573	619	514	480
feedwater/steam					
inlet temperature, F	700	380	420	369	340
outlet temperature, F	1000	830	850	500	485
feedwater pressure, psia	3752	2072	2520	667	
SG outlet pressure, psia	3600	1800	2200	740	593
tube bundle length, ft	76.4	27	40		
# tubes	393	495	612		
overall HT surface, ft ²	3929	31,102	64,089		2600
steam flow, 10E + 6 lb/hr	0.633	1.6	3.0	2.74	0.53
efficiency, %	44.4	35.1	36.1	32.65	29.1

A2.0 Flow Rates and Component Dimensions

At this time, not a single ABC component or system has been designed. However, to better sketch out a plant layout, an attempt was made to determine flow rates and component dimensions, at the least as rough estimates that help to define a basic layout for the ABC system.

The following basic assumptions were made:

module size:	750 MWt
average core power density	100 MW/m ³
# of coolant loops per module	2
# of IHX per module	2
# of steam generators per module	2
fuel salt constituents	LiF-BeF ₂ -PuF ₃
coolant salt	NaBF ₃ -NaF
fuel inlet temperature	1050°F
fuel outlet temperature	1300°F
moderator fraction in core	0%
core length/diameter ratio	1.0
fuel region diameter	2.121 m
fuel flow area in core	3.533 m ²
axial/radial reflector thickness:	125 cm
core barrel thickness:	2 cm
gap between core barrel/vessel	4 cm
vessel wall thickness	3 cm
vessel diameter	4.80 m
inner diameter of inner outlet pipe	0.381 m (15 inches)
wall thickness of inner pipe	2.54 cm (1 inch)
inner diameter of outer pipe	0.557 m (21.93 inches)
wall thickness outer pipe	2.54 cm (1 inch)
outer diameter of outlet/inlet pipes	0.582 m (22.93 inches)
flow area in inner/outer pipe	0.114 m ²

Use was made of double-walled piping for the fuel leaving and returning to the core. The physical properties for the fuel⁵ were those for BeF₂-LiF because it is assumed that at the least initially the Pu content is close to 0.05 mole% which is too small to affect physical properties.

⁵ taken from "Physical Properties of Molten-Salt Reactor Fuel, Coolant, and Flush Salts", by S. Cantor, ORNL-TM-2316 (August 1968),

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2.1 primary system

Fuel flow rates were calculated according to:

$$P \text{ (power)} = FR \text{ (flow rate)} \times c_p \text{ (specific heat)} \times dT \text{ (temperature rise)}$$

$$7.5 \times 10^8 = FR \times 2385 \text{ J/kg-}^\circ\text{C} \times 138.9^\circ\text{C}$$

or

$$\text{Flow Rate} = 2264 \text{ kg/s}$$

The average core flow velocity is calculated from the correlation:

$$FR = \text{fuel density (kg/m}^3\text{)} \times \text{fuel flow area (m}^2\text{)} \times \text{fuel velocity (m/s)}$$

$$2264 \text{ kg/s} = 1947 \text{ kg/m}^3 \times 3.533 \text{ m}^2 \times \text{fuel velocity}$$

or

$$\text{fuel velocity} = 0.329 \text{ m/s}$$

The volumetric flow rate is calculated from the correlation:

$$FR = VFR \text{ (volumetric flow rate, m}^3\text{/s)} \times \text{density (kg/m}^3\text{)}$$

or

$$\text{volumetric flow rate} = 1.163 \text{ m}^3\text{/s}$$

The fuel travel time through the core region is 6.4 s.

The outlet flow is conducted through two double-walled pipes, each with a flow of 1132 kg/s. Using the above-cited correlations, the average fuel velocity in either the outer or inner pipe is 5.18 m/s (17 ft/s).

As shown in section 3, the two pipes separate shortly after leaving the vessel. Heat transfer from the hot to cold fluid should be minimal.

