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HIGH-LEVEL ASSESSMENT OF LANL ABC DESIGN

**BASED ON INFORMATION PRESENTED BY LANL AT THE
JASON JANUARY 18, 1994 AND
DOE NP/NE FEBRUARY 17, 1994 MEETINGS**

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SUMMARY OF RECOMMENDATIONS

In the following, the recommendations made in this report are summarized.

An annual weapon's grade Pu disposition goal should be stated and related to the amount of Pu that needs to be disposed of.

It needs to be determined to what an extent it is possible to destroy Pu without building up any new Pu, i.e., how realistic this goal is. The strong positive Doppler coefficient for a Pu core might require the addition of some fertile material to ensure a negative Doppler coefficient. This in turn will affect the net Pu disposition rate.

If a fertile material is required throughout the life of the ABC to ensure a negative Doppler coefficient, the difference between the molten salt ABC and other reactors in regard to Pu disposition is not a principled difference anymore but one of degree. A rationale has then to be developed that explains why "x" kg production of fissile material are acceptable but "y" kg are not.

It is important to determine how a requirement for electricity production will impact on the ABC design choices. It is conceivable that DOE will not insist on electricity generation. In this case advantage has to be taken in terms of design simplifications and relaxed operating conditions.

I had proposed earlier to perform a ranking of the different systems that were investigated at LANL based on a MUST/WANT methodology that has been used in the past to rank different reactor concepts. Such a ranking would have provided a logical framework for the decision-making and identified clearly the criteria that each design would have to meet to be considered at all (MUST criteria), and then use WANT criteria to rank the different candidates. If the decision stands not to perform such a ranking, as a minimum the various designs should be documented.

A more detailed and technically unambiguous statement should be prepared as to why the molten salt system was selected over the aqueous homogeneous systems, and also over other systems (e.g., IFR-types, LWRs, CANDU, HTGRs with prismatic or pebble-type fuel).

I recommend not copy MHTGR design features for the ABC without fully understanding the basis on which the MHTGR features were selected. What works well for the MHTGR might not work at all for the ABC.

Be prudent in selecting the initial operating conditions. The ABC system faces many challenges. To go beyond what a few hundred man-years of molten salt reactor design experience produced in form of the MSBR should have a solid backup. The latest data on inlet and outlet temperatures seem more reasonable than data presented

before but it is not stated by which process they were changed.

It should be kept in mind that the operating conditions relate to inlet and outlet temperatures that are averages. However, what limits performance are usually not average values but hot channel data, hot streaking and worst condition temperatures.

Design detail should only be presented if it can be technically justified; if there is a compelling reason to show design detail without a technical basis to support it, state clearly that the design detail is shown for illustration purposes only.

Develop design and safety requirements documents to establish a framework for the design. At the expense of several million dollars each, such documents had been developed for the advanced LWRs, MHTGR, ALMR and NPR MHTGR and HWRF. This is not a luxury but a necessity.

Base the design on a higher power density, for example 20 w/cc instead of 6 w/cc.

Develop a rationale for the 500 MWt module power and develop a rationale for the need of six IHX to serve one 500 MWt blanket, i.e., using a 83 MWt IHX rather than larger units. The scaling laws apply over a wide range of component sizes. Typically, a 160 MWt IHX costs much less than two 80 MWt IHX. When the system is laid out, these matters need to be considered. It is also important to consider plant availability and partial load operation with one or more IHX isolated. How an IHX can be isolated, both in terms of fuel salt and secondary salt supplies, needs to be evaluated.

I don't know of any reactor that could not be built because of a lack of proper flow mixing. There are many design options available to mitigate the consequences of hot streaking and to provide for better mixing in the first place. To focus at this stage in the design evolution on flow mixing is premature considering the lack of a defensible basis for the overall design.

It needs to be investigated whether the inlet flow should first be directed along the vessel wall in a downward flow directed toward an inlet plenum. This would ensure that the vessel, under normal operating conditions, is kept at a lower temperature. It would also keep down heat losses to the cavity.

A better containment design might be a vessel-inside-a-vessel design that I had mentioned before where a blanket vessel is surrounded by another vessel/tank that contains the steam generators that are surrounded by coolant.

To provide the required lateral blanket restraint, a core barrel or similar structure should be used. This would also mechanically decouple the blanket region from the rest inside the vessel.

Functional requirements for decay heat removal need to be developed first before any hardware design begins. The fuel drainage system should be part of any decay heat removal system. The use of DRACS and IRACS should be considered, especially when a vessel-inside-a-vessel concept is pursued that I proposed earlier. One of the major design challenges for any molten salt system is to keep the salt temperatures high enough to avoid salt freezing (note: the melting temperature for the MSBR fuel salt is about 500°C which is the same as the coolant outlet temperature for the ALMR), and the maximum allowable peak temperature is 1200°C (which is 400°C below that for the MHTGR).

Based on the premise that DOE will not anymore support generic technology development work the ABC Technology Development Program has to focus on support of the design and licensing of an ABC system as the sole justification.

Avoid statements such as "ABC Demonstration at LAMPF can [emphasis added] address important technical issues" because this implies that the experiment has a meaning in itself and what it can do for the project will be determined later.

Do not separate demonstration-scale experiment and techbase activities because the experiment has to be an integral part of the technology development program.

Develop a logic for a technology development program.

For the Technology Development Program Plan (TDPP) I suggest to begin with the identification of Design Data Needs (DDN) and use them as the basis for the TDPP. A similar format was very useful for the MHTGR and NPR programs.

To give at least a minimum level of assurance that a TDPP is complete, one has to do some functional analysis (top-down design approach) and identify design requirements.

The DDN format could look like this:

1. Title
2. Requirement or Design Feature Requiring Experimental Data or Validation Testing
3. State of the Art
4. Design and Analysis Data Needs
5. Designer Alternatives
6. Required Experimental Work
7. Consequences of Non-Accomplishment and Fallback Position
8. Schedule Requirements
9. Priority

The format of the outline can be modified for as long as all the topics are addressed.

The development plan has to address issues related to design, licensing, fabrication, operation, recovery and possible even decommissioning of the plant.

I cannot overemphasize the importance of a clear and logical structure for design, analysis and technology development work. This might delay the beginning of the "real" work, but it pays off to plan in detail the work that needs to be done and to have good reasons why it has to be done.

At issue is going from a 1/25th scale experiment to a full-size module, i.e., going from a 20 MWt experiment to a 500 MWt module. I do not think that this extrapolation can be done without an intermediate step. Very often a factor 10 in size is the largest extrapolation considered acceptable. It should be noted that the MSBR development program relied on the ARE and MSRE experiments and had as an essential part the design and operation of a Molten Salt Test Reactor of sufficient size so that experience could be obtained for components that are full scale for a demonstration reactor.

Before any time and cost estimates are made on designing and building a facility the availability of suppliers for the required materials, systems and components has to be investigated.

... to a logic and top-down approach.

I recommend to include conventional control rods in the reactivity control system. Even if the feasibility of accelerator control had already been demonstrated - and it has not been demonstrated so far - insisting on not using conventional control rods at least as a backup can be expected to be extremely expensive in the regulatory arena. LANL has been involved for many years not only in regulatory research but also in licensing review and knows that the time for licensing review can be increased by many years, with no assurance of success, whenever major novel features are proposed that have a profound impact on plant safety.

While graphite is a good moderator, it also introduces significant problems (shrinkage, swelling, fission product adsorption, replacement issues). Therefore, other moderators besides graphite need to be evaluated, too. A system without moderator should also be investigated.

HIGH-LEVEL ASSESSMENT OF

LANL ABC Design

**Based on Information Presented at the
January 18, 1994 JASON and
February 17, 1994 DOE NP/NE Meetings**

1.0 INTRODUCTION

During my visit to LANL in January 1994 (JASON review) I already commented on the ABC design that was presented in the first and second dryruns. I had explained then that some of the major design features and operating conditions selected for the ABC had no clear technical basis and some of the design features might even introduce feasibility problems. Some of the operating condition data presented by LANL at the DOE-HQ meeting on February 17 seem more reasonable but the design detail I had questioned was still contained in the viewgraph handout even though it was not presented.

I will try to explain why, in my view, it is better at this stage of the design to show only schematic presentations of the design, emphasizing the preliminary nature rather than showing design details that, in many instances, cannot be justified and, in all instances, are not the results of a disciplined engineering design process. There are many conceptual decisions that still need to be made and a good basis has to be developed for them first.

My design review is based on the information provided at the dryruns for the JASON meeting in January, the viewgraph handouts describing the design from the JASONs meeting in January and the DOE NP/NE meeting in February, as well as discussions I had with LANL staff during my January visit.

2.0 OVERALL DESIGN APPROACH

At the outset of any design description has to be a statement of the objectives this design is expected to accomplish. However, nowhere in the JASONS and DOE presentations is stated quantitatively what the ABC design is expected to accomplish. I realize that the Pu disposition targets are moving but there must be a numerical goal that has to be met by a design. The Pu disposition issue cannot be solved generically.

Recommendation: State an annual weapon's grade Pu disposition goal and relate this goal to the amount of weapon's grade Pu that needs to be disposed of.

Note: It needs to be determined to what an extent it is possible to destroy Pu without building up any new Pu, i.e., how realistic this goal is. The strong positive Doppler coefficient for a Pu core might require the addition of some fertile material to ensure a negative Doppler coefficient. This in turn will affect the net Pu disposition rate. More information on this topic will be provided in a separate report on analysis tasks.

If a fertile material has to be present throughout the life of the ABC to ensure a negative Doppler coefficient, the difference between the molten salt ABC and other reactors in regard to Pu disposition is not a principled difference anymore but one of degree. A rationale has then to be developed that explains why "x" kg production of fissile material are acceptable but "y" kg are not.

It is important to determine how a requirement on electricity production will impact on the ABC design choices. It is conceivable that DOE will not insist on electricity generation. In this case advantage has to be taken in terms of design simplifications and relaxed operating conditions. This is not an easy task because a molten salt system always requires high temperatures.

For several years, LANL has been investigating aqueous homogeneous systems mostly for waste transmutation, with the molten salt system being the backup, with focus on Pu disposition only as of lately. I don't know whether this work has ever been documented except in form of viewgraphs and a few brief memoranda on specific topics.

Recommendation: I had proposed earlier to perform a ranking of the different systems that were investigated at LANL, based on a MUST/WANT methodology that has been used in the past to rank different reactor concepts. Such a ranking would have provided a logical framework for the decision-making and identified clearly the criteria that each design would have to meet to be considered at all (MUST criteria), and then use WANT criteria to rank the different candidates. If the decision stands not to perform such a ranking, as a minimum the various designs should be documented.

An attempt to justify the selection of the molten salt technology has been made on two viewgraphs, one showing solid fuel systems and the other one showing liquid fuel systems. However, it is not clear why this is the road map that necessarily leads to the molten salt technology.

There are important omissions on the viewgraphs and some of the characterizations of reactor concepts are not correct. Here are some examples:

1. missing from the list under "solid fuel systems" are:

- gas-cooled reactor with prismatic fuel
- gas-cooled reactor with pebble fuel (e.g., AVR, THTR)
- CANDU reactors

Gas-cooled reactors with prismatic fuel do not permit continuous refueling. However, GA has proposed an MHTGR for Pu disposition and discussed such a design with the Russians.

The AVR-type and CANDU reactor types have continuous refueling. The AVR operated successfully for 20 years. The THTR operated only for a brief period of time and was then permanently shut down, with technical reasons playing only a minor role in this decision. An 8 MWt pebble bed reactor is under construction in China. The ≈ 150 MWt Russian MARR One is also based on pebble fuel. It is expected to operate in a gas-turbine cycle and achieve thermal conversion efficiencies of 45% using essentially off-the-shelf technology.

Some of the intrinsic problems with pebble bed reactors relate to the very low core power densities which, however, can be overcome by using a cross flow arrangement that would lower the pressure drop while at the same time permit a higher core power density.

CANDU reactors, that are synonymous with continuous refueling, operate world-wide for over 20 years and have the highest capacity factors of any

reactor type. CANDU-3 is currently under licensing review by the NRC.

2. missing from the characterization of the solid fuel reactors LWR and LMR is the difficulty to control large excess reactivities when Pu is burned and no new fuel built up.
3. not addressed are systems that could use Pu in connection with Th (no Pu buildup) and a small amount of U to eliminate the possibility of chemically separating ²³³U from ²³⁸U
4. the statement that there are "safety concerns with liquid metal coolant" can easily be misunderstood; sodium technology is well developed, practiced for about 40 years and the handling of sodium is not a safety concern anymore. What is a concern in an LMR is the loss of sodium accident and the associated reactivity increase. However, this problem is tackled in the design and is not a sodium coolant concern.
5. it is not clear why the "limited thermal-to-electric conversion efficiency" in LWRs leads to the molten salt technology. The thermal efficiency of LWRs is not an obstacle for their deployment. At this time, the goal for any new reactor type is to meet the electricity cost for LWRs. The conversion efficiency becomes a non-issue when DOE does not want to sell steam to a utility.
6. leaving out pebble bed and CANDU systems but including a helium-cooled liquid fuel system is somewhat confusing; to cite as reason for not pursuing the latter system that it couldn't meet blow-down criteria is only one of the reasons (and in my view, a minor one) why this is not an attractive concept.
7. molten salt technology goes back to the 1940's and the 1954 ARE had already molten salt as a carrier. I.e., the path to molten salts did not begin with gaseous fuel, as suggested on one of the viewgraphs.

Recommendation: A more detailed and technically unambiguous statement should be prepared as to why the molten salt system was selected not only over the aqueous homogeneous systems, but also over other systems (e.g., IFR-types, LWRs, CANDU, HTGRs with prismatic or pebble-type fuel).

Note: At the DOE meeting, reference was made to an alleged problem for the MHTGR with a coolant blowdown. The fact is that the MHTGR is designed specifically for successfully dealing with a blowdown (in MHTGR lingo, a depressurized conduction cooldown accident). This is one of the accidents considered, at the conception of the MHTGR design, when operating conditions and module power as well as core configuration are selected. More about this later.

In regard to the current ABC design, I had recommended earlier that the ABC design should initially be based on an existing molten salt reactor design (e.g., the MSBR) and whatever modifications are made, they should be justified. The MSBR design reflects hundreds of man years of extensive molten salt reactor experience most of which is not even written up in reports. Instead, the ABC design seems to be based more on the MHTGR design, however, without fully appreciating why and how the MHTGR design feature selections had been made. When I reviewed the ATW molten salt design in 1992, I commented on the similarity of the ATW system and the MHTGR and pointed to some of the problems.

Note: There are many good reasons for deviating from the MSBR design, most of them are related to safety issues, in particular the greater emphasis placed today on inherent and passive safety. For example, the MSBR would be designed today with a greater distance between the thermal centers in the core and IHX to better promote natural convection flow.

I had recommended on several occasions that the ABC design should be developed in an integrated, top-down approach, beginning with overall objectives, translating the objectives into goals, requirements and functions, and proceeding with the selection of systems, structures and components based on trade-off studies. In recent years, this approach has been successfully used in the development of the MHTGR. The ABC system has to meet requirements that are different from those for a MHTGR. The ABC system is a different reactor with a different characteristics.

Recommendation: Do not copy MHTGR design features for the ABC without fully understanding the basis on which the MHTGR features were selected. What works well for the MHTGR might not work at all for the ABC.

The ABC design approach currently employed underlines the need for a structured integrated design approach that avoids all the trappings of the current ad hoc approach.

3.0 OPERATING CONDITIONS

Several sets of operating conditions are cited without explaining the basis for their selection or stating which set will be used for the ABC. One JASON viewgraph showing *ABC Operating Parameters* lists 971°K (698°C) as the temperature of the fuel salt entering the heat exchanger, and 775°K (502°C) as the temperature of the fuel salt leaving the heat exchanger.

The JASON viewgraph describing the *primary heat exchanger* lists as fuel salt inlet temperature 750°C and as fuel salt outlet temperature 598°C. In the former case, the coolant temperature rise is 196°C, in the latter it is 152°C; the difference in coolant outlet temperatures for the two sets of operating conditions is 52°C.

The respective blanket inlet and outlet temperatures shown at the DOE meeting four weeks later are 850°K (577°C) and 1000°K (727°C); the coolant salt inlet and outlet temperatures from the IHX are listed as 770°K (497°C) and 840°K (567°C).

Compare this to the MSBR (MSRE) and ALMR temperature conditions (°C):

<u>Blanket Conditions</u>	<u>MSBR (MSRE)</u>	<u>ABC1</u> (1/94)	<u>ABC2</u> (1/94)	<u>ABC3</u> (2/94)	<u>ALMR</u>
Fuel Inlet Temperature	566 (632)	502	598	577	338
Fuel Outlet Temperature	705 (654)	698	750	727	485
Coolant Temperature Rise	139 (22)	196	152	150	147

No rationale is presented for the ABC temperature conditions. At the least, there should be a justifiable minimum margin above the melting temperature for the fuel salt inlet temperature and another margin for the outlet temperature as well as the coolant temperature rise. In terms of coolant temperature rise, there are many considerations that need to be carefully evaluated. The MSBR value is consistent with that for liquid metal reactors. The ABC1 and ABC2 data are very advanced and can be expected to introduce potential design problems. The ABC3 data seem to be defensible but no basis is provided as to why the data were changed.

It is well known that pressure drop, pumping power and component size can be reduced in proportion to the coolant temperature rise. However, how to achieve those goals is another matter. It is also well known that increasing the coolant temperature rise can cause severe thermal stress and creep problems throughout the primary system, aggravated bowing, more severe hot streaking, mixing problems, more severe hot spots, etc. In the LMFBR Follow-on studies of the late 1960's, sodium coolant temperature rises of 325°F (180°C) were assumed, but when a more detailed design analysis began, those temperature rises were significantly reduced. For the ALMR, they are now typically in the 250°F (139°C) range. Major areas of concern are the outlet plenum and all structures in it, the vessel nozzle, and IHX.

It should be noted that the only experience base for the IHX design is the 8 MWt MSRE IHX that operated successfully under the following conditions (°C):

	Fuel Salt	Coolant Salt
Outlet temperature	654	546
Inlet Temperature	632	579
Temperature Rise	22	33

This is the only salt-salt IHX ever built and operated. The operating experience extends over 4 years.

Recommendation: Be prudent in selecting the initial operating conditions. The ABC system has many challenges. To go beyond what a few hundred man-years of molten salt reactor design experience produced in form of the MSBR should have a solid backup. The latest data on inlet and outlet temperatures seem more reasonable than the data presented before, but it is unknown to me by which process they were changed.

Keep in mind that the operating conditions relate to inlet and outlet temperatures that are averages. However, what limits performance are usually not average values but hot channel data, hot streaking and worst condition temperatures.

4.0 PRIMARY SYSTEM LAYOUT

On one of the viewgraphs is noted that the ABC design presented to the JASONS is an example but not a point design. However, this prudent thought is lost later when some design details are presented that go well beyond the illustration of a technical approach to dispose of 50 tons of Pu in 20 full-power years.

[A statement like the last one could very well be an objective for the ABC system; however, from the presentation it was not clear if this was an example or a program objective.]

Recommendation: Present design detail only if it can be technically justified; if there is a compelling reason to show design detail without a technical basis to support it, state clearly that the design detail is shown for illustration purposes only.

The primary system layout has to be related to the objectives for the ABC system and the design and safety requirements.

Recommendation: Develop design and safety requirements documents to establish a framework for the design. At the expense of several million dollars each, such documents had been developed for the advanced LWRs, MHTGR, ALMR and NPR MHTGR and HWRF. This is not a luxury but a necessity. I had proposed earlier to develop a "low-budget" version of those documents.

In the following two issues will be discussed, firstly the selection of power level and power density, and secondly the primary system layout with focus on the location of primary IHX vs blanket, hot leg vs cold leg pumps, primary salt pumps, IHX size, and the need for a well-defined flow path.

4.1 power level and power density

In the presentations of the molten salt ABC, no rationale has been presented for the selection of:

- a modular design as compared to a single vessel design and nowhere is stated what the modular design is supposed to accomplish;

Going from a large single vessel design to a multi-vessel modular design introduces cost penalties unless the use of small vessels permits certain cost benefits. Some of the cost penalties come from the use of multiple vessels instead of one vessel; multiple primary circuits (piping, valves,

pumps, instrumentation and controls, safety and protection systems, IHX, etc.) associated with each vessel rather than one primary circuit; and a larger containment building.

Some of the expected benefits come from the elimination or at least the reduction in on-site manufacturing requirements; factory fabrication with better QA and QC; system simplification because of enhanced passive safety; regulatory inspection at the factory; etc.

For commercial power plants an additional reason for going modular was to permit a better match between power demand and power supply that was expected from the sequential installation of modules spread over a period of time to match the demand. However, upon closer examination this perceived advantage is not as clear-cut as originally thought.

- 500 MWt as reference module power

The power that can be extracted from a single vessel is an important cost driver. The more power can be extracted the lower the power cost; however, the module power level is a highly constrained parameter that will be discussed in the following.

4.1.1 basis for MHTGR power density

The ABC presentations do not provide a basis for the selection of 6 w/cc as power density. This choice seems to be based on the MHTGR value of 5.9 w/cc without, however, fully appreciating the basis for this selection. (The MSBR design had a 22 w/cc power density.)

Let me briefly explain the basis for this power density in the MHTGR which will also illustrate the complexities and interactions in design and performance parameters that have to be considered:

The MHTGR power density of 5.9 w/cc was selected for the 350 MWt (450 MWt later) annular MHTGR core based on extensive safety and design analyses to ensure that under the worst accident conditions assurance is provided that **maximum fuel particle temperatures stay below 1600°C**. This fuel temperature limit is shorthand for many analyses related to compliance with quantitative regulatory criteria about radionuclide releases during severe accidents:

- Extensive fuel analyses had shown that for high-quality fuel compacts/particles radionuclides can be retained at levels higher than 99.99% for as long as peak fuel temperatures stay below 1600°C.

- The safety requirements and utility/user requirements for the MHTGR demanded that the (quantitative) regulatory accident criterion 10CFR100 and the EPA Protective Action Guidelines (PAGs) had to be met.
- Fuel analyses had shown that so-called "standard fuel particles" will not fail at temperatures below 1600°C; only particles with initial defects are expected to fail below 1600°C.
- If the initial fuel particle defect fraction is in the 10^{-5} range then both the 10CFR100 and PAGs can be met under accident conditions.

Analyses of a wide spectrum of accidents carried out in parallel showed that the highest possible fuel temperatures will occur during rapid depressurization cooldown events. This led to the formulation of the following objective:

In case of a rapid depressurized conduction cooldown accident in an MHTGR, heat conduction through the reactor vessel wall into the reactor cavity, from where it is removed passively by a Reactor Cavity Cooling System, has to be sufficient to limit peak fuel temperatures in the core to below 1600°C.