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2.2 secondary system

Table 6 compares the MSBR IHX with that for the CRBRP. It is noteworthy that the smaller-capacity IHX for the CRBRP has less than half the tubes of comparable length but is housed in a vessel over twice the height and over 50% larger diameter than the MSBR IHX. The CRBRP IHX already passed licensing review whereas the MSBR IHX has never been reviewed by the NRC.

The IHX for the ABC system is a scaled-down version of the MSBR IHX design. However, it is assumed that inlet and outlet plena are reached from the sides to eliminate the need for the space on top of the IHX to accommodate the inverted U-shaped outlet/inlet secondary flow arrangement.

For scaling purposes it was assumed that the same MW per meter of tube length applies, and that the tube length is 3.5 m. This leads to the following IHX characterization:

	ABC	MSBR
power, MW	375.	556.3
central downcomer diameter, m	0.4	0.508
tube length, m	3.5	7.44
number of tubes	8315	5803
tube diameter, in.	same	0.375
tube pitch, in.	same	0.75
shell diameter, m	2.0	1.717
overall height, m	5.0	7.2
<u>Tube side conditions</u>	fuel salt	fuel salt
inlet temperature, °F	1300	1300
outlet temperature, °F	1050	1050
mass flow rate, kg/s (lb/hr x 10 ⁶) ⁶	1164 (9.24)	2955 (23.4)
<u>Shell side conditions</u>	coolant salt	coolant salt
inlet temperature, °F	850	850
outlet temperature, °F	1150	1150
mass flow rate, kg/s (lb/hr x 10 ⁶) ⁷	1495 (11.9)	2218 (17.6)

⁶ The MSBR flow rate cannot be scaled-down because the fuel salt for the ABC is essentially LiF-BeF₂ whereas the MSBR fuel salt is LiF-BeF₂-ThF₄-UF₄.

⁷ Because the same coolant is used, the MSBR flow rates were scaled down to obtain the ABC flow rates.

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2.3 steam system

The only available steam generator design data for a molten salt steam generator are those for the small 121 MW horizontal U-tube, U-shell heat exchanger for the MSBR that employs a supercritical steam cycle with 1000°F steam at 3500 psia. The tube length for this steam generator is 23.29 m (76.4 ft), and the inner shell diameter is 0.464 m (18.25 inches). For illustration purposes this steam generator will be scaled up to 375 MW but limiting the shell diameter to 1.50m.

Using the MSBR 121 MW steam generator as reference, the total length of tubing for a 375 MW ABC steam generator is:

$$(375/121) \times 393 \times 23.29 \text{ m} = 28,366 \text{ m.}$$

The number of tubes that can be accommodated in a 1.5 m diameter shell in relation to the 393 tubes accommodated in the 0.464 m shell is:

$$393 \times (1.5/0.464)^2 = 4115 \text{ tubes.}$$

The tube length for the ABC steam generator is then:

$$28,366/4115 = 6.893 \text{ m.}$$

The overall length of the steam generator has to account for the curvature and the height of the plena. The tube length consist of two straight sections and one curved section. Assuming an average curvature radius of 0.6 m, the length of the straight sections is 2.5 m, and the height of the bundle is 3.1 m. Assuming a radius of 0.75 m for the inlet and outlet plena, respectively, yields an overall height for the steam generator of 4.6 m.

For the supercritical steam cycle of the MSBR steam reheaters had been added which were sizeable structures (603 tubes, 13.2 ft long tubes, shell ID of 20.25 inches, and overall height of 15 ft).

Using a different steam cycle without reheat requires a new design. It is proposed to look closer at a vertical U-tube design (like CANDU) that would include primary and secondary moisture separators and may yield a height of 15 m for the ABC system. It is also proposed to evaluate a helical coil steam generators.

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The steam generator design and estimate of flow rates is fairly complex. To illustrate some of these difficulties properties of superheated steam at 900°F are cited for 2400 and 3500 psia.

	2400 psia	3500 psia
Thermal conductivity, Btu/(hr-ft- F)	0.052	0.062
Viscosity, lb/ft ² x 10 ⁶	63	67
Specific volume, ft ³ /lb	0.285	0.176
Specific heat, Btu/(lb-°F)	0.74	0.91
Relative film resistance to heat transfer	1.9	1.0

This illustrates why the high-pressure supercritical cycle has such an attraction from a heat transfer perspective. The film coefficient for heat transfer in the 3500 psia system is about twice that in the 2400 psia system and, therefore, the surface area required would be significantly less.