This objective is consistent with the selection of:

- an annular core with a large centrally-located graphite reflector region that acts as a large heat sink,
- a 258°C helium coolant inlet temperature and a 687°C nominal outlet temperature,
- downward coolant flow in the core,
- an inside vessel surface temperature under normal operating conditions of 258°C,
- use of a core barrel to restrain the fuel columns and heat transfer across the helium gap between core barrel and vessel,

taking into account:

- the change in thermal conductivity of the graphite blocks with fluence and temperature,
- changes in graphite swelling/shrinkage with fluence, burnup and temperature

- changes in the gap dimensions between graphite blocks,
- changes in vessel emissivity during life,
- (little) natural convection flow in the depressurized system (upward flow through center channels and downward flow through outer channels),
- movement of the thermal center of the core toward the upper plenum,
- a passive air-cooled reactor cavity cooling system that operates continuously.

Additional constraints came from the requirements that:

- the vessel size had to be compatible with factory fabrication capabilities and shipment of the completed vessel to the site which imposed a 24 ft diameter limit on the reactor vessel,
- no insulation on the reactor vessel will be used, and
- a conservative vessel fluence limit will be used that permits a 40 year vessel life.

This MHTGR design approach is different from that used for the Ft. St. Vrain plant where the reactor sits in a PCRV structure. Passive decay heat rejection to the environment is not possible for a PCRV design. In case of a depressurized conduction heatup event, massive fuel failures were expected and fission products would be contained inside the containment (confinement) that is equipped with cooling tubes inside the concrete to insure its integrity. However, such a depressurization event is more unlikely for the FSV plant than the MHTGR because a total system depressurization is a very low probability event due to the use of a PCRV that is expected to contain the coolant even under severe accident conditions.

The bottom line: even in case the MHTGR loses its coolant (very low probability event), fuel overheating does not occur because of the passive heat rejection characteristics. For FSV, fuel overheating was made an even lower probability event through the use of a PCRV and a special vessel cooling system that assured PCRV integrity. However, should the FSV reactor have ever depressurized, very high fuel temperatures were expected and also high fission product releases from the fuel.

This illustrates the importance of a clearly defined **safety approach**. In case of the FSV reactor that was designed and built during the early 70's, the emphasis was on **assuring the presence of coolant all the time** by the use of a PCRV. Losing all the coolant was, therefore, a very low probability event. However, should such a loss

ever happen, large releases from the core could be expected.

Note: During the 1960's, there was a concern that commercial HTGRs might not be feasible because of the large helium losses during normal operation and the very high accident temperatures in case of a loss of helium. The experience with PCRVs was very promising because of the containment of the helium even under severe accident conditions.

In the early 80's, various small HTGRs were designed, among them cylindrical and annular PCRV designs and steel-vessel designs, both in-line and side-by-side. The safety philosophy was then to assume that somehow the coolant can all be lost and ensuring that under those circumstances, no excessive radionuclide releases would occur. This led to the elimination of the PCRV designs and the selection of the side-by-side steel vessel design as the reference MHTGR design because of its capability to safely and passively cool the core even when all coolant is lost.

4.1.2 ABC power density of 6 w/cc

I don't see any similarity between the MHTGR and ABC design and performance parameters. Fuel salt temperatures are different from gas and MHTGR fuel temperatures; the vessel temperature are different; instead of having a large heat sink in the core (blanket) center, there is another heat source in the ABC system, the target assembly. What causes the high fuel temperatures in the MHTGR is the fact that the depressurization effectively removes the coolant and heat has to be transferred from the core to the cavity because the steam generator is not available as a heat sink. The MHTGR steam generator is located at a lower elevation (intentionally!) than the core vessel and is de facto thermally isolated so as not to expose the steam generator to the very high core temperatures in a depressurization accident (investment protection, not a safety issue). The highest metallic structure temperatures in the MHTGR under those conditions will be less than 650°C and 550°C for the core barrel and reactor vessel, respectively. The peak fuel temperature during such an accident is expected to rise over several days by about 400°C.

The ABC primary circuit is configured totally different in that the fuel/coolant outside of the blanket is contained in several large metallic structures (i.e., the IHX). The normal operating temperatures for the ABC vessel are already close to the MHTGR peak temperatures under the most severe accidents. Decay heat is produced in the blanket region and all throughout the primary circuit. In addition to blanket vessel temperature limits there are temperature limitations that come from the IHXs which, when properly designed, will be expensive pieces of equipment that need to be protected. One effective way to cope with overheating accidents is the drainage of the fuel - an option that does not exist for the MHTGR.

I don't see any technical reason, based on a comparison of ABC and MHTGR accidents, why the ABC power density was limited to 6 w/cc.

I believe one reason cited for the low power density for the ABC was the desire to achieve a 30-year graphite life. However, this is not a very prudent design objective when compared with the benefits from operating at a power density of, for example, 22 w/cc on which the MSBR design was based. I have not done a cost analysis but, in my view, the financial benefits coming from operating at a higher power density will totally overshadow the cost benefits from having a 30-year graphite life. Here, again the question is: what are the objectives? Power density is the most important cost driver and one must have a very good rationale for selecting a very low power density instead of a high power density.

What further complicates the accident situation in an ABC blanket is the presence of the target that, as far as I understand, is housed in a Hastelloy N structure. The melting point of Hastelloy N is 1370°C but it loses its strength long before that. For the MHTGR, even with the large heat sink in the core center, very high temperatures were predicted in a depressurization accident (much higher than the melting temperature for Hastelloy N) that did not allow any metallic structures in this region, including control rods.

Note: For these assessments it is very important to use a conservative (i.e., low end) thermal conductivity for graphite. It changes with fluence and temperature; it might also change with increased fission product depositions which might also affect the swelling/shrinkage of graphite.

Recommendation: Base the design on a higher power density, for example 20 w/cc. In a separate memo I will propose a series of calculations that will provide an envelope for setting an initial value for the power density.

Develop a rationale for the 500 MWt module power and develop a rationale for the need of six IHX to serve one 500 MWt blanket, i.e., using a 83 MWt IHX rather than larger units.

4.2 primary system layout

I realize that the viewgraphs show just a sketch of a primary system layout. I will, therefore, focus only on a few conceptual deficiencies.

location primary IHX vs blanket

Natural convection flow is driven by both the fluid density difference and the distance

between thermal centers. To establish effective natural convection flow, the IHXs for the ABC need to be raised. I realize that the MSBR design shows the core and IHX at about the same elevation. However, this is one of the deficiencies in the MSBR design.

The proximity IHX (and pump) to the blanket causes a large amount of fissions to occur in the IHX which has a series of implications. A large amount of fission gas is produced; I expect that a significant fast flux is produced in the IHX that is detrimental to the vessel (magnitude of fast fluence, azimuthal distribution) and affects also any flux monitoring outside the vessel.

Both pumps and IHX structures need to be shielded. This requires space and an effective shielding material. I think shielding will be one of the major challenges for the molten salt ABC design.

hot leg vs cold leg pumps

This issue has to be addressed for any reactor type, together with down-flow vs up-flow. There are pluses and minuses for either option. Whatever is selected has to be the result of trade-offs.

primary salt pumps

The schematic drawings show a compact vessel configuration in which pumps and IHX extend to the blanket midplane. As they are raised, the fuel salt inventory will increase and more graphite is needed to keep the fuel inventory low. The pumps are significantly larger than the MSRE pumps and might require a second (salt) bearing. They are part of the primary system boundary and are very complex in design. There are extensive shielding and insulation requirements (especially for the motors), as well as access requirements to maintain and repair the pumps (and IHX). The pumps will be much larger than shown for a more realistic design.

In summary, if one wants to show only that there are multiple HX-pump combinations that will serve one module, the viewgraph is fine. To go beyond this is risky.

IHX size

Recommendation: The scaling laws apply over a wide range of component sizes. Typically, a 160 MWt IHX costs much less than two 80 MWt IHX. When the system is laid out, these matters need to be considered. It is also important to consider plant availability and partial load operation with one or more IHX isolated. How an IHX can be isolated, both in terms of fuel salt and secondary salt supplies, needs to be evaluated.

need for a well-defined flow path

Based on the pictures describing the primary system layout, the flow path inside the blanket is well-defined (except for the bypass flow to be discussed later). Outside the blanket region the primary system flow path is defined by spaces that are not occupied by graphite. Any movement in those graphite structures will affect the primary system flow path. This is not an acceptable design approach. It can easily lead to flow and power imbalances among the IHXs with all kinds of repercussions. It can also lead to fuel salt flow bypassing the IHX and moving along gaps in the graphite.

shielding

Graphite-moderated systems have a much harder neutron spectrum than water-moderated systems, and shielding is very difficult. I suggest to read some of the shielding reports for the MHTGR that show how difficult it is to effectively reduce the neutron damage flux in a graphite system. Well-thermalized neutrons are easily taken out by boron. However, large distances of graphite structures are required to slow down neutrons so they can be caught in boron.

melt plug

To open and close valves, the MSRE used freeze valves and not mechanical valves. They performed well but have two essential shortcomings. For one, they are either open or closed and do not allow to throttle the flow. Secondly, they are effective only for small-diameter piping. It was recognized that for the MSBR the proper design of a freeze valve for 6-inch diameter (and greater) piping was a serious challenge requiring a fairly large refrigeration plant. Mechanical valves would have been considered for this plant, too.

The ABC design eliminates this "freezing" problem by using a melt plug that would be solid under normal operating conditions, but would melt if temperatures exceeded a certain limit. The flow throttling issue is the same for the two concepts.

The potential problem with the melt plug idea is the use of a different material inside the molten salt "containers" that would have to be replaced after use. In case of a freeze valve, such a replacement is not needed because the frozen salt is an effective "plug" to contain the salt. For a melt plug, a pipe section would have to have, for example, two valves with a replaceable pipe section in between that contains the melt plug. A closing of the valves would have to precede the replacement of this section. However, if mechanical valves are required for the replacement of the melt plug, the question arises: why not use mechanical valves in the first place, with assurance of redundancy.

5.0 GRAPHITE BLOCKS

No quantitative basis has been provided for the use of a graphite moderator as compared to an unmoderated configuration where the carrier salt provides for some level of moderation.

No basis has been provided for using graphite instead of a LiF-BeF_2 salt as moderator.

No basis has been provided for selecting MHTGR graphite blocks as moderator.

There is no basis for the 37 holes for fuel salt, each hole with a 1.2 cm diameter.

There is no basis for using only one type of graphite block. Varying the fuel volume fraction might be a more effective way of equalizing outlet temperatures than the proposed orificing that I will address later.

5.1 cold bypass flow

I don't understand what the "3 side slots to minimize stagnation of fuel outside moderator elements" are, as explained on one of the viewgraphs. There are gaps between the graphite blocks which are required to facilitate loading and unloading and for accommodating bowing and swelling. Having gaps between the blocks has an **undesirable but unavoidable side effect**, namely the creation of cold bypass flow channels. The ABC layout viewgraph shows sizeable gaps between the blocks. The hydraulic diameter of these gaps is much larger than that for the fuel/coolant hole. For the MHTGR, in excess of 20% cold bypass flow is possible. This means that a mixed mean coolant temperature rise of 200°C could be the result of an average 250°C temperature rise in the fuel channels and a near zero temperature rise in the gap channels. This has many other undesirable side effects. For example, the outside of the graphite block will be overcooled and a sizeable temperature gradient can be generated between the gap channel and the first row of holes in the blocks.

Block bowing must be expected not only because of the radial and axial power gradients (fluence effects) but also because of the unevenness in coolant flow through the block and the gaps.

5.2 flow orificing

Flow orificing is proposed to equalize outlet temperatures. The Ft. St. Vrain reactor (FSV) had even variable orifices that could be adjusted at each shutdown. However, the lesson from FSV is: never again. It is a very problematic undertaking that, to be effective, requires a very good knowledge of what the temperatures and power

distribution are.

Fixed orificing has as essential feature that it is fixed. In fuel elements that are replaced regularly and where a broad experience base exist, it is justified and beneficial. For a lifetime structure, however, this means it is fixed for the life of the reactor. If the power shape changes, this will throw off the orificing. If the analysis was off, this will also throw off the orificing. For FSV, even for later cores, power shape predictions were off as much as 15%, even when using properly adjusted cross sections and 3-d analysis methods (similar to DIF3D).

A major problem with flow adjustments in high-temperature reactors is the lack of in-core instrumentation that could measure coolant/fuel outlet temperatures on a coolant channel or even column basis. Typically, one measures only mixed mean outlet temperatures. However, those data are not good indicators for the distribution of outlet temperatures inside the blanket.

The bottom line is: orificing looks good on paper. In reality, the risks are too great and the best is not to orifice lifetime structures or, at the most, use only a very coarse orificing without attempting to optimize anything but just add to the safety margin.

Showing how the MHTGR coolant flow is collected at different elevations is a level of design detail that cannot be justified.

5.3 inlet and outlet plena

Reference is made to the MHTGR in that the ABC outlet plenum is similar to that for the MHTGR, but smaller in size. No information is provided on the ABC inlet plenum.

However, the ABC picture shown in the JASON handout is neither the MHTGR outlet nor the inlet plenum. The lower plenum in the MHTGR (outlet plenum because of the downflow) is a highly complex structure. It consists of support posts that fit into the bottom portion of graphite blocks. The graphite columns rest on those graphite support posts. The support posts in turn sit on a heavily insulated lower support plate. These structures allow for some movement to accommodate thermal expansion as well as design-basis earthquakes. The design issues for the plena are to ensure that all coolant channels receive the same amount of coolant flow (inlet plenum) and that there is sufficient mixing in the outlet plenum to keep hot streaking to a minimum.

It is not clear yet how the inlet flow in the ABC will be channelled to the respective graphite columns and how hot streaking will affect the upper outlet plenum. These effects can generally be minimized by using large plena. However, there is an incentive for the ABC to minimize fuel inventory. Trade studies need to be performed

to resolve these issues. How to support the blanket and provide sufficient mixing will be one of the serious design challenges for the ABC.

Note: The description of the bottom plenum of the ABC is a good example to illustrate the need for a structured approach in the design of the ABC. Firstly, what has been referred to as the ABC lower plenum shown in the viewgraph is actually the lower axial reflector region of the MHTGR and not the lower plenum. The lower plenum in the MHTGR is underneath this region.

Secondly, there are good reasons why the MHTGR has such a region. It is important to remember that **the MHTGR has coolant downflow**. These complex block structures are intended to provide some level of mixing to reduce the effects of hot streaking on the outlet plenum structures and the steam generator. The MHTGR inlet plenum has no such structure. The coolant inlet temperature is constant across the inlet plenum.

Thirdly, it is important to remember that **the ABC has coolant/fuel upflow** and, therefore, the inlet plenum is at the bottom and the outlet plenum is at the top. There is no need to introduce such complex flow patterns in the inlet plenum because there is no need for any mixing in the inlet plenum.

Recommendation: I don't know of any reactor that could not be built because of a lack of proper flow mixing. There are many design options available to mitigate the consequences of hot streaking and to provide for better mixing in the first place. To focus at this stage in the design evolution on flow mixing is premature considering the lack of a defensible basis for the overall design.

5.4 vessel wall cooling

Recommendation: It needs to be investigated whether the inlet flow should first be directed along the vessel wall in a downward flow toward an inlet plenum. This would ensure that the vessel, under normal operating conditions, is kept at a lower temperature. It would also keep down heat losses to the cavity.

5.5 vessel-inside-a-vessel approach

Recommendation: A better design approach might be a vessel-inside-a-vessel design that I had mentioned before where a blanket vessel is surrounded by another vessel/tank that contains the steam generators that are surrounded by coolant.

5.6 generic problems with graphite structures in molten salt

Graphite is lighter than molten salt. Graphite structures in a molten salt pool will tend to rise and have to be held down. During fuel drainage or filling of the primary system with molten salt, helium will either be added or removed from the primary system which leads either to a settling of the graphite structures or a lifting. Considering that these structures (graphite blocks) will experience differential shrinkage, accommodating both lifting and settling of structures that will change dimensions is a substantial design challenge. This situation is made worse by the bowing of the graphite columns as described in the following.

5.7 need for a core barrel

Maintaining core geometry is an important function that can be accomplished in many ways. For cores that contain fuel assemblies, thermal and irradiation induced bowing would tend to change the core geometry and introduce negative or positive reactivity changes. Upper and lower core support plates are used to control the geometry of the core, as well as lateral core restraints.

For fuel blocks, the assurance of core geometry is particularly important because graphite blocks are stacked on top of each other forming columns and, without a lateral restraint, those columns could break apart due to bowing or during earthquake transients.

As designed right now, the moderator blocks sit in a graphite reflector region that also surrounds the heat exchangers and pumps. In such a configuration, any loads coming from block bowing or block dislocations in general, would be transferred directly to the heat exchanger/pumps and the vessel wall. Furthermore, any block bowing would open up gaps that promote cross flow and flow between the side gaps. This would permit fuel to move from the blanket region to the heat exchanger/pump region and the vessel wall where the fuel would be trapped. It is conceivable that under certain conditions this fuel could even freeze. It is also conceivable that this fuel would not drain into drainage tanks.

Recommendation: To provide the required lateral blanket restraint and avoid the other consequences just listed it is advisable to introduce a blanket barrel or similar structure as a lateral blanket restraint. This would also mechanically decouple the blanket region from the rest inside the vessel.

6.0 CAVITY COOLING

No information is provided on the functional requirements for a cavity cooling system, nor is there a top level function stated that requires cavity cooling, i.e., the need for a cavity cooling system is not established.

There are some inconsistencies in the JASON viewgraphs (VG) describing the cavity cooling system. One VG states that "maximum temperatures in the blanket should not exceed 1200°C", and then continues that this temperature is "reached 75 hours into the transient". Another VG shows a figure where the maximum blanket temperature is <1100°C in an as of yet undefined heatup transient.

No information is provided on the basis for a limiting temperature of 1200°C.

No information is provided on the transient that is referred to as "heatup transient", how it was initiated, how it evolved, how it was terminated and how it was analyzed. There is no generic "heatup transient" because anything that leads to a temperature rise could be termed a "heatup transient".

The heatup transient curve is somewhat confusing. At time zero, the maximum temperature in the blanket is about 730°C. However, on the next page is stated that the inlet temperature for the IHX is 750°C. This is clearly a discrepancy. In addition, what is cited as "maximum temperature in the blanket" does not seem to be a "maximum" temperature but an "average" temperature. I would expect the maximum nominal blanket temperature to be at least 50°C higher.

In one of the VG, arrows point to what looks like pipes in concrete and refers to them as "reactor cavity cooling panels". At the DOE meeting on February 17 was stated that the cavity cooling system consists of pipes that are placed inside a concrete structure.

This reactor cavity cooling system is another example where functional requirements should be established before any claims can be made about a natural circulation loop that transfers heat to an ultimate heat sink. Here are some of the issues, assuming that a cavity cooling system is needed:

During the dryruns I pointed out that **thermal stresses in the vessel wall** have to be a major concern. I was then told that the vessel would be insulated. However, if the vessel is insulated one cannot use heat rejection through the vessel wall as a means of transporting decay heat to an ultimate heat sink, as has been proposed.

This is an example of what I call an **ad hoc design approach**. Cure thermal stresses by vessel insulation; cure decay heat removal by moving decay heat passively from blanket vessel into the cavity and from there passively to the outside - which would

require an uninsulated vessel. However, it is not possible to perform both functions.

What is the cavity temperature under normal operating conditions? For the MSBR, vessel temperatures are kept below 1300°F, and the cavity temperature is kept at 1000°F. If one places into such an environment a passive cavity cooling system (like the RCCS for the MHTGR), it would bring in air at ambient temperature and heat it up significantly by at least several hundred degrees. This means that a **large amount of heat would be rejected through the RCCS under normal operating conditions** (and the cavity possibly undercooled).

I think you have to keep the cavity temperature up to (1) **reduce thermal stresses** in the vessel wall and (2) to **avoid salt freezing**. As mentioned before, it might be necessary to use the inlet fuel/coolant flow to cool the vessel wall in a downflow path so that the vessel temperature can be lowered as well as the cavity temperature.

The cavity cooling system that has been described has similarities with the RCCS for the MHTGR and the containment wall cooling system of FSV without, however, appreciating the reasons behind those design selections for these reactors. **As described in the viewgraph, the passive cavity cooling system will not work.** The concrete in the vault has to be protected from high temperatures (typically, concrete temperatures have to stay below 150°F under normal operating conditions, and should stay below 350°F under off-normal conditions). This means there has to be thermal insulation on the concrete surface to protect it. However, **if there is thermal insulation on the concrete, the heat that has to be removed will never reach the cooling pipes located inside the concrete.**

The passive decay heat removal system for the ABC has to be addressed in a somewhat different fashion. Duplicating the RCCS from the MHTGR is not the answer.

Recommendation: Functional requirements for decay heat removal need to be developed first before any hardware design begins. The fuel drainage system should be part of any decay heat removal system. The use of DRACS and IRACS should be considered, especially when a vessel-inside-a-vessel concept is pursued that I proposed earlier. One of the major design challenges for any molten salt system is to keep the salt temperatures high enough to avoid salt freezing (note: the melting temperature for the MSBR fuel salt is about 500°C which is the same as the coolant outlet temperature for the ALMR), and the maximum allowable peak temperature is 1200°C (which is 400°C below that for the MHTGR).

7.0 ABC PROGRAM PLAN LAYOUT

I will only address the ABC Demo at LAMPF.

7.1 rationale for the LAMPF demonstration

As I pointed out during the January dryruns for the JASON meeting, no clear rationale has been developed for such a demonstration plant. The presentations involving a demonstration plant come always across as a justification for continued LAMPF operation ("demonstration-scale experiment is a program focus"). I agree with the general usefulness of a technology and design demonstration involving LAMPF but such an expensive experiment has to be part of an overall technology development program.

Furthermore, if the ATP is ever built it would already require the linkage of an accelerator with a core/blanket region, which further complicates the reasoning for a LAMPF demo.

Recommendation: Based on the premise that **DOE will not anymore support generic technology development work** the ABC Technology Development Program has to focus on support of the design and licensing of an ABC system as the sole justification.

Avoid statements such as "ABC Demonstration at LAMPF can [emphasis added] address important technical issues" because this implies that the experiment has a meaning in itself and what it can do for the project will be determined later.

Do not separate demonstration-scale experiment and techbase activities because the experiment has to be an integral part of the technology development program.

Develop a logic for a technology development program.

For the Technology Development Program Plan (TDPP) I suggest to begin with the identification of Design Data Needs (DDN) and use them as the basis for the TDPP. A similar format was very useful for the MHTGR and NPR programs.