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TABLE 6: IHX FEATURES FOR THE MSBR AND CRBRP

IHX FEATURES	MSBR	CRBRP
thermal duty, MW	556.5	325
overall length, ft	24	52.1
shell diameter, ft	5.7	8.8
tube length, ft	24.4	25.8
# of tubes	5803	2850
overall heat transfer area, ft ²	13,916	11,810
overall heat transfer coefficient, Btu/hr-F-ft ²	784.8	1,374
primary flow, 10 ⁶ lb/hr	23.4	13.82
intermediate flow, 10 ⁶ lb/hr	17.8	12.78
primary system		
hot leg temperature, °F	1300	995
cold leg temperature, °F	1050	730
intermediate system		
hot leg temperature, °F	1150	936
cold leg temperature, °F	850	651

A3.0 Drawings for System and Component Layouts⁸

A series of drawings has been prepared to better illustrate the ABC system design concept.

Not addressed here are the accelerator-molten salt system interface and the accelerator with the target assembly. Cited often as a complication for an ABC system are the penetrations required for the beam tube and target assembly. However, the penetration through the vessel to install the target assembly is in principle not different from the penetrations required for control rods drive lines in a reactor.

What should help greatly in the design of the target assembly and especially the target window, as well as the seals for the penetrations is the very low operating pressure for the primary system. The primary system operating pressure is affected by two considerations, (1) the need for a high pressure to avoid fuel boiling, and (2) the primary system pressure drop.

Because the vapor pressure of molten salts is very low even at high temperatures, a pressurization of the primary system is not required to delay fuel boiling.

The pressure drop in the core and piping should be minimal. Assuming the same tubing dimensions and layout as in the MSBR IHX should more than cut in half the pressure drop in the IHX to less than 50 psi because the ABC IHX tubing is less than half the length for the MSBR IHX. Further reductions, if desired, can be obtained by opening up the pitch and increasing the tube diameter. Under those conditions it is conceivable that the primary system pressure is significantly less than 100 psia which greatly simplifies not only the target design but also the primary system design and should also result in significant cost savings.

These benefits, however, can only be realized if the secondary and steam system pressures are not excessively high so that any tube rupture in these two system does not lead to damage in the primary system. This is an illustration for the correctness of the safety-first design approach. The scope in the design has to cover not only success and reliability of operation under normal operating conditions, but also safety under accident conditions.

⁸ Extensive use will be made of the work reported in "Preconceptual ABC Design Definition and System Configuration Layout" and submitted to LANL in March 1995.

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Drawings for the blanket system, intermediate system, steam system, gas sparging system, off-gas system, fuel drainage system, containment system and overall plant layout have been prepared. The drawings are roughly on scale. At the preconceptual design level it was not considered meritorious to tie down dimensions to be within 10 or 20% of the final dimensions because not a single component has been designed yet.

Fig. 4 shows the flow sheet for the ABC NSSS. Fig. 5 shows schematically the heat transport system hydraulic profile with its staggered thermal centers.

3.1 core

Fig. 6 shows a schematic drawing of the ABC blanket configuration/module in a reflected system. Cold fuel coming from the IHX enters, through the outer of two concentric pipes, the gap between core barrel and vessel, thus keeping both structures at inlet temperature. The fuel flows downward, passes through holes in the lower core support plate to enter the inlet plenum. From here the fuel flows upward through holes in the lower axial reflector/shield into the core region. The hot fuel proceeds through a gap formed by the radial reflector and the upper axial reflector/shield to the inner of the concentric pipes on its way to the IHX. Each of the modules has two outlet/inlet pipes. At the bottom of the vessel is a drainage hole that is kept closed by a freeze plug located underneath that is not shown in the figure.

3.2 primary and intermediate system

Fig. 7 shows the inside of the primary system cell that contains the blanket module, two fuel pumps and two IHXs. To better illustrate the elevations and sizes of the components they were rotated such that a single train core-pump-IHX is visible. In reality, these components are not fully visible in a side view as shown in Fig. 18.

The arrangement is such that each component can be withdrawn from this cell in a straight-pull motion. Not shown is the drainage line for the coolant from the IHX that facilitates complete drainage by gravity. Fuel drainage by gravity is complete and from the bottom drain line in the core vessel. The fuel pump is in the hot leg at the highest elevation. The fuel pump motor is located outside the containment. The fuel pump bowl is about 1.80 m in diameter. The concrete shielding surrounding the cell is about 2.50 m thick. This cell is a safety-related structure. The inside of this concrete structure contains a Hastelloy N (or steel) liner with insulation material between the liner and the concrete. The cavity is filled with NaF-LiF-BeF₂ molten salt. Should NaF-

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BeF_2 become the carrier salt for the fuel, it would be both coolant salt and also the salt in the cavity.

Fig. 8 shows the cell containing the IHTS isolation valves and the secondary (coolant salt) pump. It needs to be determined whether it is necessary to fill this cell with a salt like NaF-BeF_2 to contain leaks, should there be any, and serve as a heat sink. The pump and IHTS valves can be removed from this cell in straight-pull motions.

Underneath this cell is a salt storage area that has not been specifically defined yet. This cell, like all the others is surrounded by a 2.5 m thick concrete wall. It is a safety-related structure. The walls are stainless-steel-clad and heavily insulated. Should the cell be filled with salt, the use of Hastelloy N will be considered.

On top of the primary and secondary system cell and the IHTS and coolant pump cell is a containment building that contains maintenance enclosures and a polar crane together with an auxiliary crane.

3.3 steam system

The steam system is schematically shown in Fig. 9. The steam generator, a drain tank, and a set of MSIV are contained in a separate containment that has 2.5 m thick walls. It is a safety-related structure. Next to this containment building is the feedwater heating bay that contains another set of MSIV and feedwater heating equipment. The Feedwater Heating Bay is built to high industry standards. It is not a safety-related structure.

It needs to be determined whether another set of isolation devices is required to protect the IHX in case of a steam generator tube rupture. The MSBR used rupture disks that were located at the secondary-salt outlet of each steam generator. Furthermore, quick-acting block valves at the inlet and outlet of each steam generator were used to isolate the steam generator.

At present, one set of MSIV is placed inside the steam generator cell, and another set is located in the feedwater bay area. This might very well be an overdesign where one set of valves could be saved. The intent here was to provide a very high reliability through redundancy to isolate the steam generator to significantly lower the probability that radionuclides could leak from the primary system into the secondary system, from there into the steam system and ultimately the into the feedwater bay area.

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On top of the steam generator cell is the steam generator bay that provides a work area for the components inside the steam generator cell.

3.4 gas sparging system

Fig. 10 shows a schematic flow diagram for a gas sparging system, including both the bubble generation and removal bypass. This figure describes the MSBR system as shown in Fig. 3.37 of ORNL-4541. Because gas sparging has less importance for the ABC system than the MSBR, a less complex system can most likely be used for the ABC system.

3.5 off-gas system

Fig. 11 shows a schematic flow diagram of the MSBR off-gas system, as shown in Fig. 7.1 of ORNL-4541. Several options were considered for the ABC system. Option 1 shown in Fig. 12 eliminates the bubble generator system and the long-delay Xe holdup beds. The off-gas leaves the pump bowl through an entrainment separator and proceeds to the fuel drain tank. After a 2 hr holdup, the off-gas enters the 47-hr Xe holdup system and then returns through a chemical trap and monitor to the primary system.

Fig. 13 shows a second option that maintains the bubble generator system shown in Fig. 11 but eliminates the long-term Xe holdup. After the gases leave the fuel salt drain tank they pass through a filter to remove particulate matters. After the 47-hr holdup all gases are returned to the primary system.