To give at least some level of assurance that a TDPP is complete, one has to do some functional analysis (top-down design approach) and identify design requirements.

The DDN format could look like this:

1. Title
2. Requirement or Design Feature Requiring Experimental Data or Validation Testing
3. State of the Art
4. Design and Analysis Data Needs
5. Designer Alternatives
6. Required Experimental Work
7. Consequences of Non-Accomplishment and Fallback Position
8. Schedule Requirements
9. Priority

The format of the outline can be modified for as long as all the topics are addressed.

The development plan has to address issues related to design, licensing, fabrication, operation, recovery and possible even decommissioning of the plant.

I cannot overemphasize the importance of a clear and logical structure for design, analysis and technology development work. This might delay the beginning of the "real" work, but it pays off to plan in detail the work that needs to be done and to have good reasons why it has to be done.

Last but not least is the issue of going from a 1/25th scale experiment to a full-size module, i.e., going from a 20 MWt experiment to a 500 MWt module. I do not think that this extrapolation can be done without an intermediate step. Very often a factor 10 in size is the largest extrapolation considered acceptable. It should be noted that the MSBR development program relied on the ARE and MSRE experiments and had as an essential part the design and operation of a Molten Salt Test Reactor of sufficient size so that experience could be obtained for components that are full scale for a demonstration reactor.

7.2 design

Over 20 years have past since molten salt technology was practiced the last time in the U.S. How do handle molten salts, pump, transfer and store them, monitor, clean and analyze them, etc. are techniques that have not been practiced for a long time.

On the engineering side, no loop with forced circulation, pumping and valving off helium and molten salts has been built for a long time. I don't see in this plan any allowance for regaining this expertise.

It is projected that by the end of 1994, a design will have been developed. Today, there is not even a pre-conceptual design available. The design of a demonstration experiment has to encompass not only the blanket region and the accelerator, but also all support facilities for later experiments with fissile materials. There have to be a drainage system, off-gas systems, various cooling systems, heating systems, heat rejection systems, etc. etc. In my view, to complete the design work in three years is a rush job.

No attention has been given to the suppliers of materials, systems and components. As Dave Nulton explained at the February DOE-NE/NP meeting, there is no domestic supplier of graphite left. Cokes produced from new feedstocks need to be qualified for preproduction as well as production use. This problem was recognized by the NP-MHTGR Program and a time period of at least 12 month was estimated for the qualification of the coke source.

Recommendation: Before any time and cost estimates are made on designing and building a facility the availability of suppliers for the required materials, systems and components has to be investigated.

Before this design can be submitted for any ESH or regulatory-type review there has to be a technology development program that addresses the design data needs for the ABC systems that will be met by such a demonstration facility, but also the particular design data needs for such a facility. This Demo program has to be part of an overall technology development program for the ABC system.

It must be clearly stated what this Demo is expected to demonstrate. This Demo should be part of an overall technology development program and answer very specific questions that cannot be answered elsewhere.

The Development Plan Component Linkage is somewhat confusing. It shows only a conceptual design stage, an ABC demo at LAMPF, and then a 500 MWt prototype. Missing are the preliminary and final design stages that typically dominate the reactor design. Furthermore, technology experiments are listed only in support of the conceptual design and the ABC demo at LAMPF. No linkage is shown between the prototype and technology experiments. Only the ABC demo feeds into the prototype. This schematic can easily be misunderstood.

7.3 ESH approvals

You would set precedents if any review by ESH would begin a few month into the design. In my interactions with the NRC on the MHTGR it was made clear at the end that the NRC has no guidelines for a preapplication review. While this experiment would not really be a reactor, and its heat output is very low, it is laced with novel features.

It is worth remembering that the molten salt reactor has never been reviewed by the NRC. In today's climate the MSRE could never be built and operated without lengthy reviews along the line of a licensing review.

Note: The NRC had reviewed Ft. St. Vrain as well as the Fulton Plant, a commercial HTGR that was never built. HTGRs and coated particle fuel were very familiar to the NRC. However, when contacts were established between DOE and the NRC in 1985 on the licensing of the MHTGR, nearly two years were filled with NRC briefings on the MHTGR design before a Preliminary Safety Information Document was submitted to the NRC for review in October 1986. When the first meetings were held between the DOE and NRC in 1985, a licensing strategy had already been developed by DOE and a pre-conceptual design was available that had been developed in a top-down (integrated) fashion and that showed considerable design detail (that could also be defended).

Based on this experience it seems that the timelines shown on the Program Plan Layout are overly optimistic.

7.4 construction

To begin construction one year after design begins would set a new record, by many years. It is worth remembering that the CRBRP design began at Westinghouse ARD around 1969 and the NRC review for a construction permit was completed by 1983.

7.5 operation

To believe that demo operation can begin six months before construction is completed would set many records.

7.6 cost

I trust that the beam delivery costs of \$50M before operation starts and from then on \$30M per year, has a good basis. However, the other costs could easily be off by a factor 2-3. Firstly, there is nothing that is off-the-shelf technology anymore. Secondly, there are major development and testing programs required ranging from valves and pumps to IHX and steam generators. Thirdly, finding suppliers will be very difficult. As mentioned before, at this time, there is not even a domestic supplier for nuclear-grade graphite left in the U.S. Most likely, graphite structures would have to be imported from Japan (DOE will love this; this should provide an additional incentive to look very closely at an unmoderated system).

Note: **There are no molten salt component test facilities available anymore.**

To expect the completion of all the design work for an ABC system, including a demo at LAMPF, and nearly all the development work by the year 2000 is overly optimistic.

8.0 GENERAL OBSERVATIONS

8.1 Complexities in Design

Designing a reactor is a multi-disciplinary exercise into which feed considerations of physics, thermal-hydraulics, mechanical, safety and licensing, hardware design, reliability, fabricability, maintainability, inspectability, etc. My observations so far, supported by discussions I had with LANL design staff, seem to indicate that the complexities in the design process are not fully appreciated.

I often found in discussions that the issue was not to justify a particular choice of a design features but to ask the other side to explain why this particular design feature would not work or be acceptable. For example, DOE and the NRC require that all reactors have a prompt negative temperature coefficient. To offer the ABC with a strongly positive Doppler coefficient as a viable system requires a justification from the designer of this system, and not from the other side to explain why such a system is not viable.

8.2 Prioritization of Design and Analysis Work

The design and analysis work should be prioritized. Instead of drawing the design details of the manifolds and orifices in the lower part of the blanket, which is technically not very meaningful and most likely would not work anyway, issues of a more conceptual nature should be addressed. I still maintain that it would be very worthwhile had temperature coefficients been calculated for different fuel compositions and burnups as well as other analyses been performed as I suggested in another memorandum.

8.3 Reactivity Control

I have doubts that reactor reactivity control done solely through the accelerator will be acceptable. For reactor design it is not only important to show how well the reactor performs if everything works, but also to show how the reactor responds when components and systems fail. This is particularly important for components and systems that have never been used before, such as an accelerator in conjunction with a subcritical configuration. A probabilistic risk assessment for the accelerator would have to consider the possibility that you want it to be turned off but it continues to work. There was once an incident at a GE nuclear power plant where turning off the power was expected to automatically insert control rods. However, the circuit breakers did not break the circuits because they were stuck. Someone had painted them during maintenance work. The accelerator-reactor coupling is an untried system and accelerator control is a novel feature. To propose this untried system as the only

means by which to control and protect the reactor is insufficient.

Recommendation: I recommend to include conventional control rods in the reactivity control system. Even if the feasibility of accelerator control had already been demonstrated - and it has not been demonstrated so far - insisting on not using conventional control rods at least as a backup can be expected to be extremely expensive in the regulatory arena. LANL has been involved for many years not only in regulatory research but also in licensing review and knows that the time for licensing review can be increased by many years, with no assurance of success, whenever major novel features are proposed that have a profound impact on plant safety.

8.4 Radiological Source Term

Reference is made on a viewgraph to the 60% lower source term because of the removal of volatiles and noble metals. This figure has to be substantiated. To arrive at such a reduction, certain assumptions have to be made about the effectiveness of sparging and the removal of noble metals.

However, it should be remembered that those fission products have not been eliminated but only stored elsewhere on site. It still needs to be determined how vulnerable those storage facilities are. The reactor itself is a pretty good storage facility and designed for a very low probability for radionuclide releases. **We have not yet addressed the storage of fission products that have been removed from the reactor.**

Note: During the NRC licensing review of CANDU-3 one of the concerns expressed by NRC staff was the release of radioactivity from the spent fuel storage pool in case water would leave the storage pool (e.g., draining, vaporization) and the assemblies would overheat. For the MSR ABC, fission products outside the blanket exist in very high concentrations.

8.5 Graphite Moderator

Using graphite blocks as moderator is just one means to moderate the neutrons. It is not without problems. Here, again, one has to understand why this block fuel was selected for FSV and the MHTGR. HTGRs have a very low power density: about 6 w/cc compared to LWRs with 80 w/cc and LMRs with >300 w/cc. For the same power output, HTGRs are much larger structures than LWRs. If rodged fuel would be used, the fuel rods in a small 350 MWt HTGR would already be about 30 ft long. This is an impractical design. - It led to the idea of moving away from rodged fuel by

"chopping" the fuel assembly into small pieces (the blocks) and drilling holes through those blocks for the coolant and having blind holes for the fuel. These blocks are then stacked on top of each other and aligned through dowels. An additional major benefit in this design approach is the performance of those blocks under temperature and fluence gradients. They can more easily accommodate axial and radial temperature and fluence gradients without cracking than the very long fuel assemblies.

Note: The MSBR had 13 ft long graphite elements which is comparable to the length of fuel elements in LWRs.

Note: There is no more domestic supplier of nuclear-grade graphite.

Recommendation: Other moderators need to be evaluated, too. A system without moderator should also be investigated.

Note: While graphite is a good moderator, it also introduces significant problems (shrinkage, swelling, fission product adsorption, replacement issues).

8.6 Graphite Lifetime

I alluded earlier to the very low power density that has been selected for which graphite life extension was cited as the main reason. In my view this is not a good reason considering the price you have to pay. There is nothing generically beneficial in having a 30-year graphite life because there are always pluses and minuses that have to be weighed. Using a low power density will tend to extend blanket component and structure lifetimes. However, power density is a major cost driver. Any financial penalties coming from a shorter graphite lifetime are easily offset by an increased power density because more power can be extracted from the same vessel size. In general, two items of anything cost more than one item with twice the size.

A low power density could also improve the safety characteristics of the ABC. Some of the issues that need to be addressed in this regard will be addressed later.

8.7 Moderated vs Unmoderated Systems

It is quite possible that the current design efforts for the ABC with graphite block moderators go in the wrong direction. At this time, only vague plausibility arguments have been presented to show the need for an overmoderated ABC blanket. It is true that an unmoderated system has a higher fissile inventory than a well-moderated system. We also know that there are limits on the solubility of PuF_3 (and rare earth trifluorides) in the salt. However, it is insufficient to base the need for a moderator

on a comparison of the fission cross section for Pu-239 in a thermal and a fast spectrum. For one, in a fast spectrum, the parasitic absorption cross sections (carrier salt, fission products) are also much lower than in a thermal spectrum, and secondly, any system that contains large concentrations of beryllium, fluorine and lithium can hardly be called a fast system.

In the end, it might very well be that a thermal system has advantages over an unmoderated system. However, this case still needs to be made. An important topic in such an evaluation is the elimination of a graphite moderator that introduces many problems, not only in regard to lifetime, but also due to Xe adsorptions (and other fission product adsorptions) and the need for an effective sparging system to quickly remove gaseous fission products. Last but not least, we don't know yet what the deciding life-limiting constraint is for graphite. So far, it has been assumed that the life limit is reached when the graphite has passed through the shrinkage regime and has swelled to its original dimensions. However, we know from cladding swelling that impurities have an important impact on the swelling characteristics. We have only a very limited data base on graphite swelling/shrinkage in the presence of fission products obtained from the MSRE. We can safely assume that over a 30-year lifetime large amounts of fission products will have accumulated in the blanket graphite. It is conceivable that graphite lifetime limits might have to be correlated to the accumulation of fission products inside the graphite, some of which might have reacted chemically with the graphite.

8.8 Design Status

I was surprised to read on one of the viewgraphs: "optimization around this design not yet complete", because not even a basis has been stated for this design selection.

Recommendation: Do not select a design for further study that cannot be fully justified, qualitatively and quantitatively.

8.9 Accident Scenarios

It is worthwhile at the concept definition phase to spend some time identifying possible accident scenarios and to assess qualitatively their severity. For example, in case the heat sink is lost, advantage has to be taken of the possibility to drain the blanket and remove the decay heat passively from the drained fuel in the drainage system. This is a unique characteristic of liquid fuel, especially when the fuel can be drained passively from the blanket. It is not too meaningful to copy MHTGR design features to cope with LOHS accidents when fuel drainage provides such an easy way to terminate this transient. Furthermore, it is conceivable that the presence of graphite in the blanket region, the amount of which needs to be determined, provides

for sufficient heat storage in such an accident.

8.10 Structured Design Approach

In the interest of saving time and money, a structured design approach should be used that is supported by well-defined and targeted analysis the detail of which should be compatible with the level of detail in design available at this time. Any design has to meet specific objectives in regard to performance and safety.

8.11 Use of Fertile Material

It has been argued, and by and large with justification, that this system does not contain any fertile material and, therefore, does not breed fissile material. However, the initially very high positive Doppler coefficient will most likely require the addition of a fertile material. It needs to be determined how much fertile material (U-238, Th) needs to be added to have an assured negative Doppler coefficient, for how long this has to be added, and how this affects Pu disposition. Other options to produce a negative Doppler coefficient (e.g., addition of Li-6, Eu) need to be assessed.

8.12 Use of Solid Fuel Reactors to Burn Pu

On one of the viewgraphs is explained why LWRs and LMRs are not well suited for the disposition of Pu. I suggest to either strengthen the arguments considerably or to leave this line of argumentation out.

Omitted from consideration among the solid fuel systems is the CANDU reactor that has a well established continuous refueling system.

Not included in the evaluation is a pebble bed reactor that has continuous refueling and can achieve very high burnups. Its technology has successfully been demonstrated in the AVR. Development work would be required to produce Pu-bearing coated particle fuel. Such a reactor could be coupled with a gas turbine (like MARR One) and achieve 45% thermal efficiency. In a recent memorandum I proposed a pebble bed reactor with cross flow that might be attractive for Pu disposition.

I don't know to what an extent LWRs have been evaluated for Pu disposition. Using weapons-grade Pu together with Th would eliminate any Pu production. The net thermal-to-electric conversion efficiency in a LWR is in the 32-34% range. Considering the power requirements for the accelerator and the other power needs for the plant, I doubt that the ABC will have a higher net efficiency than LWRs.

"safety concerns with liquid metal coolant" are listed as a handicap for LMR technology. However, the LMR has a large negative Doppler reactivity and the likelihood for coolant boiling (that could introduce about +\$5 reactivity) is very low. Analyses conducted so far for severe accidents in an ALMR show that energy releases are low enough so that core integrity can be maintained under severe accident conditions.

8.13 Plant Availability

I discussed this subject during the JASON dryruns. Achieving a goal availability will affect the design of both the accelerator/target system and the blanket system. An availability assessment is typically conducted on a point design that is subsequently modified to achieve the goal availability. Both scheduled and unscheduled downtimes have to be accounted for. One of the complications for such an assessment comes from the fact that the reliability data bases for molten salt systems and accelerator/target systems are very limited. Operating a LAMPF demo would contribute to such a data base, even though it is (hopefully!) unlikely that any data on low probability-high consequence events could be obtained.

8.14 Overall Recommendation

I strongly recommend to make better use of the state of the art in our knowledge of reactor design, analysis, safety and licensing, technology and technology development. The design presented at the JASONS and DOE meetings has serious flaws that could easily have been avoided by a more appropriate understanding and use of the experience base that exists today.

The ad-hoc design approach currently in use should be replaced by a structured design approach that begins with the statement of objectives, translates them into goals, defines requirements and functions, leading to design selections. I have proposed to develop an outline for such an approach that would eliminate the current vulnerabilities in the design of the ABC.

Far greater emphasis has to be placed on tradeoff studies. During the evaluation of the aqueous homogeneous concept a large variety of design/concept options were investigated, some of them better than others. I think that this approach had great merits in that it led to a better understanding of the concept, its advantages and disadvantages. The molten salt concept had always been considered a backup system and did not benefit from the broad-based brainstorming activities for the aqueous homogeneous concept. After LANL decided to have the molten salt system as the only concept for Pu disposition, I have not seen any tradeoff studies that led to the current design. While the current graphite block design is far superior to earlier

designs, it only corrected serious deficiencies by the other design options without establishing itself as a feasible or even desirable design. This basis still has to be established.

Documentation requirements need to be established so that analysis results and design choices can be independently assessed.

A series of analyses need to be carried out at the stage where the program is being defined. I will submit such a list of analyses (neutronics, thermal-hydraulics, accident) that will provide a rational basis for the definition of the design.

In regard to the required technology development program I strongly suggest to look at the 1974 ORNL Program Plan for the Development of Molten Salt Breeder Reactors. Excluding the breeder-specific tasks from this Plan the following level of efforts were listed for base technology development:

Development of structural materials	11 years
Fuel processing	11 years
Chemical research and development	11 years
Analytical research and development	11 years
Reactor safety	11 years
Graphite development	11 years
Reactor technology	11 years
Maintenance	8 years(*)
Instrumentation and control	8 years(*)

(*) those activities were scheduled to begin in the fourth year of the program

To appreciate these estimates of the required level of effort one has to remember that:

1. when the Development Plan was written, ORNL had most of the required facilities still in operation
2. ORNL had several hundred scientists and engineers available who had recently worked on the molten salt reactor program
3. while some work on graphite and metals continued at a low level at ORNL after 1974, it is more difficult today than was perceived in 1974, to establish a sufficient technology basis for licensing

Comparing these efforts (plus the provisos) to the total projected base technology development effort at LANL of only about seven years clearly shows a level of optimism that is unjustified.

ABC RESPONSE TO VARIOUS SAFETY CHALLENGES

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1.0 APPLICABILITY OF STABILITY REQUIREMENTS FOR CIRCULATING FUEL REACTORS

1.1 DELAYED NEUTRON FRACTION FOR CIRCULATING FUEL REACTORS

As the molten salt fuel moves through the blanket region it carries with it the delayed neutron emitters. Some of them decay in the fuel loop outside the blanket region. This leads to a significant reduction in the effective delayed neutron fraction β for the blanket region. The following table shows delayed neutron fractions for stationary fuel reactors (reference 1) and those measured for the MSRE.

DELAYED NEUTRON FRACTION	SOLID REACTOR FUEL	MSRE
U-235	0.0065	0.0045
U-233	0.0026	0.0017
Pu-239	0.0021	0.0016*

* inferred

It should be noted that despite the lower delayed neutron fraction for U-233 fuel, the MSRE stability was actually greater for U-233 than for U-235 fuel, due primarily to the larger negative temperature coefficient of the fuel salt.

For a circulating fuel reactor, the reactivity ρ necessary to render the reactor super-prompt critical, namely $\rho = \beta$, is smaller than for a solid fuel reactor. Because reactivity insertions such as control rod withdrawals, poison removal, cold coolant additions and pressure increases are calculated as increases in eigenvalue, the lower effective delayed neutron fraction yields higher dollar values for those changes in a circulating fuel reactor.

The delayed neutron fraction for circulating fuel reactors depends on the time the delayed neutron emitters spend outside of the blanket region which in turn depends on the velocity of the fuel salt flow and the piping length and diameters outside the blanket region. The half-lives of the precursors range from 0.257 seconds to over 54 seconds. An analytical treatment of the effect of delayed neutron has to take into account both the leakage of precursors from the blanket and their re-entry into the

blanket region.

In case of a fuel flow stoppage, the delayed neutron fraction is increased and so is the reactivity in the blanket. For the MSBR, it has been estimated that a sudden flow stoppage in the primary system increases the reactivity by $+0.2\% \Delta k/k$. With a prompt temperature coefficient of $-87 \times 10^{-6}/^{\circ}\text{C}$, a 23°C fuel temperature rise would be sufficient to compensate for this reactivity increase.

In the following, two analytical approaches will be cited for the analysis of the stability of reactors with circulating fuel. Both of them go back to the mid-1950's when interest in this fuel type was considerable.

1.2 zero power kinetics for circulating fuel reactors

The conventional zero power kinetics equations are (using the common notations, with $R(t)$ being an external source):

$$\frac{du(t)}{dt} = \frac{K-1}{\Lambda} u(t) + \sum_i \lambda_i c_i(t) + \frac{R(t)}{\Lambda} \quad (1)$$

$$\frac{dc_i(t)}{dt} = -\lambda_i c_i(t) + \frac{\bar{\alpha}_i u(t)}{\Lambda} \quad (2)$$

For circulating fuel reactors, these equations have to be modified to account for the movement of the precursors:

$$\frac{du(t)}{dt} = \frac{K-1}{\Lambda} u(t) + \sum_i \lambda_i c_i(t) \quad (3)$$

$$\frac{dc_i(t)}{dt} = -(\lambda_i + \lambda_c) c_i(t) + \lambda_c e^{-\lambda_i/\lambda_c} c_i(t - 1/\lambda_c) \quad (4)$$

The left side of equation (3), $dc_i(t)/dt$, is the rate of change of the concentration of the i -th group of delayed neutron emitters in the blanket.

The term $-\lambda_c c_i$ is the flow rate of $c_i(t)$ out of the blanket region. λ_c is the specific flow rate of the molten salt out of the blanket in inverse seconds. If t_c is the residence time in the blanket of the circulating fuel, then $1/t_c = \lambda_c$.

The term $\lambda_c \exp(-\lambda_i/\lambda_c) c_i(t - 1/\lambda_c)$ is the immediate flow rate into the blanket of the i -th delayed neutron emitters that were $1/\lambda_c$ seconds away, in the external loop. However, these emitters have been reduced by a factor $\exp(-\lambda_i/\lambda_c)$ since they decay

exponentially with the decay constant λ_i , having spent $1/\lambda_i$ seconds in the loop exterior to the blanket.

To obtain the aggregate form of equations (3) and (4) the Laplace transform is used and the transfer function is calculated. References 2-5 discuss these analyses.

1.3 neutronics and temperature interaction for circulating reactors

Using the same formalism as before, references 4 and 5 investigate the stability of circulating fuel reactors by introducing power and reactivity feedbacks into the kinetics equations. This work is summarized in reference 7.

1.4 non-linear reactor kinetics

In reference 8, the stability of a reactor is discussed from which the heat Q is removed at a constant power P_0 and whose criticality factor C changes linearly with the temperature. With S being the thermal capacity of the system, the following equations apply:

$$\frac{dQ}{dt} = S \frac{dT}{dt} = P - P_0 \quad (5)$$

$$C = 1 + \delta C(t) + \alpha T \quad (6)$$

where α is the temperature coefficient of the reactivity and $\delta C(t)$ represents reactivity supplied by external means.

For the reactor power P which is proportional to the neutron population, we have:

$$\frac{dP}{dt} = \frac{\delta C + \alpha T}{l} P \quad (7)$$

with l being the generation time.

Equations (5) and (7) describe a kinetically non-linear reactor system whose stability will depend on the value of $\delta C(t)$ and α . To illustrate the stability of such a system, we assume that at $t=0$, $\delta C(t) = \delta C_0$, and divide (7) by (5) and obtain:

$$\frac{dP}{SdT} = \frac{(\delta C_0 + \alpha T)P}{l(P - P_0)} \quad (8)$$

Equ. (8) does not depend on time anymore, but is a correlation between power and temperature. Integrating (8) with the initial conditions $P = P_0$ and $T = 0$, we obtain

$$\frac{P - P_0}{P_0} - \ln \frac{P}{P_0} = \frac{S}{l P_0} \left(T \delta C_0 + \frac{\alpha}{2} T^2 \right) \quad (9)$$

A plot of this correlation (reference 8) is called a phase diagram. If we assume that

$$P = P_0 + \delta P \text{ with } \delta P \ll P_0, \delta C_0 > 0$$

the left side of (9) will be zero, which requires that $\alpha < 0$, i.e., the temperature coefficient must be negative for this system to be stable. Any perturbation that raises the reactor temperature leads to an increase in reactivity, which raises also the power, which in turn raises the temperature, etc.

The above treatment of the stability is simplified but transparent. Adding the effects of delayed neutrons complicates the equations significantly. For a constant $\delta C = \delta C_0$, the corresponding equations are

$$l \frac{dP}{dt} = (\delta C_0 - \beta + \alpha T)P + \beta P_0 \quad (10)$$

$$S \frac{dT}{dt} = P - P_0 \quad (11)$$

Only the stability against small oscillations can be handled easily; however, even then it is not possible to directly correlate temperature coefficient to stability.

$$\begin{aligned} P - P_0 &= -\frac{S \delta C_0 \omega}{\alpha} \left(1 + \frac{\beta^2}{4 l^2 \omega^2} \right) \sin \omega t \exp -\frac{\beta t}{2 l} \\ T &= -\frac{\delta C_0}{\alpha} \left[\cos \omega t + \frac{\beta}{2 l \omega} \sin \omega t \right] \exp -\frac{\beta t}{2 l} \\ \tau &= \frac{2\pi}{\omega} = \frac{2\pi (l S)^{1/2}}{(k l P_0 - \beta^2 S / 4 l)^{1/2}} \end{aligned}$$

2.0 INITIATION OF SAFETY CHALLENGES, ABC RESPONSE, AND ACCIDENT TERMINATION

2.1 SAFETY ISSUES FOR THE MOLTEN SALT ABC SYSTEM

The following is a list of safety issues for the ABC system:

- number and effectiveness of barriers to radionuclide releases
- assurance of negative temperature and void coefficients at all times
- assurance of reactivity control
- assurance of the liquid state of fuel and coolant salts at all times
- molten fuel salt - coolant salt interactions
- tritium production, transport and retention in coolant
- primary system activation and personnel doses
- control of fuel properties
- continuous salt surveillance
- performance of fuel salt containing high concentration of fission products is unknown
- rare earth, actinide and plutonium solubility
- noble metal transport and plateout mechanism
- blanket - beam tube interface
- graphite life limit/degradation under irradiation
- structural material performance under irradiation
- shielding to ensure low personnel doses and fluences on structures
- component replacements and remote handling
- offgas system performance
- drainage system reliability

All these issues come into play when certain safety challenges are imposed on the ABC.

2.2 EVENT DEFINITIONS

Following is the nomenclature used by the NRC in the recent CANDU-3 system assessment (reference 9), whose safety challenges are the AOOs, DBAs, SAs, and RRs defined below.

AOO: anticipated operational occurrence
DBA: design basis accident
SA: severe accident
RR: residual risk event

I recommend to use these classifications for the ABC safety assessments.

2.3 SEQUENCE CLASSIFICATIONS

Following are the probability ranges for the four classes of accidents.

P...probability for occurrence per year

Anticipated Operational Occurrence (AOO):	$P \geq 10^{-2}$
Design Basis Accident (DBA):	$10^{-2} > P \geq 10^{-4}$
Severe Accident (SA):	$10^{-4} > P \geq 10^{-6}$
Residual Risk (RR):	$P < 10^{-6}$

AOOs are expected once or more in a plant lifetime.

DBAs are expected to occur once or more in the lifetime of a population of plants

FAULT TREE ASSUMPTIONS	
Human error (non-routine or low training factor)	10^{-1}
Human error (routine or high training factor)	10^{-2}
Mechanical active failure	10^{-2}
Mechanical passive failure	10^{-3}
System failure (safety-related)	10^{-3}
System failure (non-safety related)	10^{-2}

CRITERIA FOR CATEGORIZATION OF EVENT SEQUENCES

EVENT CATEGORY	CRITERIA
AOO	Initiating Event Frequency $P \geq 10^{-2}$ PLUS No Failures (Bin A) or 1 single active component failure (Bin B) or 1 operator error (Bin B)
DBA	Initiating Event Frequency $10^{-2} > P \geq 10^{-4}$ Plus No Failures (Bin E) or 1 single active component failure (Bin F) or 1 operator error (Bin F)
SA	Initiating Event Frequency $P \geq 10^{-4}$ Plus 1 system failure (Bin C) or 2 single active component failures (Bin G) or 2 operator errors (Bin G) or 2 combination failures (Bin G) or 1 common mode failure (Bin G)
	OR
	Initiating Event Frequency $10^{-4} > P \geq 10^{-7}$ Plus no failures (Bin I) or 1 single active component failure (Bin J) or 1 operator error (Bin J)
RR	All other sequences (Bin D) (Bin H) (Bin K)

It should be noted that these classifications relate only to probabilities for occurrence and not to consequences. They are to be used for surveys only and not for the identification of safety-grade vs non-safety grade systems, structures and components. Furthermore, the term "safety-related" is used here only to highlight that there is a relation to safety and no relationship to radiological consequences is implied. NRC-NRR uses different definitions in the regulatory review process.

The reactor must accommodate the AOOs and DBEs without permanent damage to the reactor, without excessive radionuclide releases, and with a restart of the reactor typically within less than one year of outage. User requirements will identify the maximum permissible forced outage times. The designer must show that the reactor can ride through a severe accident without excessive radionuclide releases; however, a restart of the reactor is typically not expected (e.g., a large break LOCA in CANDU-3 will most likely lead to a permanently shutdown reactor).

Because the residual risk is only defined as an accident with a probability of less than 10^{-6} , an assessment of those accidents is focused mainly on identifying "cliffs", i.e., events that are just below the SA threshold. However, during the recent licensability review of the MHTGR, the NRC requested analyses of events with probabilities in the 10^{-12} range and below.

2.4 Initiating Event List

At the beginning of all safety assessments are the initiating events. Whenever a list of initiating events is presented the standard question is: how do you know that this is a complete list? For any new system one can only present plausibility arguments but no 100% assurance of completeness (the corollary is: write down everything you don't know and make the list complete).

I suggest to approach this issue by first looking at initiators for other reactors, listing initiating events in the basic categories of overpower, undercooling, overcooling and loss-of-heat-sink events, and supplement this list by ABC design-specific events.

The next step would be the selection of dominant initiators and to derive scenarios for AOOs, DBAs, SAs, and RRs.

A significant problem for molten salt reactors is the absence of applicable reliability data bases. This affects the ability to perform credible probabilistic risk assessments and to properly classify accidents. In several instances, for example, for the steam generator and mechanical valves, no performance data are available because these components have never been tested. This problem is further aggravated by the linkage of a molten salt blanket to an accelerator whose reliability is unknown, just as the interaction of the two systems is unknown.

For screening purposes, and in the absence of applicable reliability data, I recommend the use of conservative failure probabilities that were derived and used for the CANDU-3 systems assessment (reference 10). The following values were used:

•	mechanical active failure	10^{-2}
•	mechanical passive failure	10^{-3}
•	system failure (safety-related)	10^{-3}
•	system failure (non-safety related)	10^{-2}
•	human error (nonroutine)	10^{-1}
•	human error (routine)	10^{-2}

These frequencies are order-of-magnitude values. Surprisingly, they gave good agreement for accident sequence predictions when compared with AECL data that were the results of elaborate reliability assessments.

2.5 NORMAL OPERATING CONDITIONS

Single active component failures or operator errors under normal operating conditions can lead to safety challenges. To identify those challenges requires typically a fairly detailed design. Because the response to those challenges is benign by design, they will not be addressed at this time.

2.6 OFF-NORMAL CONDITIONS

For off-normal conditions, the following accident groups will be described:

- transient overpower events
- transient undercooling events
- loss of heat sink events
- transient overcooling events

For each accident group, accident initiators, accident evolution and termination will be described.

Regarding the ABC layout, I refer to a vessel-in-vessel system that would place the steam generator also into a vessel. This is different from the current ABC design where the steam generators are located outside the vessel in the containment building. However, I think the current design is most likely not licensable for the following reasons:

1. The liquid fuel ABC system lacks two of the barriers to radionuclide releases that LWRs have, namely the fuel pellet and the fuel cladding. In LWRs, the

probability for cladding failure even at end-of-life conditions is in the range of $< 10E-4$ (a few gas leakers at EOL). The fission product concentration in the primary system is very low and limited to gaseous fission products. When solid fission products are detected indicating a fuel washout from the fuel pellet, the reactor will be shut down. Massive releases of fission products into the primary system are very low-probability events.

For the ABC system it can be assumed that, as far as radionuclide releases are concerned, all fission products are distributed throughout the primary system, i.e., they have already been released to the (coolant) fuel salt.

2. The first barrier to releases in the ABC system is the IHX and tube failure probabilities are typically in the 0.1 to 0.01 range. If an IHX tube fails, fission products can enter the primary side of the steam generator.
3. Steam generator tube failures are also in the 0.1 to 0.01 range. This means that the probability that fission products can enter the steam loop are in the 0.01 to 0.0001 range. Any failure or leak in the steam isolation valves will result in releases to the containment. Redundancy in the design can lower this failure probability significantly especially when the steam generators are located in a vessel that surrounds the blanket vessel.

However, in the current design the secondary coolant salt from the IHX is piped to the steam generator that is located outside the blanket vessel (in the containment building). If this pipe breaks, following an IHX tube failure, fission products will be spilled into the containment building. This scenario is different from the steam generator tube rupture where steam isolation valves can bottle up the steam loop and greatly mitigate the chances for releases to the containment building.

4. The failure probability for the containment is in the range of 0.1. This means that at best, a probability of $10E-5$ can be achieved for massive releases of fission products to the environment. This is at least a factor 10 too high.

If the steam generator and possibly one set of isolation valves are contained in a separate vessel, the probability for releases can be greatly reduced. Details of such a vessel system need to be worked out. However, the current ABC system that places the steam generators inside the containment building is, in my view, not licensable.

The following tables discuss transient overpower events, transient undercooling events, loss of heat sink events, and transient overcooling events, their evolution and termination.

TRANSIENT OVERPOWER EVENTS	
INITIATOR	TERMINATION
loss of reactivity control	
uncontrolled control rod withdrawal	only a very small reactivity addition is possible because of continuous refueling and small excess reactivity requirement to compensate for burnup; slight rise in fuel temperature; negative temperature coefficient terminates transient
injection of excess fuel salt	only a small reactivity addition is possible because of continuous refueling; slight rise in fuel temperature; negative temperature coefficient terminates transient
return of overcooled fuel salt	slow transient with only small reactivity addition that will be terminated by negative temperature coefficient
loss of pressure control	<u>lowering the pressure</u> will not lead to fuel salt boiling because of the wide temperature margin to boiling; can lead to expansion of dissolved gas bubbles which would increase neutron leakage and tend to shut down the reactor <u>increasing the pressure</u> can lead to bubble collapse adding a small amount of reactivity (e.g., 0.2%) that can be controlled by negative temperature coefficient and lead only to a small fuel temperature increase
loss of flow	the resulting increase in delayed neutron fraction will lead to a slight fuel salt temperature increase; negative temperature coefficient will terminate transient
loss of inventory control	because of the continued salt surveillance a loss of inventory control is most unlikely; any change in fuel salt composition will only very slowly change the neutron spectrum and add or remove only very small amounts of reactivity; operational control rods and negative temperature coefficient will terminate the transient
loss of fission product inventory control	because of continued salt surveillance, a loss of fission product inventory control is most unlikely; because of the continuous removal of gaseous fission products and noble metals in form of "smoke" or aerosol no sudden loss can occur that cannot be controlled by control rods and/or temperature coefficient

GAS ENTRAINMENT

The ABC layout shows a helium plenum on top of the blanket region that is not explained any further. It is not clear whether this is part of a pressure control system or any other system. Such a system would bring helium in contact with the molten salt. Even though helium is not soluble in molten salts, there can be an entrainment of gas in the molten salt. Furthermore, the primary cooling system will most likely have bubble generators and separators to remove gaseous fission products.

Through either means, or both, the possibility exists for gas entrainment, gas bubble growth and collapse. It is also conceivable that somewhere in the primary system helium gas had been trapped when the blanket vessel was filled with molten salt. How large those bubbles can grow and how their behavior will affect criticality is unknown. However, this phenomenon was one of the concerns for the MSBR. Even if the bubble formation and behavior does not pose a severe accident risk, reactivity fluctuations could potentially be a problem. Those fluctuations had been observed in the MSRE when U-233 fuel was introduced. Over a period of one month, power fluctuations were observed leading to sporadic power increases (about 5-10%), lasting for a few seconds and occurring with varying frequency around 10 per hour.

Fission gases such as Xe and Kr are expected to be adsorbed in pores in the graphite structure inside the blanket vessel. The helium bubbles that are introduced into the fuel salt to offer free surfaces and compete with the graphite porosity for the fission gases (especially Xe-135 with its high thermal neutron absorption cross section of > 1 million barns), are expected to circulate 20 or more times around the primary system before they are saturated with Xe and removed from the salt. It is not known how reliable a bubble generator will operate and how bubble size and adsorption characteristics can vary. It still needs to be investigated whether these variations could give rise to power fluctuations.

It is also unknown whether power density/neutron flux (i.e., Xe-135 generation rate) and extraction efficiency through helium bubbles are correlated and whether there are power density/neutron flux limits for the efficient removal of gaseous fission products and how the potential for power fluctuations relates to power density/neutron flux.

The easiest solution to this potential problem would be an unmoderated system where a bubble generator would not be needed to strip the noble gases from the salt to avoid their adsorption in graphite. The second best solution would be an effective sealing of the graphite and/or the use of a low-porosity graphite so that only a small amount of the noble gases would be adsorbed.

Note: The MSRE did not have a bubbler as proposed for the MSBR. Contact with the helium cover gas was provided by a 50-gpm spray in the fuel pump bowl, which not only produced salt spray in the gas phase but also carried bubbles of helium into the salt, some of which entered the circulating loop.

TRANSIENT UNDERCOOLING EVENTS	
INITIATOR	TERMINATION
Loss of Heat Transport System Inventory (LOCA) depressurization without break in-blanket breaks out-of-blanket breaks	In the primary system, the liquid fuel is also the heat transport medium. The vessel-in-vessel concept combined with a pool configuration inside the blanket vessel eliminates the possibility of a loss of liquid fuel Because the primary system pressure is only ~ 200 psi, a depressurization of the primary system does not lead to fuel boiling and/or loss of fuel Because the fuel is also the primary coolant, and the graphite moderator has no structural function, there are no in-blanket breaks that could lead to a loss of fuel Because the blanket and primary heat exchanger are located in the blanket vessel, fuel is not lost in a pipe or hot-cold interface break from the system; small- break or large-break LOCA are not possible in the MS ABC
Loss of Heat Transport System Coolant Flow (LOF) loss of off-site power pump seizure	Any loss of off-site and/or on-site backup power creates a possibility for fuel and/or coolant freezing. While natural convection flow will commence in all loops, the different heat loads in the various loops and the different salt melting points, molten salt flow rates, and steam generator feedwater flows and temperatures render it highly doubtful that each system can be fine-tuned such that all the decay heat is removed and the molten salts remain liquid. To avoid freezing, highly reliable backup power is required to ensure that the appropriate flow rates can be maintained. multiple loop operation together with loop isolation followed by the required repair or replacement deal with this accident

LOSS OF HEATER CAPABILITY

A unique feature of molten salt systems is the need for maintaining high salt temperatures to avoid salt freezing. For the MSBR this goal was achieved through heated cavities into which salt-carrying components and systems were located. If a vessel-in-vessel concept is employed, with both vessels containing molten salt, the heat losses during normal operation and under accident conditions from the blanket vessel into the surrounding vessel could be sufficient to avoid salt freezing. If this vessel system is contained in a heated cavity, a high level of assurance would be provided for maintaining the salts in their liquid state. If it is not contained in a heated cavity, a double-walled outer vessel could have to be used where the space between the two walls is used for heating purposes. A simpler design would use a single wall vessel that is heavily insulated. In all cases, an MHTGR/ALMR-like passive cavity cooling system would not work. Instead, DRACS and IRACS, similar to those developed for the ALMR, could be used.

The heater requirements would be lessened for a vessel-in-vessel configuration compared to the proposed ABC layout. As the decay heat production decreases the time will come when an external heat source is required to avoid salt freezing. If this heater capability is lost, freezing will occur.

How the systems respond to a freeze-up still needs to be investigated. There is only a minor change in volume when a molten salt freezes. This means that the freeze-up in itself might not pose a safety problem but an investment protection problem due to the difficulties to melt the frozen salt. During the MSRE operation, off-gas lines were plugged several times because the salt came into contact with air. Heating up the plugged lines easily corrected this malfunction.

LOSS OF HEAT SINK EVENTS	
INITIATOR	TERMINATION
Steam line break	The vessel systems are designed such that the steam generator is located in the outer vessel. The only (major) penetrations in the outer vessel are for feedwater lines going in, and steam lines going out. This accident needs to be assessed for a steam line break either inside the vessel or outside the vessel, in the latter case (possibly) defining the containment load design requirements.
loss of feedwater/secondary heat removal system	After accident initiation, the blanket is brought to decay heat level; circulation in the various loops is maintained to ensure that the salt remains liquid or the loop without circulation is isolated. Decay heat is rejected through the blanket vessel wall into the outer vessel from where it is removed via DRACS and IRACS.
loss of service water	The fission product decay heat is passively removed through the vessel walls.
loss of dump tank cooling	The dump tank cooling system is a passive heat removal system with a high level of redundancy to make a complete loss of dump tank cooling an extremely low probability event.
loss of shutdown cooling system	If the active systems fail and/or have not been initiated manually, the passive heat removal systems will remove the decay heat
loss of off-gas system cooling	Because of the very low power density in this system, it can be designed for passive heat removal.

TRANSIENT OVERCOOLING EVENTS	
INITIATOR	TERMINATION
Increase in Heat Removal by Secondary System	The consequences of the following malfunctions, for example, need to be assessed: ● feedwater system malfunctions that result in an increase in feedwater flow or a decrease in feedwater temperature; ● steam generator regulator malfunctions or fails that results in increasing steam flow (turbine overspeed); ● inadvertent opening of a steam generator relief valve or safety valve; ● increased heat transfer flow or inventory; ● startup of an inactive heat transfer loop or recirculating loop at an incorrect temperature; ● the air in the condenser is too cold and cools down the secondary salt below the allowable inlet temperature. If no corrective actions are undertaken, salt overcooling could result and possibly salt freezing.

3.0 POTENTIAL RADIONUCLIDE RELEASES

3.1 FISSION PRODUCT RELEASE PATHS¹

Fission product release paths are design-dependent. In the absence of a conceptual molten salt ABC system design, only generic release paths can be addressed at this time.

Unlike for solid fuel reactors, the fuel pellet and cladding barriers to radionuclide releases do not exist for liquid fuel reactors. Loop-type arrangements have been proposed in the past. They were always predicated on the assumption that the fuel would circulate in at least double-walled piping to reduce the possibilities for leakage of the fuel and fission products.

The vessel-in-vessel concept I had proposed uses a blanket vessel that houses the blanket and intermediate heat exchangers (e.g., once-through IHXs) in a pool configuration. The only vessel penetrations are for the secondary loop coolant salt inlet and outlet, and possibly for pumps housed in the vessel. The secondary vessel contains the steam generators, with penetrations only for the feedwater inlet and steam outlet.

The details of such a configuration have not been developed yet.

As fission product release paths are identified those paths that would permit fission products to escape from the primary system. Examples are the following:

- pumps (primary system, jet pumps in drainage tank) through seals and overflow lines
- pipe breaks (minimal effect in vessel-in-vessel concept)
- IHX tube failures
- steam generator tube failures
- instrument and monitoring lines
- refueling lines
- bypass flow lines including bubblers
- off-gas system lines
- drainage lines

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¹not considered here is tritium that diffuses rapidly through intact metal walls and will reach the steam system very rapidly.

3.2 RADIONUCLIDE RELEASE PATHS FOR DIFFERENT SPECIES

A summary of different potential radionuclide release paths is given below.

NOBLE GASES

- continuously removed through off-gas system
  - bypass flow lines including bubblers
  - off-gas system lines
  - cover gas system (if used)

NOBLE AND SEMI-NOBLE METALS

- deposited on metallic surfaces and to a lesser extent on graphite
- found in off-gas system
  - bypass flow lines including bubblers
  - off-gas system lines

ALL REMAINING RADIONUCLIDES

- releases through breaches in the primary system boundary such as:
  - pumps (primary system, jet pumps in drainage tank) through seals and overflow lines
  - pipe breaks
  - IHX tube failures
  - steam generator tube failure
  - instrument and monitoring lines
  - refueling lines
  - bypass flow lines including bubblers
  - off-gas system lines
  - drainage lines

3.3 RADIONUCLIDE INTERACTIONS WITH GRAPHITE, METALS, AND SALT

Radionuclides will interact with graphite, metals and salt. As the table on the next page shows, the fission product distribution is not always well understood.