Fig. 14 shows the recommended off-gas system. It is very similar to option 2 shown in Fig. 13 but eliminates the particle trap and leaves open the holdup times in the fuel drain tank and in the Xe holdup system.

Fig. 15 shows a cross section of one bank of Xe holdup charcoal beds similar to the MSBR. Hairpin tubes are filled with charcoal through which the off-gases are directed. The heat is rejected to water that converts to 230 F steam which is condensed in a condenser and the condensate is directed back into the holdup bed.

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3.6 fuel drainage system

Fig. 16 shows a schematic flow diagram for the ABC primary drain and heat removal system using NaF-BeF₂-LiF as coolant. The decay heat produced in the fuel drain tank is passively removed by a coolant salt and transported to a water-salt heat exchanger. The heated water is transported passively to a water-air heat exchanger located in a heat reject stack.

In case NaK is used as coolant in the fuel drain tank, the ultimate heat sink, according to a design study for the MSBR could be a water tank where, not unlike in case of the charcoal beds, water is brought to a boil and then either returned after passing through a condenser, or released to the outside and makeup water is supplied.

3.7 containment system

The containment system is shown in Figs. 17 and 18. Fig. 17 shows the relation to each other of the mainline systems. To show core vessel, pump and IHXs, those components had to be rotated out of their position because otherwise they would not have been all visible. Fig. 18 shows a top view of the mainline systems. The dimensions are:

core vessel diameter:	4.80 m
core diameter:	2.10 m
reflector thickness:	1.25 m
IHX vessel diameter:	2.00 m
pump bowl diameter:	1.80 m
containment wall diameter:	2.50 m
approximate primary cell outer diameter:	16 m
approximate primary cell inner diameter:	11 m
approximate primary cell height:	9 m
approximate building height on top of cells:	12 m

At this stage of the ABC design development no attempt was made to optimize the design and layout. Basic design analyses and decisions still have to be made that will affect the sizing of the cells and perhaps even their relationship to each other, as well as permissible module size, power density and system requirements. It is most likely that the reactor building wall thicknesses (the structure on top of the cells) can be significantly reduced, perhaps even to less than the MSBR building dome/wall thickness that was 3.5 ft. Inside the concrete building structure is a 1 cm thick liner made of carbon steel which acts as a sealing membrane to permit the achievement of

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less than one inventory loss per year. (The requirement for the MSBR was 0.1% per 24-hr period. A confinement building such as that proposed for the MHTGR had an expected leakage rate of one inventory per day).

It should be noted that none of the cell or building dimensions are excessive. For example, the MSBR reactor cell (inner) diameter was 22 m which is twice the diameter for the ABC cell. All MSBR structures were above ground with a containment building height of 57 m. The total height above ground for the ABC containment building can be less than 8 m.

It is assumed that the accelerator beam will enter the reactor building below grade. How far below grade it has to be directed has to be based on well-justified requirements because it will affect building cost.

The primary system cell and the cell containing the IHTS isolation valves and coolant pumps are under the primary and secondary system containment bay which is another containment structure. On top of the two cells is a maintenance enclosure. This enclosure is a thin-walled containment vessel designed primarily to contain airborne contaminants during maintenance operations. It has access ports over the vessel, pumps and IHXs.

As shown in Fig. 18, the primary system cell, IHTS-coolant pump cell and steam generator cell are safety-related structures. The feedwater heater bay requirement is to be built in accordance with high quality industrial standards.

3.8 overall plant layout

Fig. 17 and 18 already showed a single train configuration. Fig. 19 shows how two modules will share one off-gas system and one fuel drain system. Off-gas system and fuel drain system are placed next to each other. Shown is also the stack area assuming the coolant in the drain system is a molten salt.

No attempts was made at this stage of the design development to show hot cells, offices, areas for salt processing and storage, etc. These are not critical areas in regard to design viability albeit the cost for these facilities is not insignificant.

A4.0 Impact of Power Density and Module Power on Containment Size

To assess the impact of changes in power density, power level and also height-to-diameter (L/D) ratio, parametric studies were done with the following assumptions:

- power density varies between 75 - 300 kW/l
- module power was varied between 500 and 1000 MWt
- L/D ratio was varied between 0.6 to 1.3
- reflector thickness of 125 cm
- core barrel thickness of 2 cm
- gap between barrel and vessel wall of 4 cm
- vessel thickness of 3 cm

4.1 impact of power density

Fig. 20 shows the relationship between power density and vessel diameter assuming a 750 MWt module power level and an L/D ratio of 1.0. Increasing the power density by 400% decreases the vessel diameter by less than 20%. Increasing the power density from 100 kW/l to 150 kW/l decreases the vessel size from 4.80 m to 4.53 m, or by 27 cm.

This is too small a change to be reflected in the containment size.

4.2 impact of module size

Fig. 21 shows the relationship between vessel diameter and power level assuming a power density of 100 kW/l and an L/D ratio of 1.0. Doubling the module power level increases the vessel diameter from 4.53 m at 500 MWt to 5.01 m at 1000 MWt. Going from 750 MWt to 1000 MWt in module power increases the vessel diameter from 4.80 m to 5.01 m, or by a mere 21 cm.

This is too small a change to be reflected in the containment size.

4.3 impact of L/D ratio

Fig. 22 show the relationship between vessel diameter and L/D ratio, assuming a module power of 750 MWt and a power density of 100 kW/l. Flattening the core from an L/D os 1.0 to an L/D value of 0.6 increases the vessel diameter from 4.80 m

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to 5.19 m, or by 39 cm. Increasing the core height so that the L/D ratio increases from 1.0 to 1.3, decreases the vessel diameter from 4.80 m to 4.62 m.

These changes are too small to be reflected in the containment size. This means that the L/D ratio should be determined largely based on physics considerations.

Fig. 1: Flowsheet for the Supercritical-Pressure Cycle with Feedwater Heated by Mixing

MODIFIED LOEFFLER
CYCLE FOR
FEEDWATER HEATING

Fig. 2: Modified Loeffler Cycle for Feedwater Heating

Fig. 3: Vertical U-Tube Steam Generator Similar to CANDU

Fig. 4: Flowsheet for the ABC Nuclear Steam Supply System (NSSS)

Fig. 5: Heat Transport System Hydraulic Profile

Beam Tube/Target Assembly

Fig. 7: Primary System Cell

Fig. 8: Cell for IHTS Isolation Valves, Coolant Salt Pump, and Salt Storage Area

Fig. 9: Steam Generator Cell and Feedwater Bay

Fig. 10: Schematic Flow Diagram for the Sparging System

Fig. 11: Schematic Flow Diagram: MSBR Off-Gas System

Fig. 12: Schematic Flow Diagram for the Off-Gas System: Option 1

Fig. 13: Schematic Flow Diagram for the Off-Gas System: Option 2

Fig. 14: Schematic Flow Diagram for the Recommended ABC Off-Gas System

ABC OFF-GAS SYSTEM

Cross Section of one Bank
of 47-hr Xe Holdup Charcoal
Bed Similar to MSBR

Fig. 15: Off-Gas System Charcoal Bed

Fig. 16: Schematic Flow Diagram for the Fuel Drain System

Schematic Flow Diagram
ABC Primary Drain and
Heat Removal System
Using NaF-LiF-BeF₂
as Coolant

Fig. 17: ABC System Nuclear Steam Supply System

Fig. 18: Plan View of 750 MWt ABC Module Train

Fig. 19: 2x750 MWt ABC Module Trains with Shared Off-Gas and Drainage System

ABC VESSEL DIAMETER VS CORE POWER DENSITY

750 MWt MODULE POWER, $L/D = 1.0$

Fig. 20: ABC Vessel Diameter as a Function of Core Power Density

ABC VESSEL DIAMETER VS POWER LEVEL

100 kW/l POWER DENSITY, $L/D = 1.0$

Fig. 21: ABC Vessel Diameter as a Function of Power Level

ABC VESSEL DIAMETER VS L/D RATIO

750 MWt MODULE POWER, 100kW/l

Fig. 22: ABC Vessel Diameter as a Function of L/D Ratio