GRAPHITE

- retention of noble gases
- adsorption of Cs
- retention of noble and seminoble fission products
- carbide formation

METALS

- noble metals exist in elemental form
- noble metals present in off-gas
- noble metals behavior not fully understood

SALT

- only a very small fraction of the noble gases is dissolved in the fuel salt
- fluorine balance in MSBR is controlled by ratio of UF_3 to UF_4 ratio to maintain a non-corrosive environment; fluorine balance and control needs to be determined for PuF_3 system
- lanthanide and actinide trifluorides are only moderately soluble in molten fuel salt (limit is expected to be near 1.3 mole %, but no good data base available)
- a large fraction of iodine remains in the fuel
- tritium leaves the fuel salt and is expected to be trapped in coolant salt $NaF-NaBF_4$; the effectiveness of trapping and trapping mechanisms need to be determined
- nearly all of the Zr, Sr, and Ba remain in the fuel salt.

The analysis of MSRE gas samples offered several surprises, among them the very high concentrations of Mo, Nb, Te, and Ru in the gas phase. In addition, appreciable fractions of Ag-110 and of palladium isotopes have also been observed in the purge gas samples.

While the MSRE experience is extremely valuable it should be noted that the MSRE operated only for about 1 ½ full-power years. How the radionuclides interact with graphite and metals over a 20-30 full-power year period is difficult to extrapolate from the MSRE experience. An extensive surveillance system might be necessary to monitor any long-term effects that might not even have been identified yet.

FISSION PRODUCT DISTRIBUTION FOR MSRE AT END OF U-235 OPERATION

QUANTITY FOUND (% OF CALCULATED INVENTORY)					
Nuclide	In Fuel	On Graph-ite	On Hastelloy N	In Purge Gas	Total
Mo-99	0.17	9.0	19	50	78
Te-132	0.47	5.1	9.5	74	89
Te-129	0.40	5.6	11.5	31	48
Ru-103	0.033	3.5	2.5	49	55
Ru-106	0.10	4.3	3.2	(130)	138
Nb-95	0.001-2.2	41	12	11	64
Zr-95	94	0.14	0.06	0.43	95
Sr-89	83	8.5	0.08	17	109
Ba-140	96	1.9	0.05	0.48	98
Ce-141		0.33	0.03	0.88	
Ce-144		0.92	0.03	2.7	
I-131	60	0.11	0.3	19	80

References:

1. J. R. Lamarsh, "Nuclear Reactor Theory" (1972)
2. M. A. Schultz, "Control of Nuclear Reactors and Power Plants", 1961
3. J. MacPhee, "The Kinetics of Circulating Fuel Reactors", Nucl. Sci. Eng., 4, 588 (1958)
4. W. K. Ergen, "Kinetics of the Circulating Fuel Nuclear Reactors", J. Appl. Phys., 25 (6), 703-711 (1954)
5. T. A. Welton, "Kinetics of Stationary Reactor Systems", Proc. U.N. Intern. Conf. Peaceful Uses At. Energy, Geneva, 1955, 5, 377-388 (1955)
6. J. A. Fleck, "Theory of Low Power Kinetics of Circulating Fuel Reactors", Nucleonics 12 (11), (1954)
7. Milton Ask, "Nuclear Reactor Kinetics", (1965)
8. A. M. Weinberg, E. P. Wigner, "The Physical Theory of Neutron Chain Reactions", 1958
9. Letter, Zoltan Rosztoczy (NRC-RES) to Jim Dyer, dated August 17, 1992.
10. "System Analysis of the CANDU 3 Reactor", NUREG/CR-6065, (1993)

RESEARCH AND DEVELOPMENT TASKS FOR THE ABC

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The purpose of this report is to identify the design support and safety data needs for the Molten Salt ABC to meet its objective of safely, efficiently, and economically disposing of weapon's grade plutonium.

The report in its current form is an attempt to develop an outline for such a research and development program that is narrowly focused and design data needs driven. It is not intended to represent a broad technology development program for the generic molten salt reactor. The elements of such a research and development program are:

1. design confirmation
2. basic physical property data
3. development and testing
4. validation

Because the design is very sketchy at this time, and in many areas not defined yet, the list of research and development tasks is necessarily incomplete. The tentative nature of this list of tasks is further enhanced by the fact that in some instances the general design approach for the ABC has not even been defined. For example, the containment concept, at this time, is tentative at best.

The process by which the R&D plans should be developed consists of three major tasks:

1. establish design and safety requirements for the ABC

These documents have to be prepared, stating design and safety requirements which, when met, will accomplish the following mission¹:

Provide a weapon's grade plutonium disposition capability in a safe, economical, and environmentally sound manner on a prudent schedule for x tons of weapon's grade plutonium to be disposed of in y years.

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¹the following is used as an example and should not be interpreted as a proposed mission statement and objectives.

This mission is supported by the following four major objectives:

- provide a reliable plutonium disposition capability within the next 2 years
- provide a level of safety and of safety assurance that meets or exceeds that afforded to the public by modern commercial nuclear power plants
- meet or exceed all applicable federal, state, and local regulations or standards for environmental compliance
- manage the program in a cost effective manner

The design and safety requirements documents will typically address:

- design features and processes
- safety and regulations
- safety, security, and emergency planning
- operation
- environmental issues
- quality assurance and control
- economics
- construction
- decommissioning
- management

2. develop an ABC design using a top-down, integrated design approach that is consistent with the mission objectives, based on the translation of objectives and goals into specific requirements and functions, leading to design selections based on tradeoff studies

As the design is developed in this manner, use will be made of existing data/information bases as well information on existing hardware (components, systems, and structures). This design process will lead to the **identification of design data needs (DDNs)**(see table 1) whenever existing databases are incomplete or non-existent (but required) and required components, systems, and structures have either not been developed at all, or are not directly applicable to the ABC concept. In regard to the latter, attention has to be given to existing industrial capabilities in the U.S. and overseas.

In other words, whenever the designer has to make an assumption because of deficiencies in the state of the art, a design data need has to be addressed.

An outline for a DDN could be as follows:

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING
2. STATE OF THE ART

3. DESIGN AND ANALYSIS DATA NEEDS
4. DESIGNER ALTERNATIVES
5. REQUIRED EXPERIMENTAL WORK
6. RECOMMENDED TEST FACILITY
7. CONSEQUENCES OF NONACCOMPLISHMENT AND FALLBACK POSITION
8. SCHEDULE REQUIREMENTS
9. PRIORITY

These DDNs will be prioritized and evaluated. It is very beneficial to involve already at the DDN stage those who will perform the R&D work to identify, for example, realistic uncertainty goals. For example, during the development of DDNs on coated particle failure model validation it was found that the desired uncertainty and level of confidence for the model validation data could not be achieved with current experimental uncertainties and biases.

Depending on the number of DDNs, the description of all DDNs is either a separate report or an addendum to the Technology Research and Development Plan.

It should be expected that new DDNs will be developed as the design progresses and some already existing DDNs might be dropped or modified as more is learned about the ABC.

3. develop a Technology Research and Development Plan that responds to the DDNs

The DDNs serve as the basis for the **development of technology development tasks (TDTs)**(see table 2) which describe experiments and analyses that have to be performed to respond to the DDNs.

Note: the number of TDTs can differ from the number of DDNs because several DDNs could be addressed in one experiment.

The Technology Development Tasks that represent the content for the Technology Research and Development Plan should have a structure similar to the following:

1. REFERENCE
2. PRIORITY
3. DATA NEEDED
4. REQUESTED CONDITIONS
5. APPROACH

6. WORK BREAKDOWN
7. APPARATUS/FACILITY
8. SCHEDULE AND COST

What follows are samples of the ingredients of an ABC Research and Development Plan.

Note: In 1974, ORNL prepared a comprehensive Program Plan for Development of Molten Salt Breeder Reactors¹ that is very useful as a summary of (1) the state of technology and (2) perceived gaps in the technology. There are two reports published later that also contain information on what the development needs are^{2,3}.

While these are very valuable documents I suggest to evaluate the content carefully. To illustrate this point: I have some doubts about the level of certainty expressed in regard to the state of graphite technology. I have no reason to doubt the correctness of the factual information but question some of the conclusions. For example, the irradiation performance of different graphites has been examined in irradiation experiments in HFIR. Swelling and shrinkage rates were determined and material properties measured. Upon completion of this work, however, one can only conclude that these are the properties of graphite that has been irradiated in HFIR. To conclude that we now know how graphite performs in a molten salt reactor requires at least three more sets of information.

For one, one has to recognize the difference in flux level between HFIR and the molten salt reactor. Even in low-flux positions in HFIR, the irradiation is greatly accelerated. How applicable the results are from accelerated irradiation experiments to real time irradiation is a topic that has to be addressed.

Secondly, graphite irradiation in a molten salt reactor happens when this graphite is immersed in salt and exposed to large amounts of fission products that can enter the pores of the graphite and, in some instances, react chemically with graphite. It needs to be determined to what extent HFIR data are applicable to describe shrinkage and swelling of graphite in a molten salt reactor.

Thirdly, a major area of uncertainty is the damage to graphite coming from the very high energy neutrons from the accelerator. In terms of damage, the fusion program might have developed some level of understanding. However, how such a damaged graphite behaves in a molten salt environment is unknown.

There are several other issues of importance that need to be addressed², such as the irradiation sample size vs the real size of the graphite structure, the differences in temperature gradient across the sample vs the "real" value, and the fact that in the reactor, the graphite is exposed to flowing salt that in case of the formation of cracks can enter the graphite body with unknown consequences; there could also be some level of corrosion.

The same line of reasoning can be applied to material properties of irradiated graphite, obtained from HFIR irradiation and obtained from graphite irradiated in a molten salt reactor. While graphite is present in the core only to form flow channels, it has to meet certain strength criteria because, at the very least, it has to support its own weight.

Another issue concerns the wetting of graphite. In the literature are references to the effect that during filling and draining, i.e., in the presence of a gas phase, there can be some wetting of the graphite. The significance of this observation needs to be understood.

While the MSRE experience will be very valuable one has to remember that the peak graphite damage flux in the MSRE was considerably lower than the value calculated for the MSBR (the power density in the MSRE is about one-sixth of that in the MSBR) and the equivalent full-power hours of operation were 13,172 hrs. The resulting damage in the graphite was very low. However, it should also be noted that during the MSRE a small reactivity imbalance was observed between the measured reactivity value and the calculated eigenvalue, that grew and was attributed to the graphite shrinkage.

It needs to be determined whether there should be one program plan that deals with both design and safety issues, or two reports addressing the subjects separately. Either approach as pluses and minuses.

While references 1-3 are very useful in guiding the thoughts on needed development work - and I already recommended to exercise some care in using all of the information provided - even greater care should be exercised on issues discussed in those reports that are related to safety. None of the molten salt systems has ever been subjected to regulatory review and the approach to safety often reflects the thinking of the 1960's rather than today's safety approach. For example, reference 3 states on page 91:

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² how the very high energy neutrons affect the modified Hastelloy N is still unknown. It is conceivable that these neutrons cause increased embrittlement due to high-energy (n, α) reactions (e.g., in Ni)

"The breach of the reactor primary system boundary, resulting in a spill of highly radioactive salt into the primary containment, will probably provide the design-basis accident. The analogous event in a solid fuel reactor would be a major cladding failure".

"In the event of a salt spill, a possibly redundant system of drains would be activated to channel the salt to the continuously cooled drain tank. The system primary containment, which is defined as the set of hermetically sealed concrete-shielded equipment cells, would probably not be threatened by such a spill, but cleanup operations would be difficult".

The first statement is incorrect in regard to the analogy to solid fuel reactors. In case of a major cladding failure, all gaseous and some solid fission products are released to the coolant. It is this accident state for a solid fuel reactor that is (mildly) analogous to the normal state of operation of a molten salt reactor. The qualifier is necessary to account for the fact that fuel pellets retain a large amount of fission products (nearly all the solids, and some of the gases). However, even if one assumes that all fission products are released to the coolant, an analogy between the molten salt reactor fuel salt spill and a solid fuel reactor fission product release to the containment requires an additional barrier failure in the solid fuel reactor, namely a breach of the primary system boundary.

In terms of its regulatory impact the claim in reference 3 is also incorrect. Design basis events are those with probabilities down to 10^{-4} . To release highly radioactive material to the containment whose failure is typically assumed to be in the 10^{-1} range, would lead to a probability of 10^{-5} for a major radionuclide release to the environment. This is totally unacceptable. Therefore, a spill of fuel salt into the containment has to be considered a severe accident and not a design basis accident. To drain the spilled salt into a drain tank is not the proper remedy for such a severe accident. We design reactors today so that under the most severe conditions not more than a few hundred Curies are released to the containment. A salt spill could represent millions of Curies.

RECOMMENDATIONS

The development of DDNs and TDTs is an expensive and time-consuming exercise that needs a tie-in to design efforts. Those design efforts need to be conducted in a top-down fashion as explained before. For preliminary estimates on development cost a condensed development plan should be prepared that deals with the key development tasks shown in Tables 1 and 2.

Especially for the development of cost and schedule estimates it is necessary to engage industry. For example, nuclear graphite would have to come from non-domestic sources. Modified Hastelloy N has only been under development at ORNL, and additional work is needed. After the composition of this alloy has been fixed,

specifications have to be developed and the procurement of commercial heats has to be initiated. Those commercial heats (e.g., four heats at 10,000 lb each) would have to be tested and evaluated with respect to weldability, mechanical properties, irradiation behavior, compatibility with fuel salt containing fission products and coolant salt, and physical properties. Test durations are typically at least 10,000 hrs. These test results will be analyzed and correlated to obtain analytical correlations for use by the designers. For the development of high-temperature design methods, model tests need to be performed. At this time, the documentation can be prepared for an ASME Pressure Vessel Code Case for use of the material in nuclear service. Such a code case can extend over many years.

This means that many years (10?) will pass before even a demonstration test with fissile material can begin. To design a (nuclear) experiment and finish construction within three years, as presented to the JASONS, is overly optimistic and can be realized only through extensive waivers by DOE.

I suggest for your consideration to have the LANL accelerator experiment, that is currently proposed, performed strictly as a **non-nuclear experiment**. There is plenty that can be learned about molten salt, graphite and Hastelloy N technology, instrumentation and control, flow behavior and a coupling of an accelerator to a molten salt blanket. Such an experiment would eliminate any nuclear safety review and nuclear material qualifications that can take many years (like 10 years?). Furthermore, this approach would allow the re-introduction of molten salt technology at a low risk.

Furthermore, it should be investigated to what an extend uranium-containing salts can simulate the performance of Pu-containing salts so that, for example, loop experiments with fuel salt can be done with uranium salt. The presence of fission products can be simulated by the use of stable isotopes (e.g., stable rare earths). This would bring the radionuclide inventories to zero.

TABLE 1: LIST OF RECOMMENDED DDN GROUPS

1. Blanket System
2. Vessel Systems
3. Primary Heat Transport System
4. Secondary Heat Transport System
5. Residual Heat Removal System
6. Cover and Off-Gas Systems
7. Drainage System
8. Instrumentation
9. Control and Protection System
10. Component/System/ Structure Maintenance, Repair, and Replacement
11. Tritium Handling
12. Salt Cleanup
13. Refueling System
14. On-site Waste Management
15. Containment and Cell Heating
16. On-line monitoring and surveillance
17. Accelerator Systems
18. Accelerator-Blanket Interface

TABLE 2: TECHNOLOGY DEVELOPMENT TASKS

1. properties (structural materials, graphite, salt)
2. structural material behavior and development
3. graphite behavior and development
4. salt behavior and development
5. neutronics
6. flow behavior
7. chemistry
8. processing
9. analysis methods development and validation
10. instrumentation and control
11. development, construction, and testing of components, systems and structures
12. instrumentation and control
13. accelerator

SAMPLE OF DDN LISTING

1.0 BLANKET SYSTEM

1.1 molten salts development and characterization

1.1.1 fuel salt chemistry

1.1.1.1 solubility of actinide oxides

- 1.1.1.1.1 solubility of plutonium oxide
- 1.1.1.1.2 solubility of neptunium oxide
- 1.1.1.1.3 solubility of uranium oxide
- 1.1.1.1.4 solubility of other higher actinide oxides

1.1.1.2 solubility of rare earth elements

1.1.1.3 solubility of lumped fission products

1.1.1.4 phase behavior of PuF_3

1.1.1.5 fuel-coolant interaction

1.1.1.6 physical property determination

- 1.1.1.6.1 thermal conductivity
- 1.1.1.6.2 specific heat capacity
- 1.1.1.6.3 surface properties (surface tension, interfacial tension between salt and graphite)
- 1.1.1.6.4 vapor pressure and vapor composition
- 1.1.1.6.5 density
- 1.1.1.6.6 viscosity

1.1.1.7 $\text{Pu}^{3+}/\text{Pu}^{4+}$ equilibrium

1.1.1.8 fluorides of higher actinides

1.1.2 coolant salt chemistry

1.1.2.1 phase diagrams

1.1.2.2 physical properties determination

- 1.1.2.6.1 thermal conductivity
- 1.1.2.6.2 specific heat capacity
- 1.1.2.6.3 surface properties (surface tension, interfacial tension between salt and graphite)
- 1.1.2.6.4 vapor pressure and vapor composition
- 1.1.2.6.5 density
- 1.1.2.6.6 viscosity

1.1.2.6.7 stability under radiation

- 1.1.2.3 composition stability vs BF_4 partial pressure
- 1.1.2.4 tritium trapping and release mechanism
- 1.1.2.5 behavior of oxide and hydroxide ions in fluoroborate melts
- 1.1.2.6 behavior of corrosion products in fluoroborate melts

1.1.3 fission product chemistry

- 1.1.3.1 analysis of MSRE and post-MSRE data
- 1.1.3.2 solubility in coolant salt
- 1.1.3.3 noble metal chemistry
  - 1.1.3.3.1 redox chemistry as a function of $\text{Pu}^{4+}/\text{Pu}^{3+}$ ratio
  - 1.1.3.3.2 reaction with oxide ion
  - 1.1.3.3.3 reaction with graphite
  - 1.1.3.3.4 noble metal agglomeration and deposition
  - 1.1.3.3.5 distribution of noble metals between molten salt and helium
- 1.1.3.4 iodine distribution
- 1.1.3.5 cesium distribution
- 1.1.3.6 removal of rare earth fluorides
- 1.1.3.7 tritium behavior
- 1.1.3.8 distribution of fission products in the primary system

1.2 graphite development and characterization

- 1.2.1 multi-axial strength data
- 1.2.2 fatigue data
- 1.2.3 mechanical property data
- 1.2.4 irradiation-induced dimensional change data
- 1.2.5 irradiation-induced creep data
- 1.2.6 thermal properties data
- 1.2.7 fracture mechanics data
- 1.2.8 corrosion data
- 1.2.9 corrosion effects on graphite design properties
- 1.2.10 corrosion data for methods validation
- 1.2.11 permanent reflector graphite properties data
- 1.2.12 graphite destructive and non-destructive examination data
- 1.2.13 graphite coke source qualification
- 1.2.14 sealing graphite to reduce permeability

1.3 metals development and characterization

1.3.1 modification of Hastelloy N

- 1.3.1.1 modified with Nb
- 1.3.1.2 effect of Ti addition
- 1.3.1.3 reactions with fission products (e.g., Te)
- 1.3.1.4 optimization

1.3.2 material and irradiation properties of modified Hastelloy N candidates

- 1.3.2.1 reactor neutron damage
- 1.3.2.2 accelerator neutron damage
- 1.3.2.3 effect of presence of salt

1.3.3 usability of stainless steel

1.3.4 material corrosion

- 1.3.4.1 corrosion processes (incl. reactions with fission products)
- 1.3.4.2 corrosion depth
- 1.3.4.3 effect on material properties

1.3.5 fabricability (e.g., welding, joining)

1.3.5 limiting conditions (e.g., temperature, fluence, stress, etc.)

1.3.6 ASME pressure vessel code approval

1.4 salt monitoring and surveillance

1.5 tritium behavior

1.6 data for validation of blanket neutronics behavior

1.7 upper, lower and side reflector neutron shielding validation data

1.8 fuel channel flow data

1.9 pressure drop data for primary circuit

1.10 flow mixing data

1.11 blanket cross flow data

1.12 blanket fluctuation data due to mechanical/thermal movements

1.13 blanket graphite column vibration data

1.14 control rod vibration data

1.15 graphite blanket support strength data

1.16 blanket element dynamic strength data

1.17 blanket element failure mode data

1.18 blanket component seismic load data

1.19 fission product transport

- 1.19.1 gaseous fission product solubility in fuel salt
- 1.19.2 gaseous fission product solubility in graphite blocks and graphite support structures
- 1.19.3 effectiveness of helium sparging of gaseous fission products
- 1.19.4 radionuclide (e.g., noble and semi-noble fission product) plateout/-deposition characteristics on metals and graphite
- 1.19.5 radionuclide reentrainment characteristics
- 1.19.6 tritium transport in blanket materials
- 1.20.7 tritium permeation of IHX tubes
- 1.20.8 tritium permeation if steam generator tubes
- 1.20.9 fission product transport validation data

1.20 seismic qualification of neutron control assembly

1.21 in-service inspection (ISI)

- 1.21.1 graphite core support
- 1.21.2 blanket lateral restraint design verification
- 1.21.3 upper and lower plenum design verification

2.0 VESSEL SYSTEMS

2.1 NDTT shift data for blanket and IHX vessel materials

2.2 properties of vessel material

DDN SAMPLES

ABC-DDN 1.2.8: Graphite Corrosion Data

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The graphite blanket components may be corroded by fuel salt impurities, fission products reacting with the graphite, and through corrosion caused by the flowing fuel salt. The effect on corrosion of neutron damage to graphite is unknown. The graphite structures inside the blanket vessel define the fuel salt flow paths in the blanket region and a deterioration of those lifetime structures could affect the fuel salt flow paths and graphite lifetime.

Similarly, coolant salt and its impurities may interact with graphite structures leading to a deterioration of those structures.

Related DDNs are: ABC-DDN 1.2.9 (corrosion effects on graphite design properties) and ABC-DDN 1.2.10 (graphite corrosion data for methods validation).

2. STATE OF THE ART

Graphite is not expected to react chemically with molten fluorides. The MSRE operation did not indicate problems with graphite corrosion; none were expected for the MSBR based on MSRE and corrosion loop data. Graphite is not wetted by molten fluoride mixtures which should greatly reduce any mechanical interactions.

A variety of graphites are of potential interest to a molten salt blanket, among them N3M, IG-110, AXF-8Q, and S-2020. In regard to neutron-induced shrinkage and swelling these candidates might allow long lifetimes. Their corrosion behavior in a PuF_3 -containing salt with high concentrations of fission products and certain levels of impurities is unknown. Enhancement of corrosion due to irradiation-induced shrinkage or swelling has not been studied for high fluences.

3. DESIGN AND ANALYSIS DATA NEEDS

Correlations are needed that describe the corrosion behavior of the (still to be selected) reference graphite in relation to temperature, salt impurities, fission product content, salt flow rates and possibly neutron damage. No graphite

degradation is expected during the first few years of operation. However, because the graphite structures are expected to be lifetime structures, their corrosion behavior needs to be characterized for long-term use.

4. DESIGNER ALTERNATIVES

Graphite corrosion is not expected to be a limiting constraint. Should graphite corrosion become an issue, earlier salt processing than every 10 years might be considered. Graphite replacement is another option. Graphite corrosion disappears as a potential issue altogether in case of an unmoderated system.

5. REQUIRED EXPERIMENTAL WORK

The corrosion characteristics of the reference graphite in the presence of molten salts needs to be determined. To avoid the formation of plutonium carbides, tolerable $\text{PuF}_3/\text{PuF}_4$ ratios and O^{2-} concentration limits need to be determined. The interaction of (especially) noble metal fission products with graphite at usable $\text{PuF}_3/\text{PuF}_4$ ratios needs to be determined.

Flow tests should be performed on graphite that had undergone irradiation-induced shrinkage and swelling that caused some cracks in the graphite structure. Corrosion rates should be determined.

If available, such tests should also be conducted on graphite that has been exposed to very high energy neutrons (from an accelerator).

For the coolant salt NaF-NaBF_4 , the corrosive effects of steam inleakage need to be determined, as well as the effects of salt impurities. This will require the long-term operation of a demonstration loop that can simulate steam inleakage and coolant repurification.

6. RECOMMENDED TEST FACILITY

A test loop is required to study the effects of steam ingress into coolant salt and subsequent ingress of such a mixture into the fuel circuit.

7. CONSEQUENCES OF NONACCOMPLISHMENT AND FALLBACK POSITION

The use of a unmoderated system should seriously be considered. Most of the graphite issues would be eliminated.

Should corrosion be a problem, more frequent salt cleanup could resolve this problem. Graphite replacement is another option.

8. SCHEDULE REQUIREMENTS

TBD

9. PRIORITY

low

Graphite corrosion is an issue that needs to be addressed but it does not seem to be a feasibility issue or even an issue that could severely impact on the ABC design.

ABC-DDN 1.1.1.5: FUEL - COOLANT INTERACTION

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The ABC design has to be capable of dealing with the consequences of a tube rupture in the primary heat exchanger that leads to a mixing of coolant and fuel salt (anticipated operational occurrence). It also has to be capable of dealing with such an incident that follows a steam generator tube rupture that leads to steam ingress into the coolant salt with subsequent IHX tube failure that leads to a mixing of coolant salt containing steam, and fuel salt (design basis accident).

2. STATE OF THE ART

A eutectic mixture of NaBF_4 and NaF is proposed as coolant in the secondary circuit because of its low melting point (74°C lower than for LiF-BeF_2), lower cost than $^7\text{LiF-BeF}_2$ used in the MSRE, and its capability of trapping tritium. It is known that fluoroborates show an appreciable equilibrium pressure of gaseous BF_3 at elevated temperatures and that they are not ideally compatible with either steam or the fuel salt. How the presence of corrosion products affects the eutectic mixture is not known; unknown is also the behavior of the coolant when fuel salt containing fission products is added.

3. DESIGN AND ANALYSIS DATA NEEDS

To properly design the fuel salt and coolant systems, their interactions must be known qualitatively and quantitatively. Areas of concern are composition stability vs BF_4 partial pressure, tritium trapping and release mechanism when coolant and fuel salt mix, behavior of oxide and hydroxide ions in fluoroborate melts and the behavior of corrosion products in fluoroborate melts when fuel salt and coolant salt mix.

4. DESIGNER ALTERNATIVES

While LiF-BeF_2 is a proven coolant based on MSRE experience its major disadvantages are the high melting point and the high cost due to the enrichment of Li. If natural Li would be used, any mixing of coolant with fuel salt would require to discard the fuel because an isotopic separation of the Li-7 would be too expensive.

Other coolant choices need to be evaluated.

5. REQUIRED EXPERIMENTAL WORK

Two types of experimental work are required. One set of experiments deals with an understanding of the basic chemistry of the fuel-coolant interaction. Of particular interest are non-equilibrium conditions. A second set of experiments studies these interaction phenomena in a test loop environment where coolant and fuel salt are circulating with the capabilities to inject fuel/coolant salt, respectively, control the BF_3 pressure and repurify the coolant/fuel salt.

6. RECOMMENDED TEST FACILITY

A flow test loop is required for both fuel and coolant salt experiments

7. CONSEQUENCES OF NONACCOMPLISHMENT AND FALLBACK POSITION

If the experiments show that the fuel-coolant interaction based on the current choice, namely NaF- NaBF_4 coolant, are unacceptable (for example, because of high pressure surges or drastic changes in the liquidus temperature when steam is injected or high levels of Pu precipitation) another coolant has to be selected.

8. SCHEDULE REQUIREMENTS

TBD

9. PRIORITY

High

The selection of the coolant is a basic design decision. At this time, only a limited number of experiments has been conducted at ORNL on the fluoroborates that have not indicated a feasibility problem. However, accident phenomena like fuel-coolant interaction have not been studied in detail yet.

REFERENCES

1. "Program Plan for Development of Molten Salt Breeder Reactors", ORNL-5018 (December 1974)
2. "Development Status and Potential Program for the Development of Proliferation-Resistant Molten-Salt Reactors", ORNL/TM-6415 (March 1979)
3. "Conceptual Design Characteristics of a Denatured Molten-Salt Reactor with Once-Through Fueling", ORNL/TM-7207 (July 1980)

SAFETY-RELATED FUNCTIONS OF THE CONTROL AND PROTECTION SYSTEM

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In the following an attempt is made to present a summary of a series of safety-related functions of the Control and Protection System of the ABC. This summary is based solely on engineering judgement. It is not the result of an integrated design approach where overall objectives are translated into requirements, functions that need to be performed, system/component/structure tradeoffs and design selections.

The dividing line between "control" and "protection" is not well defined and a different approach for plant protection, control, and instrumentation might very well be called for, that is derived from a top-down approach for the design. The ABC system is far more complex in its requirements for control, protection, and instrumentation due to its linkage to an accelerator and, to a lesser extent, the systems required to cope with a molten salt fuel, such as the drainage system, the off-gas system, salt processing systems, cell heating systems, etc.

The emphasis in this report is not so much on finalizing control and protection categories but to identify issues that have to be addressed in the design of the control and protection systems.

How the terms "safety-related" has to be interpreted is another issue that has to be addressed. At one time it was related to 10CFR100 and/or the PAGs, but other definitions are possible.

Whenever references are made to systems and components which have not yet been developed for the ABC, they are assumed to be similar to those developed for the MSBR for which considerable design detail is available. While the MSBR might not be an optimal choice, it is the only one available in the absence of documented design detail on the ABC. As a minimum, the following will identify the issues that need to be addressed for the ABC. However, it will also be shown how they could be addressed.

1.0 REACTIVITY CONTROL ISSUES

The following lists safety-related reactivity control issues for normal operation and emergency shutdown.

REACTIVITY CONTROL

1.1 normal operation

- blanket is subcritical; steady-state operation is assured through an external neutron source

Contrary to conventional reactors where reactivity changes are always the result of changes inside the core which are controlled through control rods that are also located inside the core, the ABC control system has also to account for events that happen outside the core (blanket), namely in the external source. To illustrate this difference, assume that a conventional MHTGR is subcritical but made critical through a steam bypass through the core. To maintain steady-state operation, the reactivity in the core has to be controlled through control rods in the core, or a control of the flow of steam through the core, or a combination of both.

At this time it has been proposed by LANL to control the ABC system solely through the control of the accelerator. This control mode is proposed because the blanket is subcritical and steady-state operation is maintained through an external source provided by the accelerator. When this source is turned off, the blanket is immediately rendered subcritical.

I do not think that this approach will pass regulatory review.

Firstly, even if reliability assessments for the accelerator would show that there is a 100% certainty that the accelerator control is 100% reliable, the reliability data on which such an assessment is based would be questioned because this is a first-of-a-kind system and the ACRS has claimed before that there has to be a protection against the unknown.

Secondly, the primary function of the accelerator control system is the control of the accelerator with its many support systems, among them a target cooling system whose failure could impact on the blanket system. While controlling the beam of the accelerator would also control the source strength and neutron multiplication, the NRC expects a dedicated control system, i.e., not a system to which the control of

reactivity (neutron multiplication) is assigned among many other tasks.

The features of a reactivity control system have to be independence, diversity, and redundancy.

- reactivity changes due to normal operation need to be controlled (see Table 1 for breakdown of control requirements for MSBR as an example)

A reactivity margin of 0.2% is listed for "flow stoppage". Due to the circulation of the fuel, a loss of delayed neutrons by precursor decay outside the blanket (core) significantly reduces the effective delayed neutron fraction. In the MSRE, the effective fractions were 0.0045 and 0.0017, respectively, for the U-235 and U-233 fuels.

- assurance of negative temperature and void coefficients

There is no reactor in the world today in operation that has a positive prompt temperature coefficient. The NRC has not yet decided on the licensability of the ALMR and CANDU-3, both of which have positive void coefficients. It should be noted that the sparging of the fuel salt to remove gaseous fission products will leave a certain amount of helium in the fuel salt, i.e., a certain void fraction. Under normal operating conditions, these voids can grow or collapse.

During the approach to full power for the MSRE with U-233 loading, sporadic increases in nuclear power (5-10%) were observed, for a few seconds with a frequency of around 10 per hour. These perturbations increased reactivity briefly by 0.01% to 0.02% $\delta k/k$. They were attributed to changes in the gas volume in the fuel loop.

- assurance that fission product buildup in the blanket will not change the sign of those coefficients

I consider this a touchy issue because fission products will be present in the blanket in very high concentrations. During a full-power year, a 500 MWt module produces about 200 kg of fission products. Over a 10 year period, it would produce 2,000 kg of fission products. It had been reported that most of those fission products (1,260 kg) would either be removed through the off-gas system or plate out, leaving 740 kg of fission products in the fuel salt together with about 230 kg of plutonium. Regardless how well individual microscopic fission product cross sections have been measured, I expect the temperature dependence of neutron absorptions for such a lump of fission products to be largely unknown.

- maintenance of required subcriticality in primary system

Reactivity control has been shown in the MSRE to be very easy and reliable. The issues for the ABC system in this regard are the assurance of negative temperature and void coefficients throughout life.

Negative temperature and void coefficients at beginning of fuel life are achievable through judicious salt composition and neutron spectrum selection and/or the addition of certain materials that would show an increase in neutron absorptions as the temperature rises. However, as just pointed out, there are large uncertainties in the prediction of those parameters in the presence of large amounts of fission products. Before one can prescribe remedies for the maintenance of a specific subcriticality one has to define, however coarsely, the magnitude of the (potential) problem.

1.2 emergency shutdown

There is no compelling reason for listing the emergency shutdown capability under "reactivity control" and not "plant protection system". The reactivity control capability addressed here goes beyond that usually provided in the reactivity control system (Table 1). It is part of a safety and protection system. The emergency shutdown deals with the following situation:

- unforeseen and emergency situations such as major changes in salt composition, temperature effects when filling the primary system, flow stoppages in the circulating loops, gross temperature changes

If the reactivity control needs exceed those provided by the reactivity control system for normal operation, additional reactivity can be provided rapidly by

1. safety rods
2. shutdown of the accelerator

TABLE 1:
REACTIVITY CONTROL

REACTIVITY TO BE CONTROLLED	ORNL SINGLE FLUID MSBR DATA
cold-to-hot reactivity margin (to compensate for a 150°F temperature change)	0.075% $\Delta k/k$
accounts for Doppler reactivity, graphite expansion, salt expansion	
Xe reactivity override, low to full power	0.3% $\Delta k/k$
gas entrainment (Xe, He bubble formation/collapse)	0.2% $\Delta k/k$
sudden salt flow stoppage	0.2% $\Delta k/k$
sudden large load reduction or loss of load	0.05 - 0.1% $\Delta k/k$
operational margin	0.06% $\Delta k/k$
(normal load change of 100 to 50% at a rate of 5%/min)	(-0.0053% $\Delta k/k$ per minute)
long-term reactivity changes (e.g., burnup)	continuous adjustment of fuel concentration
total control required	0.3% $\Delta k/k$
(the sum of all reactivity effects is 0.85% $\Delta k/k$; however, all the effects will not occur simultaneously, and some will be of opposite sign)	

2.0 PLANT PROTECTION SYSTEM

The plant protection system is a highly complex system whose functions range from safety protection to fire detection and meteorological monitoring. The following is a list of safety-related issues, besides reactivity control, that need to be addressed in the design of this system.

2.1 reactivity reduction

- reactivity disturbances beyond the capability of the normal operation reactivity control system, such as malfunction of the control system, accidental large additions of reactivity, gross loss of core cooling capability

The Molten Salt ABC has to provide for diverse, independent, and redundant shutdown capabilities:

1. normal blanket shutdown via normal operation reactivity control system (control rods, beam tube shutoff)
2. emergency shutdown by the following means:
  - a. safety rods
  - b. accelerator shutdown
  - c. fuel drainage

2.2 control of fast shutdown

- loss of primary or secondary flow in a loop and resulting salt freezing

A fast-acting load and power reduction system is required to avoid a total shutdown of the plant and to permit resumptions of normal operation when conditions permit.

If primary flow is reduced, both the secondary-salt and steam flow must be reduced to prevent low salt temperatures. If the secondary flow is reduced, the associated steam flow must be reduced but the associated primary flow need not be reduced.

2.3 load control

- allowable rates of load changes need to be determined
  - one of the issues are excessively high stresses in components and structures, that are in contact with salt, due to rapid temperature changes

The ABC is not uniquely restricted by high thermal stresses. Based on allowable stresses, the maximum allowable rate of load changes will be determined.

- full and partial load operation
  - assurance of acceptable primary and secondary loop salt temperatures during load changes to avoid salt freezing and high thermal stresses

Plant loop control can be accomplished by the use of a steam temperature controller and a reactor outlet temperature controller. Any deviations in temperatures from their set point will result in flow adjustments in the respective loops.

2.4 temperature control (protection against salt freezing)

- loss or reduction of salt flow in any loop
- mismatch between heat produced and heat removed

It was demonstrated in the MSRE that measuring appropriate temperatures, flow rates and power balances is not a particularly difficult control problem for protective actions such as stopping circulation in various loops and shedding parts of the load. In case of an emergency shutdown, an immediate reduction of the load to the decay heat level is required.

2.5 fuel drain

- controlled fuel salt draining

Draining of fuel salt is the ultimate shutdown mechanism for the ABC blanket.

The fuel drain system can be part of a highly effective emergency shutdown and residual heat removal system because removing fuel from the reactor will not only ensure subcriticality but this fuel will also be reconfigured in the drainage tank so that residual heat can be passively removed; i.e., the fuel drain system deals with reactivity control and, more importantly, with residual heat removal.

The drain system is not required to rapidly initiate the dumping of the fuel.

2.6 off-gas system control

Reduce the concentration of undesirable contaminants, such as gaseous and gas-borne fission products, fission product daughters, water, oxygen, hydrocarbons, etc., in the primary system off-gas stream.

A salt mist is produced in the pump bowl by the xenon stripper spray that can drift into the off-gas line and freeze. This frozen mist gradually builds up and can lead to a plugging of the line. The free flow of gases through the off-gas lines needs to be monitored.

2.7 startup, standby, and shutdown procedures

- avoidance of salt freezing during startup, standby, and shutdown

Procedures need to be developed to ensure that salt freezing is avoided.

A normal startup from the cold-empty condition of the MSBR would have begun with turning on the primary and secondary cell heaters, and the primary and secondary circulation pumps circulating helium in the salt systems. When the temperature of the secondary system reaches 850 F, the loop is filled with coolant salt from the heated drain tank, and salt circulation is started. When the primary system reaches 1000 F, it is filled from the fuel-salt drain tank, and salt circulation begins. Both salt systems will continue to be circulated isothermally at 1000 F until power escalation begins. The reactor is made critical at zero power by removing safety rods and further addition of reactivity by insertion of graphite control rods [note: without those rods, the reactor is undermoderated, i.e., inserting them adds reactivity]. The automatic neutron flux level controller is used to control the power.

At the same time the salt systems are being electrically heated, the steam system is warmed and brought to operating conditions by an oil- or gas-fired auxiliary boiler. Steam is taken from this auxiliary boiler for deaeration and demineralization of the feedwater and warmup of piping, feedwater heaters, turbines, etc. When the auxiliary boiler reaches full pressure and temperature, circulation can be started through the steam generator.

The normal shutdown requires the system power to be reduced until about 8% of full-load power is reached. During this time, the flow to the main turbines is gradually reduced to zero while transferring the generated steam to the turbine condenser. To remain in a hot standby condition requires the start of the auxiliary boiler and for a new startup, to proceed as described before. If an extended shutdown is expected, then after about 10 days of decay heat removal, the fuel salt will be transferred to the drain tank. The cell heaters will maintain the cell temperature high enough for the coolant salt to remain in the molten condition. The steam system is allowed to cool after all steam generation is terminated.

2.8 accelerator-initiated blanket shutdown and startup

- accelerator-initiated shutdown
- inadvertent rapid accelerator-initiated blanket criticality

In case of a rapid shutdown initiated by the accelerator shutoff, loads have to be reduced immediately to the decay heat level.

To avoid inadvertent hot criticality due to an accelerated-initiated startup, the insertion of the safety rods during shutdown will prohibit such a criticality event. Furthermore, accelerator controls will prohibit sudden full-power operation of the accelerator.

2.9 instrumentation

- availability of reliable instrumentation and controls for high-temperature operation

MSRE operation provided valuable design and operating experience with molten salt reactor nuclear and process instrumentation. Further development work, especially on electrical insulating materials, might be required.

3.0 CHEMISTRY CONTROL - SALT SURVEILLANCE

This function is unique to the molten salt reactor. While all reactors monitor coolant chemistry, this system has to monitor both coolant salt and fuels salt.

3.1 fuel salt monitoring

- fuel salt composition and phase behavior

The primary concern is that Pu (and other trifluorides such as the rare earth isotopes) stays in the salt and that the presence of large amounts of fission products does not change the phase behavior and properties of the salt.

Furthermore, the salt composition needs to be monitored to ensure that the salt properties change as predicted.

- maintenance of a non-corrosive environment

Plutonium is dissolved in the fuel salt as PuF_3 . Operation of the MSRE has shown that the valence of the fission products is slightly less than 4 which should ensure a non-corrosive environment unless there are as of yet unidentified chemical reactions that could provide excess fluorine.

- corrosion products

Excessive levels of iron, nickel, and chromium are indicative of a corrosive environment in the primary system requiring remedial actions.

- rare earth concentration

The concentration of rare earth tri-fluorides has to be limited to less than 1 mole % because at higher concentrations they will precipitate. Unless the rare earth elements are separated from the fuel salt, their concentration in the salt could determine the time at which the salt would have to be discarded.

- noble metal concentration

The noble metal group (Mo, Ru, Te, Nb) does not form stable fluorides. MSRE experience showed that about half of the noble

metals plated out on graphite and metal surfaces exposed to the salt. The remaining fraction of the noble metals goes into the cover gas as a smoke or aerosol of sub-micron particles.

3.2 coolant salt monitoring

- salt composition

The salt composition has to be monitored to ensure that its heat transport properties do not degrade.

- impurity levels

Impurities can significantly affect the melting point of the coolant and need to be monitored.

- solubility of tritium

It is expected that tritium will be nearly completely dissolved in the coolant salt thus greatly reducing the tritium release from the plant. However, this solubility needs to be monitored.

4.0 SUMMARY OF SAFETY-RELATED CONTROL ISSUES

NORMAL OPERATION

- maintenance of blanket and primary system subcriticality
- assurance of negative temperature and void coefficients at all time
- emergency shutdown through control rods and/or beam shutoff

PLANT PROTECTION

- reactivity reduction through control rods
- emergency shutdown through control rods and beam shutoff; fuel drainage
- temperature, flow and pressure control
- fuel drainage control
- off-gas system control
- accelerator-initiated blanket shutdown and startup
- reliability of instrumentation
- load control
- control of fast shutdown
- startup, standby and shutdown procedures

CHEMISTRY CONTROL - SALT SURVEILLANCE

- fuel salt monitoring
  - salt composition and phase behavior
  - impurity levels
  - maintenance of non-corrosive environment (control of excess fluorine)
  - corrosion products
  - rare earth concentration (solubility limit of < 1 mole %)
  - noble metal concentration (distribution of deposition, gas phase)
- coolant salt monitoring
  - maintenance of salt composition
  - impurity level
  - tritium solubility
- off-gas system monitoring
  - charcoal beds
  - particle traps
  - removal of contaminants in carrier gas in gas cleanup system

APPENDIX A:

FUEL-SALT DRAIN SYSTEM

FUNCTION

Provide a place where the salt can be safely contained and cooled under normal operating and accident conditions. Certain scheduled operations require the removal of the fuel salt from the primary system, resulting in scheduled drains. Under accident conditions, the fuel salt drain system provides the ultimate shutdown capability resulting in unscheduled drains.

scheduled drains: maintenance operation (e.g., servicing pumps, heat exchanger)
blanket graphite replacement

unscheduled drains: failure of primary system pipe or vessel
slow loss of salt from primary system
loss of heat removal capability in the steam system
loss of coolant or circulation in the secondary loops
loss of power or mechanical failure of primary pumps
inadvertent thawing of the freeze valve which holds the fuel in the primary loop

The drain tank of the Fuel Salt Drain System is linked to the Off-Gas System in that, during normal operating conditions, it receives the off-gas from the entrainment separator that contains radioactive gases and metals, for a 2 hr holdup to allow the radionuclides to decay. Besides tritium, the gases are primarily Kr and Xe, and the noble metals are Nb, Rh, Mo, Ru, Tc and Te. The daughter products of Kr and Xe decay are notably Ba, La, Cs, Rb, Sr, Y, and Zr.

The drain tank also serves as a sump for the overflow streams from the bowls of the salt-circulation pumps. A combiner tank combines the gases from the separator with the salt overflow.

DESIGN REQUIREMENTS

1. the fuel salt in the fuel drain system must be in a subcritical configuration at all times
2. maximum drain tank temperatures must be kept within the safe operating range under all conditions

3. the system must be highly reliable, with a minimum of reliance on electric power supply or operator-initiated actions; to achieve this high reliability, maximum use should be made of passive and inherent safety features
4. the cooling system should impose a minimal risk for freezing either the fuel salt or the cooling system coolant
5. preference should be given to coolants that, when leaked into the fuel salt, do not require chemical processing to prevent adverse nuclear or chemical effects

FUEL SALT DRAINAGE SYSTEM DESIGN

The Fuel Salt Drainage System consists of the following subsystems:

- drain tank
- drain line and freeze valve
- pump and jet system to return salt
- off-gas heat disposal system
- afterheat disposal system
- heater equipment
- fuel salt storage tank

Salt flows from the blanket vessel by gravity, through a freeze valve into the fuel salt drainage tank. There it is cooled by a coolant (e.g., LiF-BeF₂ for the MSBR) by natural convection. This coolant then transfers the heat to a molten salt-water steam generator. The steam generated in this steam generator is condensed in air-cooled coils in a natural draft stack.

The heat transfer from the gas in the tank to the secondary coolant is by conduction.

Estimated heat loads for this system are as follows:

9 MWt	decay from plated out noble metals
9 MWt	decay of radioactive gases and daughters
18 MWt	total maximum steady-state heat load

The most severe heat loads on the drain tank are:

inadvertent thaw of freeze valve
emergency shutdown and drain

The maximum heat load under such circumstances is estimated to about 50 MWt, if 7 minutes are allowed for drainage to take place.

APPENDIX B:

SAFETY CHALLENGES FOR FUEL SALT DRAINAGE SYSTEM

Concern	Possible Remedy
criticality	subcriticality is assured by the large number of coolant pipes (parasitic neutron absorbers) in the drain tank that contain the secondary coolant
salt freezing	to avoid coolant overcooling it might be necessary on some days to preheat the air in the condenser design against any direct contact of fuel salt with water preheat concrete cells that house structures that contain fuel and coolant salts, and maintain high temperature
salt boiling	overdesign by having many separate and independent natural convection circuits
high drain tank temperatures	cool tank walls by overflow salt from blanket
freeze valve operation	the large-diameter drain line (≥ 6 inches) which passes through the freeze valve poses difficulties in regard to cooling the salt in this valve sufficiently under normal operating conditions to avoid accidental draining; a mechanical-type valve might be considered instead valve leakage drains into a catch basin from where the salt is directed toward the drain tank; the environment of the cell where the freeze valve is located needs to be controlled.
noble metal deposition	it is expected that the noble metals deposit on cold surfaces in the drainage system but uncertainties are high in regard to the locations of those deposits; cooling system overdesign can alleviate this concern

APPENDIX C:

General Comment on Safety Challenges to the Fuel Salt Drain System

It might be advisable to separate:

- the drainage function from the
- off-gas decay function for the Fuel Salt Drain System.

Scheduled fuel drainage can be accomplished after allowing for sufficient radionuclide decay in the blanket vessel which would simplify the heat removal system.

While the freeze valve has attractive features in regard to allowing for drainage without operator actions, this function can only be performed when the salt inside the valve is constantly cooled - which requires engineered systems. A replacement of the MSRE-type freeze valve is desirable.

The use of a vessel-in-vessel concept would permit the off-gases to decay in the vessel surrounding the blanket vessel and the relatively low heat input could be rejected into a cavity surrounding the vessel systems, from where the heat is passively removed to the outside environment.

Such a system would also eliminate the need for a catch basin.

APPENDIX D:

OFF-GAS SYSTEM

FUNCTION

Reduce the concentration of undesirable contaminants, such as gaseous and gas-borne fission products, fission product daughters, water, oxygen, hydrocarbons, etc., in the primary system off-gas stream.

RATIONALE FOR OFF-GAS SYSTEM

In a 3000 MWt reactor, about 1.5 kg of noble fission gases are produced per day. 15% of the gaseous fission products are short-lived and decay in the fuel salt system. 85% are either stable or have long half-lives.

Xenon is insoluble in molten salt. Because of the high Xe partial pressure at low concentrations, Xe will tend to leave the salt through any free surface available.

The free surfaces available in the blanket region are voids in the graphite and entrained helium bubbles. Unless the graphite is specially treated, almost all Xe poisoning will result from neutron absorptions in Xe-135 within the graphite. The concentration of Xe in the graphite will depend on:

- Xe concentration in the salt
- mass transfer coefficient from salt to graphite
- total surface of graphite exposed to salt
- diffusion coefficient of Xe in graphite
- void fraction available to Xe

To reduce the adsorption of Xe in the graphite, there are two options:

- treat the graphite to avoid that the bulk void volume in the graphite is available for Xe, and reduce the effective diffusivity
- reduce the Xe present in the salt

The salt will be processed for Xe removal by injecting small bubbles of helium into the fuel stream and then removing them after they have taken up a portion of the Xe present.

OPERATION OF OFF-GAS SYSTEM

For the MSBR, about 10% of the total primary salt flow is bypassed through the 135-Xe removal system which consists of the gas separator and the bubble generator.

The bypassed salt stream is directed through a Gas Separator to remove helium, fission products and some salt from the stream, and this mixture is directed toward the Entrainment Separator. The salt flows through the bubble generator to the pump inlet.

The Bubble Generator adds small helium bubbles (0.5 mm diameter) to the salt. The helium volume fraction in the salt is approximately 0.25%, and the bubbles will be recycled on average about ten times.

In the Entrainment Separator the salt will be separated from the gases and returned to the pump expansion tank. Additional gases, fission products and helium will be drawn off the pump and piped to the entrainment separation. From here, the gases will be sent to the Fuel-Salt Drain Tank for a 2-hr holdup to allow for the decay of short-lived isotopes and the plateout of noble metals.

From the Fuel Salt Drain Tank the gases go through a particle trap, if required, to the Charcoal Adsorption Bed. Here, the Xe is held for 47 hrs and about 80% of the output from this bed is returned through an accumulator to the gas bubbler and primary salt pump.

About 20% of the gas flow goes to a Long-Delay Charcoal Bed for further decay. Over a period of 90 days, all radioactive fission products except the 10-yr ^{85}Kr and tritium are allowed to decay to negligible amounts.

After leaving the long-delay charcoal bed, the off-gas stream enters the Gas Cleanup System for a further cleanup to reduce contaminants to a level that would permit the effluent carrier gas to be recycled to the reactor purge gas system. The tritium is oxidized, and the water, Xe and Kr are removed in storage bottles maintained at liquid nitrogen temperature.

Note: The discussions above apply to a thermal molten salt system. It is not clear yet to what an extent requirements can be relaxed for an epithermal or fast system because Xe adsorptions in graphite would affect less the neutron economy. Furthermore, the discussions above relate to a molten salt breeder reactor where parasitic absorption represent are a substantial penalty for the doubling time. Such considerations do not directly apply to a plutonium disposition system.

APPENDIX E:

OFF-GAS SYSTEM SAFETY CONCERNS

The Off-Gas System cleans up the helium purge gas so that it can be recycled. The design of this system has to address:

- holdup of Xe and Kr to allow for decay
- gathering the fission product particulates
- trapping the tritium
- disposal of collected radioactive materials
- effective cooling of charcoal beds, helium compressors, particle traps, etc. to remove the decay heat

Among the systems requirements are:

- double-walled piping (and better) has to be employed;
- the status of the system has to be tightly monitored;
- the system must be designed for remote maintenance and replacements.

Among the safety concerns are:

- transport mechanism for noble metals is not fully understood
- reliable decay heat removal system
- potential for hot spots due to accumulation of solids and their decay heat production at valve seats, pipe bends, etc.
- charcoal bed efficiency and lifetime
- in case of a fuel - coolant salt interaction, off-gas from fluoroborates contains condensibles that might require special handling

RECOMMENDED ABC ANALYSIS TASKS AT THE PRE-CONCEPTUAL DESIGN LEVEL

April 15, 1994

**Prepared for
Los Alamos National Laboratory
Contract No. 5079L0014-3H**

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RECOMMENDED ABC ANALYSIS TASKS AT THE PRECONCEPTUAL DESIGN LEVEL

1.0 Introduction

To establish a sound technical basis for the ABC blanket design, a series of neutronics and thermal-hydraulics, transient and accident analyses need to be performed at the very beginning of the design work. The results of these analyses will guide the designer in arriving at a pre-conceptual design that will later be refined in the conceptual, preliminary and final design stages.

1.1 parametric studies

The early calculations on any reactor project are **parametric studies**. At the early stage in the design, obtaining the correct trend is more important than obtaining highly accurate data. The computational methods used should reflect this objective. Many calculations have to be performed and both setup times and running times must be short. One-dimensional diffusion or transport calculations and first order perturbation calculations are sufficient for the neutronics analyses. Lumped model analyses are sufficient for the T&H/transient analyses.

1.2 documentation

I strongly recommend to properly document all analyses. The documentation should clearly state:

- the problem that has been analyzed,
- material properties (such as density, thermal conductivity, specific heat) and their source
- the approximations/models
- the cross section data and their source
- computer codes and the sequence in which they have been used
- the conclusions reached.

In case of calculating reactivity changes, convergence criteria should also be reported. The report detail should allow someone else to intelligently review the results, be able to repeat the same calculation and, hopefully, confirm the results.

1.3 quality control and assurance

From the very beginning, a **quality control and assurance** program has to be incorporated. Among other things it has to delineate the process by which design

decisions are reached, assign responsibilities and authorities. It will also ensure that different groups and individuals will analyze the same design.

A Software QA program has to be in place and review procedures must be established. It is important to assign responsibilities for codes and cross section sets to avoid an escalation of individual versions. It is also advisable to begin with the development of analysis methodologies to be used for steady state and transient/accident analyses. These methodologies encompass both the mathematical models and the sequential use of specific computer codes.

This may sound like bureaucracy at its best, but it is nothing but lessons learned. At the beginning of any project, many decisions have to be made, often based on very limited information. The least one has to do is properly document what has been done and have someone else review the results on which design decisions and characterizations are based. Design selections have to reflect a consensus which in turn has to be based on reviewable information. Basing a design on later deciphering of who has said what to whom at what time should be avoided at any cost.

It is highly advisable to spend some time at the beginning of the project on planning which analyses have to be done rather than doing the planning on the fly. This is particularly important during the formulation phase of the project when the demand for just another set of viewgraphs often seems to dictate which analysis has to be done.

1.4 ABC concept

The current ABC concept does not employ fuel reprocessing which is an attractive feature for at least a near-term deployment.

However, it is recognized that the fuel salt has to be removed after 10 years or earlier because of the high level of trifluorides it contains (e.g., rare earth, Pu) that have a solubility of only about 1 mole %. This salt is then either stored or reprocessed.

At this time, there are only hints that it might be possible to remove the rare earth trifluorides, perhaps even continuously. If this is possible it could have an important impact on the ABC design in that an unmoderated system might look attractive, too. The elimination of an unmoderated concept at this time, as far as I know, was based to large extent on the argument that the salt life would be very limited due to the high Pu concentration required to ensure criticality in the presence of large amounts of fission products and due to the buildup of the rare earth isotopes and their formation of trifluorides.

I recommend that LANL management considers the following options:

1. design a *moderated* system, provided the analyses described in the following support such a choice, and benefit from reprocessing technology advances by extending fuel salt life
2. design an *unmoderated* system, provided the analyses described in the following support such a choice, replace the fuel salt initially more frequently until reprocessing technology advances allow more frequent (continuous) reprocessing
3. design a *moderated* system, provided the analyses described in the following support such a choice, and switch later to an *unmoderated* system, the practicality of which has to be assured in the initial blanket design

It should be noted that the presence of Li, Be, and F in very large quantities will already provide for some level of neutron moderation. Adding large quantities of graphite will further moderate the neutrons. However, the addition of graphite brings with it also many potential disadvantages, among them the graphite lifetime issues, fission product adsorptions in graphite and the need for helium sparging to more efficiently remove gaseous fission products, the building of large graphite structures to support the graphite moderator, the much larger blanket vessel, the high negative reactivity worth of fission products, etc. Among the advantages of using graphite besides the lowering of the Pu inventory is the presence of a large heat sink in the blanket and the ability of providing at least some level of outlet temperature flattening.

2.0 Neutronics Analyses

The most important neutronics issue that needs to be addressed for the ABC is the **Doppler coefficient**. It is known that the contribution to the Doppler coefficient from Pu-239 resonance broadening is typically positive. How strongly positive it is and how it is affected by moderation, fertile materials and burnup needs to be determined.

Important for the concept development are also issues such as the required level of neutron moderation, burnup effects and a quantitative understanding of the impact of constraints such as trifluoride solubility limits and fluence effects on structures and graphite life.

Recommendation: Identify life limits for both graphite and metallic structures in terms of limiting fluences (as a function of temperature), maximum allowable dimensional changes, creep strength, fracture strain or other parameters.

The neutronics analyses should provide the following information:

2.1 Doppler coefficient and reactivity

Assurance has to be provided that the Doppler coefficient is always negative. The Doppler reactivity is the only prompt reactivity feedback available for the termination of reactivity transients; all other reactivity feedbacks are delayed. It is important that the Doppler coefficient is strongly negative because there are no experimental data to validate any Doppler coefficient calculation, i.e., sole reliance has to be placed on calculations rather than comparisons of calculations with measurements. A marginally negative Doppler coefficient solely derived from calculations will not be sufficient.

During my visit to LANL in January it had been argued that the Doppler coefficient could be allowed to be positive because the ABC is subcritical and has a fast-acting shutdown system in form of the accelerator. However, a positive Doppler coefficient would render the ABC inherently unstable. Any temperature increase would lead to a reactivity increase which in turn would lead to another temperature increase, leading to another temperature increase, etc..

A positive Doppler coefficient is unacceptable unless the Doppler coefficient is positive only over a narrow temperature range.

Preliminary analyses at LANL showed the results I expected, namely that an ABC blanket containing weapon's grade Pu has a very strong positive Doppler coefficient (DC). The following analyses are proposed:

- determine the DC as a function of:
  - temperature
  - burnup (assuming power densities of 5, 10, 15, and 20 w/cc)
  - Pu composition and concentration
  - fuel salt/graphite volume fraction (homogeneous model)
- assess heterogeneity effects by varying:
  - salt to graphite region thicknesses
  - Pu concentration and composition
  - fission product concentration
  - temperature (should be a minor effect)
- assess the effect of fertile and other materials material on DC:
  - add U-238
  - add Th
  - add a combination of Th and U-238 to ensure that U-233 cannot be chemically separated (keep U-233 concentration < 20%)
  - determine the burnup sensitivity of the DC in the presence of fertile material
  - determine the amount of fertile material that is required to ensure that the magnitude of the negative DC is never smaller during the life of the salt than 1, 2, 3 or 4 x10E-5
  - determine the effect of moderation on the effect of fertile material additions by analyzing a well-moderated system, an unmoderated system and one system in between
  - determine the effect of Li-6 on the DC (note: for the HEU NP-MHTGR, the overall Doppler coefficient was negative because of the strongly increasing neutron absorptions with temperature); if Li-6 has the desired effect, the possibility of not fully enriching Li in Li-7 should be considered
  - determine the effect of erbium and other materials to tailor the Doppler coefficient (GA had claimed that the addition of erbium would provide for a more negative Doppler coefficient)

In assessing the Doppler effect it is also important to estimate the available Doppler reactivity. While a negative Doppler coefficient is necessary for ensuring a prompt negative feedback in case of a temperature excursion, it is not sufficient to ensure that the transient can be turned around effectively. Next one has to consider the available temperature margin, i.e., the temperature increment from normal operating conditions to the limiting temperature condition. The combination of Doppler coefficient and available temperature margin will show how much of a prompt reactivity increase can

be tolerated.

2.2 Moderator Effects

During the January 1994 dryruns for the JASON meeting, two conclusions were presented that have a profound impact on the ABC design:

1. an unmoderated ABC system is not acceptable because of the required high fissile inventory and limitations in tri-fluoride solubility in the salt
2. an over-moderated system is required to ensure sufficient neutron moderation in the presence of a large amount of fission products

No analysis detail was available for review and only plausibility arguments were presented to me. However, based on these analysis results, that were undocumented and unreviewed, an un-moderated ABC system was excluded from consideration and a fuel salt volume fraction of only 5% was selected to ensure over-moderation. Furthermore, a very low average blanket power density of only 5 w/cc was selected. Even without having done any economic analysis for the ABC, such a low power density would clearly represent a serious cost penalty unless it buys considerable savings elsewhere. Those savings have not been identified.

It is important to establish a sound technical basis for any decision in regard to the need for a moderator, the degree of required moderation, as a function of power density with different solubility constraints for the tri-fluorides.

The following parametric calculations should be done:

- determine the tri-fluoride content after a 5-, 10- and 15-year residence time as a function of the moderator fraction (0 - 95%) for different power densities (e.g., 5, 10, 15, 20 w/cc) and plot the results. Continuous refueling should be assumed and the eigenvalue should be kept constant throughout the calculations (e.g., $k = 0.95$)

These calculations will also yield fluence information.

- determine the maximum achievable salt and graphite residence times as a function of moderator fraction using as constraints (1) tri-fluoride solubility (e.g., 1.3, 1.0, 0.7 mol%) and (2) fluence limits; plots should be prepared for different power densities.

These correlations can be derived from the preceding parametric studies.

- assess the effects of lumped moderator vs distributed moderator configurations, e.g.,
  - hexagonal MHTGR block-type moderator with holes for fuel salt (vary number and size of holes)
  - annular core with a large graphite block in the center that surrounds the target
  - graphite blocks arranged on a square pitch
  - concentric fuel rings separated by graphite

These analyses are based on the assumption that a moderator is necessary. It is then important to understand how fuel and moderator regions have to be spaced, i.e., how much of a flux fine structure comes from the heterogeneity. Provided a moderator is needed, it would be very beneficial if it could be provided in large chunks with large fuel spaces in between rather than in form of blocks or full-length elements.

Note: These analyses do not address the issue of block-type moderator or full-length moderator elements.

2.3 burnup effects

It is not clear at all how any validation can be achieved of burnup analysis results. I raised this issue at the January JASON dryruns and was told that LANL has done very good cross section measurements for fission products and, therefore, burnup effects are not an issue.

I disagree with this position. The evaluated nuclear data files (ENDF) in use everywhere are not just a collection of measured cross sections but the measured cross sections have been adjusted as a result of comparisons with measured data (e.g., critical experiment data). We all know of the problems the HTGR community experienced with each new ENDF/B data set, namely the lack of evaluation of the nuclear data with measured HTGR physics parameters. As a result, the HTGR analysts had to do their own adjustments.

These adjustments will be necessary for as long as all measurements have uncertainties (and often biases), and the analysis methods are imperfect, i.e., they use approximations. Fission product cross sections are no exceptions and they also need to be adjusted. Because they are formed in the reactor, sometimes in multi-stage processes, they are burdened with even greater uncertainties than other reactor constituents.

It might be worthwhile to carry out sensitivity analyses at a later stage in the design to assess the effects of uncertainties in fission product performance (based on their

cross sections and concentrations which reflect yield uncertainties and other uncertainties in their formation). However, such analysis should come later because they have only little impact on the design.

The issues that need to be addressed at this stage are:

- Pu disposition in the presence of varying amounts of fertile material and moderator

Based on the assurance of a certain negative DC, the necessary amount of fertile material was determined in section 2.1; different levels of moderation and different power densities need to be considered

- ditto, in the presence of other Doppler enhancing materials, as discussed before
- determine the required Pu feed per day as a function of burnup (salt residence time) and determine the reactivity change if this amount would be suddenly inserted
- determine reactivity increase due to an accidental doubling (tripling) of the daily feed as a function of burnup (salt residence time)
- identify the level of waste transmutation that is achieved
- identify the increased disposition of Pu due to the continuous removal of gaseous and noble metal fission products; assess the effect of this fission product removal on the required fissile inventory
- identify the potential performance penalties due to the adsorption of Xe and Kr in the graphite compared to removal of a certain fraction (parameter variation) of the noble gases
- determine the increase in neutron absorptions in graphite due to the adsorptions of fission products in the graphite

Graphite is very porous and is exposed to high concentrations of fission products, many of them will stay in the graphite.

- perform a literature search to obtain data on graphite properties vis-a-vis fuel burnup

It needs to be determined how graphite swelling, shrinkage, and other properties are changed in the presence of fission products and possibly

salt intrusions. Some of the fission products might also react chemically with the graphite.

2.4 void coefficient

The circulating fuel contains noble gases (from fissions) and helium (from sparging) that can form bubbles and affect reactivity, or the bubbles can collapse and affect reactivity. The void worth (density coefficient) needs to be calculated as a function of space and burnup. The analysis should show a breakdown for the void reactivity, i.e., positive and negative contributions.

Before these analyses can be carried out, however, a better understanding of the physical blanket size and configuration should be available. I don't expect the void coefficient to be a feasibility issue, however, just to be sure that there are no surprises, two void reactivity calculations should be done, one for a moderated system and another one for an unmoderated system.

2.5 expansion coefficients

Any increase in blanket temperature changes:

- fuel salt density
- graphite density and dimensions
- fuel scattering, fissions and absorptions
- graphite scattering and absorptions
- neutron leakage
- bubble growth

To accurately calculate these effects requires considerable design detail (for example, moderator geometry, support structures, pressure control). At the pre-conceptual design stage, it is sufficient to replicate the analysis done for the 2250 MWt MSBR.

2.6 power flattening and outlet temperature equalization

Power flattening is desirable to reduce the peak to average coolant temperature. The conventional means for power flattening, namely enrichment grading in radial and axial direction, cannot be applied to liquid fuel reactors. Using control rods for power shape control is also not possible because the ABC system does not have a conventional reactivity control system. However, a means employed for solid fuel reactors, namely using different-size fuel pins, can be translated into a change in neutron moderation for liquid fuel reactors through the change in the amount of graphite in different regions. This method had been employed for the MSBR. It aims at better equalizing coolant channel outlet temperatures. This method should be investigated, however, not at the project definition stage.

It has been proposed to use flow orificing to achieve the goal of outlet temperature equalization. Orificing is a proven means for outlet temperature equalization in (solid) fuel assemblies with at the most a few years of residence time. It is done in addition to flattening the power distribution via enrichment grading. If measurements indicate that the predicted effect of flow orificing differs from the measured effect, orifices can be changed for the next fuel batch. Because of the short lifespan for the orifices in the reactor, mechanical deterioration is usually not an issue. However, such an assurance is not possible for orifices that are expected to last a plant lifetime. Such flow readjustments cannot easily be accomplished when orificing is achieved through structures that will stay in the reactor for the life of the plant. Furthermore, there is always a desire to keep in-core instrumentation to a minimum because it would be placed in a "hostile" environment with difficulties for replacement. If only the mixed mean outlet temperature is measured, this provides a measure, together with the fuel/coolant flow rate, for the power level, but not the effectiveness of flow orificing.

Power flattening through changes in neutron moderation is usually not very effective. However, analysis should show how much of a power flattening can be accomplished for changes in the fuel/graphite ratio.

Note: This type of power flattening is possible only in the moderated system.

It should be noted that a 30-year graphite life is an objective that needs to be carefully assessed. We know that the salt will not last 30 years but that it needs to be processed after 10 years or even earlier. We also know that it is desirable to keep the peak to average outlet temperature ratio as small as possible because we have limits for the maximum salt temperature in the primary system. Orificing could be very useful if we have assurance that it works as expected. Unloading the graphite already after 10 years or earlier and replacing the orifices might be an effective way of reducing outlet temperature penalties.

2.7 fluence/shielding

No consideration has been given so far to fluence and shielding concerns. We know from the shielding analyses for MHTGRs that just adding boronated graphite will not significantly reduce the fast neutron dose on the vessel. The boron will filter out nearly all of the thermal neutrons but the graphite is not very effective in slowing down the neutrons over a short distance. Inelastic scatterers like iron are very useful in slowing down the neutrons and then having them absorbed in boron (or the likes).

Different graphite reflector/shield configurations need to be analyzed with, for example, boronated iron pins.

At the early stages in the design definition, no detailed shielding analyses are required. However, shielding requirements have to be kept in mind when blanket vessel sizes

are considered. As a rule of thumb, at least 3 ft of reflector thickness should be available for shielding purposes (vessel, IHX, pump fluences need to be considered).

2.8 control rod worth and excess reactivity requirements

The current ABC design picture does not show control rods. The accelerator is the only means to control reactivity and to shut down the blanket. The accelerator represents both the control and the protection system.

Based on current regulations, this configuration will not be acceptable. Control and protection systems have to be redundant, diverse and independent. A single reactivity control system cannot meet those requirements.

During my visit to LANL in January, it was claimed that the control rod worth in such a high-fission product environment would be very low. While this result is expected, the results of the calculations should be presented to quantitatively assess this worth reduction. Both moderated and unmoderated configurations should be investigated.

In the early stages of the design, reactivity control requirements should also be determined. Among the contributors would be the cold-to-hot reactivity margin, Xe decay, Pa-233 decay if Th is used, and a shutdown margin.

2.9 effective delayed neutron fraction and neutron lifetime

Pu-239 has already a very low delayed neutron fraction which will be further reduced due to the fuel circulation. For transient calculations, both effective neutron fraction and neutron lifetime are required and need to be calculated.

3.0 Thermal-Hydraulic, Transient and Accident Analyses

At the early design stage, no detailed T&H analyses are required. It is important to **collect and evaluate salt properties** (e.g., thermal conductivity, specific heat, density, viscosity, vapor pressure, as a function of temperature and composition/change in composition [fission product content]), graphite and Hastelloy N properties as a function of temperature and fluence.

Early T&H analyses have to focus on **normal operating condition** temperature distributions and pressure drops, for different flow geometries and cold bypass flow fractions.

Heat losses through the vessel wall have to be calculated under normal operating conditions, with and without insulation of the vessel, assuming different cavity temperatures and vessel emissivities. Because of uncertainties in graphite properties due to the effect of neutron fluence and the presence of fission products, as well as salt properties especially due to the presence of large amounts of fission products, it is advisable to treat these analyses parametrically.

In preparation of accident analyses, **shutdown and flow coastdown characteristics** have to be assessed. In transient calculations, the effect of Doppler reactivity and reduced delayed neutron fraction need to be determined for reactivity addition accidents.

Several **accident initiators** need to be assessed, including TOP, TUC and LOHS events. To better describe the severity of different accidents, **limiting conditions** have to be identified. Examples are limiting temperatures for salt, target, IHX, piping and vessel. It seems that a temperature of 1200°C has been identified as a limiting temperature, however, at this time without a justification.

It still needs to be determined whether the **worst accident for the blanket and primary system** is a LOHS accident where the only heat losses are those through the vessel wall.

I recommend to begin with the analysis of such a transient and calculate fuel, vessel, and salt temperatures by varying:

- power densities (e.g., 5, 10, 15, 20 w/cc),
- module/vessel sizes (constraining the overall size to about 24-25 ft to ensure factory fabrication),
- central blanket region diameters (4, 5, 6 ft),
- graphite volume fractions (using conservative values for heat capacity and thermal conductivity) and
- radial reflector thicknesses (3,4 ft).

The temperature profiles have to be calculated assuming different vessel emissivities and cavity heat removal capabilities. This analysis covers also the case where no heat is removed from the cavity.

The results of this analysis will shed some light on the need for a cavity cooling system and the requirements on a fuel drainage system. It will also show the importance of the blanket graphite as a heat sink and provide support for the reference blanket power density.

Another case should also be analyzed where the blanket vessel sits inside another vessel that is filled with coolant salt which serves as a heat sink and where DRACS and/or IRACS remove the decay heat that is transferred into this vessel. This case is a variant of the earlier calculation where the cavity is not filled with air but with a very large volume of coolant salt. I had proposed this type of pool configuration earlier because I think the current plant layout where blanket and IHX/pumps are in the same vessel and the steam generators are located in the containment building, might not be acceptable because of the insufficient number of barriers to radionuclide releases.

The latter type of configuration does not employ a cavity cooling system like the MHTGR or the ALMR. To remove the heat from the outer vessel one can use either a cavity cooling system or the above mentioned DRACS or IRACS. A cavity cooling system is most likely not the best choice because of the large heat losses during normal operation and the heating requirements for the cavity in which the vessel system is placed. Using IRACS and DRACS permits the insulation of the outer vessel and reduces significantly the heat losses during normal operation.

4.0 Assignment of Priorities

Assigning priorities to analysis tasks has some level of arbitrariness that is unavoidable. I assigned the following priorities to the different analysis tasks (they are summarized in Table 1):

High Priority 1: Doppler coefficient

In my view, the Doppler coefficient issue is a feasibility issue. If a sizeable amount of fertile material has to be added to the blanket to ensure a negative Doppler coefficient, the capability of the ABC system to effectively destroy Pu can be adversely affected. The argument against LWRs and the LMR have been that those systems need fertile material (and/or large amounts of reactivity control) to operate without refueling for 12-18 months and produce large amounts of Pu. It needs to be determined how much fertile (or other) material is required to ensure a negative Doppler coefficient and how this will affect net Pu destruction.

High Priority 2: moderated vs unmoderated configuration, burnup

A well-moderated system does not require large fissile inventories. However, the graphite moderator introduces a series of problems, among them the adsorption in the graphite moderator of Xe and Kr which are neutron poisons, the need for large blanket vessels, the need for an effective sparging system to remove fission gas bubbles from the primary loop, graphite lifetime limits and their uncertainties and need for moderator replacements. The design of inlet and outlet plena, graphite support structures, mitigation of hot streaking are among the design problems introduced by the use of a graphite moderator. An unmoderated systems does not have these aforementioned deficiencies but will have much higher fissile inventories and possibly shorter salt lifetimes. It might very well be that some level of fuel salt processing is required for an unmoderated system to keep the trifluoride concentrations below acceptable limits.

Any decision to either go ahead with a moderated or unmoderated system should be based on at the least a full understanding of the neutronics implications. It also needs to be investigated how one could effectively switch from a moderated system to an unmoderated system at a later time.

High Priority 3: module size, blanket configuration, power density

Module size, blanket configuration, power density are very important design parameters that cannot be chosen arbitrarily. The normal operating and accident condition analyses cited above help in identifying the basis for these design selections. These analyses will also help in the containment selection.

Medium Priority 1: void reactivity, expansion reactivity, control rod worth

Most of these parameters help to characterize the design. The control rod worth analysis for different burnup levels will help to identify the number of control rods required to safely operate the ABC.

Medium Priority 2: shielding

The blanket vessels, IHX, pumps and other structures need to be effectively shielded to avoid radiation damage. Graphite even when spiked with boron is usually not a very good shield for fast neutrons unless used in substantial thicknesses. It does not seem that radiation shielding introduces a feasibility problem, however, the required shielding will affect design.

Low Priority 1: power flattening, delayed neutron fraction, neutron lifetime

None of these issues is critical for the design definition. However, power flattening through changes in moderator volume fraction has to be assessed early on to better estimate peak temperatures.

Low Priority 2: burnup

The burnup analyses cited above need to be performed at the beginning of the project to determine the Pu destruction capabilities; the cross section sensitivities are important but can come at a later stage. I have listed here "burnup" as an analysis issue to focus on the uncertainties that exist in the neutronics analyses.

**TABLE 1:
ABC ANALYSIS PRIORITIES
AT THE PRECONCEPTUAL DESIGN LEVEL**

High Priority	1	Doppler Coefficient
	2	Moderated vs Unmoderated System; Burnup
	3	Module Size; Blanket Configuration; Power Density
Medium Priority	1	Void Reactivity; Expansion Reactivity; Control Rod Worth
	2	Shielding
Low Priority	1	Power Flattening; Delayed Neutron Fraction; Neutron Lifetime
	2	Burnup

EVALUATION OF

ACCELERATOR DRIVEN ASSEMBLY

FOR PLUTONIUM TRANSFORMATION

(ADAPT CONCEPT)

BASED ON INFORMATION PRESENTED BY BNL AT THE

DOE NP/NE MEETING FEBRUARY 17, 1994

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Appendix A: Plutonium Coated Particle Fuel Irradiation Experience

1.0 ADAPT Concept Introduction

1.1 Introduction

BNL introduced the *Accelerator Driven Assembly for Plutonium Transformation (ADAPT)* in a series of presentations at the DOE DP & NE meeting, February 17, 1994. The ADAPT concept and the design objectives were described in general terms. No details were provided on the blanket design except for a general blanket layout, a fuel element drawing by B&W and references to fuels work done during the last few years at BNL and other contractors in support of the design of a nuclear rocket propulsion system. As observed by DOE's Bob Neuhold, many of the viewgraphs presented at this meeting could apply to any blanket concept because they were the results of generic parameter variations. In 1992, BNL had proposed a Particle Bed Reactor (PBR) for Pu burning. This reactor has many similarities to the ADAPT concept but did not require an accelerator. BNL did not present a compelling argument to DOE-NE that an accelerator was required for the proposed concept.

When questions arose about the feasibility of the concept, DOE-NP defended the ADAPT concept by claiming that the BNL ADAPT and LANL ABC concepts are both at about the same level of technology development, and that both concepts require substantial development work before Pu could be burned in either system. However, while the BNL bead fuel concept represents an interesting and significant departure from the well-known coated particle fuel concept, equating the ADAPT technology base with that for the ABC concept cannot be supported by an analysis of the respective technology bases.

I found noteworthy that BNL had claimed earlier in the presentation to the JASONs on January 18, 1994 that the ADAPT concept was selected to "*avoid . . . molten salt due to historical difficulties*", which is not supportable by the facts. However, this sentence was deleted from the viewgraph describing the "BNL Approach: Overview" that was presented at the DOE meeting.

To address the claimed equivalency between ADAPT and ABC technology and the "historical difficulties" of molten salts, I will briefly summarize the molten salt reactor technology base, and then proceed with describing the ADAPT concept.

1.2 Molten Salt Reactor Technology Base

Molten salt reactor technology development began already in the 1940's for aircraft propulsion application, culminating in the Aircraft Reactor Experiment (ARE) that used NaF, ZrF₄, and UF₄ as fuel salt, BeO as moderator, and Inconel as piping material. In 1954, the ARE was operated successfully for 9 days at a steady state outlet temperature ranging up to 1580°F (860°C) and at a power of up to 2.5 MWt. The objective for the experiment was to study the nuclear stability of a circulating fuel

system. It was found to be stable and self-regulating, and no mechanical or chemical problems were encountered.

The attractiveness of molten salt reactors for civilian power applications was recognized very early and the design of the 8 MWt Molten Salt Reactor Experiment (MSRE) began in 1960. It used a mixture of BeF_2 , LiF , ZrF_4 and UF_4 as fuel salt. Already 18 months after the design work began, fabrication of the Hastelloy-N primary system components was started. Pre-nuclear tests began in 1964. Criticality was reached on June 1, 1965, and the approach to full power was started in January 1966. The MSRE was successfully operated through March 26, 1968 which marked the end of the nuclear operation with ^{235}U fuel. The fuel salt was processed in a fluoride volatility process and ^{233}U was added to the carrier salt. On October 2, 1968, the MSRE was made critical with ^{233}U fuel. On October 8, 1968, the MSRE was brought back to power marking the world's first power operation of any reactor with ^{233}U fuel. Full power operation began on January 28, 1969.

After successfully demonstrating the technical soundness of the molten salt reactor concept for civilian power production, reactor operation was terminated on December 12, 1969. From December 1966 until the shutdown on March 26, 1968, the reactor was available for planned experimental work 87% of that period. This is a remarkable and unparalleled achievement for any demonstration plant. Table 1 shows important dates and statistics for the MSRE project.

While the MSRE operation was successful, small and not unexpected minor problems were encountered during the shakedown period and also during later operation. All of those problems were corrected. However, they still reside in the memories of a few who were peripherally involved with the MSRE and who continue to equate those minor problems with the characteristics of molten salt reactors. Following is a list of the problems that were encountered and corrected.

1. During the zero-power testing of the MSRE problems were encountered with the heat-warped radiator doors. They were replaced with an improved design.
2. Plugs developed in the off-gas system. Larger filters (particle traps) were installed to control the problem. On several other occasions, off-gas lines were plugged. Plugs were removed by heating the lines. Later it was found that there was a high in-leakage of air. This was caused by leaking service-air-line disconnects. Replacing all the air-line disconnects and installing an improved particle trap solved this problem.
3. Several shutdowns became necessary because of electrical shorts in the fuel sampler-enricher drive (twice) and the power lead to a component cooling blower. These problems were easily corrected.

4. An extended shutdown became necessary when the hub and blades in one of the main blowers in the heat removal system broke up (coolant salt - air heat exchanger). Upon inspection was learned that the cast aluminum hub in the other blower and in the spare unit had also some cracks. These blowers were leftovers from the ARE project. They were replaced by new units within 11 weeks and power operation commenced.

I find noteworthy that some people classified this incident as "an accident that forced an extensive shutdown of the reactor".

5. It was discovered that the heat removal capability of the air-cooled secondary heat exchanger was less than expected. The design analysis had been based on coolant salt specific heat data that were measured for other salts. Coolant salt specific heat measurements indicated 11% higher values.

I find noteworthy that opponents of the molten salt reactor concept have argued that "the MSRE could never reach full power".

6. Air-inleakage into the reactor cell was discovered coming from pressurized thermocouple headers and the problem was corrected.
7. A malfunction in the fuel sampler-enricher caused another shutdown. The drive cable in the sampler had become tangled and, in attempts to untangle it, the cable was severed, dropping the sample capsule and the cable latch into the pump bowl. The latch was retrieved, the drive unit replaced and the capsule was left in the pump bowl.
8. During the approach to full-power in the ^{233}U system, small sporadic increases in power for a few seconds were observed that were traced to changes in the gas volume in the fuel loop. With time, these perturbations became smaller and ultimately disappeared. The disappearance was attributed to a gradual, slight change in salt surface tension and circulating gas fraction.

None of these problems were serious. All of them had been corrected. The only recurring problems during the MSRE operation were restrictions in the off-gas lines that, however, can be designed out in a future molten salt reactor.

Note: Air ingress into a molten salt loop is often cited as an accident with potentially serious consequences. Air leakage has actually been observed on several occasions during MSRE operation without causing more than a freezing of salt which was corrected by heating the salt.

1.3 ADAPT Concept

BNL explained that efforts on accelerator-based conversion of weapons plutonium began only recently. No detailed blanket design work has yet begun but will commence when DOE funding becomes available.

At this time, the BNL ADAPT design has not progressed well beyond the brainstorming stage and a detailed evaluation is, therefore, not possible.

BNL did not provide a logical framework that led to the ADAPT concept as an approach to dispose of plutonium that is better than other concepts. Just like the PHOENIX concept (BNL had proposed this concept for the transmutation of wastes), was an adaptation of FFTF technology and fuel assembly design for a different application, the ADAPT concept is an adaptation of the Particle Bed Reactor (PBR) design for Pu disposition. While in the former approach it can be argued that FFTF technology is a well-established and successful technology that can easily be used for other purposes than research and technology advancements, such an argument cannot be made for the ADAPT concept.

The ADAPT concept is based on mainly two ideas, namely:

- the use for civilian nuclear power production of a new fuel concept, the particle bed reactor (PBR) fuel that had been under consideration over a six-year period from 1987-1993 for compact, high-power nuclear rocket propulsion.
- the use of an accelerator concept, under development by the APT project, to drive a subcritical blanket

The new fuel type and the APT accelerator concept are the major building blocks for the ADAPT concept. The objectives for the ADAPT Program are to design, build and operate a reactor that effectively destroys Pu. At this time, the basic blanket concept has not been selected. It was described by BNL only in general terms:

- ADAPT would operate in either a thermal or fast neutron configuration
- use a new fuel type capable of achieving ultra-high burnups
- the ADAPT concept could have the following operating characteristics:
  - 1000°K maximum operating temperature
  - 500 kW/liter average power density in fuel beads with a 5 mm diameter
  - 750 psi He coolant pressure
  - 3 months operating time
  - $\text{PuO}_2/\text{PuC}_2$ in solid state in all beads, with or without a SiC layer

- achieve a fractional burnup of greater than 95%

It is claimed that this reactor will be capable of burning nearly 100% of the Pu which implies that ultimately the accelerator-produced neutrons will have to carry the full burden of destroying Pu in a medium that becomes less multiplying, and in the end is a non-multiplying medium.

On two flowsheets presented at the DOE meeting, ADAPT operating conditions are shown. However, no rationale for those conditions has been presented.

For a 1070 MWe steam cycle plant (3240 MWt, implying a 33% thermal efficiency) is stated that the accelerator requires a power supply of 200 to 500 MWe. This yields a net plant efficiency of 26.9% and 17.6%, respectively, when the accelerator power needs are included.

A [direct steam cycle] *power-generating system* is proposed in which 1270°F helium coolant enters a steam generator at 925 psi to produce 1000°F steam at 2400 psi which drives a turbine to produce electricity. The steam generator feedwater inlet conditions are 3000 psi and 380°F; core helium inlet temperature is 540°F.

In a *burner cooling system*, the helium coolant leaves the core at 925 psi/1000°F. It enters a water-cooled heat exchanger and heats up the water to 400°F. This water is cooled in a forced draft dry water-to-air heat exchanger and returned to the He-water heat exchanger at 250°F; the helium coolant inlet temperature for the core is 400°F. A pressurizer/makeup tank keeps the secondary pressure at 600 psi.

BNL assigns importance to electricity production and cites as disadvantages for not generating electricity:

- machine operating costs become large,
- must dump large amounts of waste heat to environment,
- Russians will be less likely to collaborate.

BNL did not elaborate on these issues. It is well known that the proposed Pu disposition systems are very expensive both in terms of capital and operating costs. It needs to be determined what the required net thermal efficiency has to be to justify the complex and expensive heat-to-electricity conversion system. Furthermore, we know from the NPR program that electricity production was not a DOE-NP requirement for the production of tritium. The HWRF was not even capable of producing electricity. While the NP-MHTGR was capable of producing electricity economically, a design requirement was to permit 100% steam bypass. Furthermore, DOE took the position that "we are not in the electricity-selling business".

In regard to the dumping of waste heat into the environment, cited by BNL as an issue, it should be noted that low thermal efficiency systems always dump large amounts of waste heat into the environment. It will be difficult to argue that dumping more than 2000 MWt of waste heat from a 1000 MWe LWR is acceptable but 3000 MWt from an ABC/ADAPT system is not.

Note: BNL distinguishes between a power and a burner system which is a good, prudent approach. The Pu burner system operating conditions are far less demanding than those for the power system which can be expected to yield a reduction in both development cost and capital cost.

No information was provided by BNL on a collaboration with the Russians on this system. The Russians have repeatedly expressed an interest in the gas-cooled reactor (see, for example, GA contract on Pu disposition, MARR One, international MHTGR Base Technology Program) but they also made it clear that they don't have money to even participate in research and development efforts.

2.0 Concept Evaluation

The ADAPT presentations at the February 17 meeting indicated to me that the proposed concept might not be technically and economically sound. There are significant feasibility questions that I will address briefly. No reason has been cited by BNL why the ADAPT concept does not use conventional coated particle fuel for which a technology base exists. No reason was given why a steam cycle rather than a gas-turbine cycle has been selected.

It seemed to me that DOE-NP's Howard Cantor looks at destroying Pu as the last phase in a multi-stage Pu disposition process that begins with Pu storage, proceeds to the use of Pu-bearing fuel in LWRs, and concludes with the rapid (net) destruction of plutonium. If this is the path that will be taken, the third phase might commence in 20 years or so, and a gas-turbine cycle would be appropriate for the ADAPT concept..

I don't know how firm this Pu disposition scenario is. I doubt very much that it is cost-effective. Providing storage for Pu is very expensive option and not an existing capability for several hundred tons of weapon's grade Pu. Disposition of Pu in LWRs means that existing fuel fabrication facilities would have to be modified to allow for the handling of Pu or new facilities must be built. GA addressed this issue in the 1992 presentation of the NP-MHTGR for Pu burning (Reference 1):

"The radiological hazards associated with processing plutonium are much greater than those associated with processing uranium-235. DOE Order 6430.1A provides mandatory and minimally acceptable requirements for facility design. Requirements for plutonium processing and storage are covered in Division 13, Section 1304, Plutonium Processing and Handling Facilities, and Section 1305, Plutonium Storage Facilities. To meet the requirements specified in these sections, the equipment processing plutonium in solution form or as powder or bare kernels must be designed to contain the plutonium within the equipment, and the equipment must be located within ventilation enclosures, cells, cubicles, or glove boxes that will retain any spills or leaks from the equipment. Ventilation air or inert gas from these areas must be filtered by double high efficiency particulate air filters, routed to the heating, ventilation and air conditioning exhaust system, and not be allowed to mix with air in the occupied areas of the fabrication facility. The cells, ventilation enclosures, cubicles, or glove boxes must be designed for easy decontamination of the equipment and/or its enclosure. The confinement provisions will require the plant to be designed for remote operation for most of the process steps. Contact maintenance can be used for maintaining most of the equipment, but remote maintenance of some equipment may be desired or required".

Furthermore, the NRC would have to license the LWRs that burn the Pu. It would also require a legal framework as to who owns and operates these reactors and what will happen after they burned the weapon's Pu backlog. After all these expenses and delays, Pu would ultimately have to be destroyed during the third phase of the Pu disposition at a time as of yet uncertain, but pretty far in the future.

A way out of these complications would be a MOX fuel cycle but that does not seem to be in the cards of the current administration.

2.1 fuel

The bead fuel proposed for ADAPT is a very interesting new fuel type that has never been tested or even fabricated. It is based on a new particle fuel concept proposed for the Particle Bed Reactor (PBR) that was under development for the SDI/AF program. This nuclear rocket fuel was expected to meet the following requirements:

- producing 1000 MWt in a 60 cm diameter reactor
- operating the reactor (engine) in a thermal spectrum with $^7\text{LiH}/\text{Be}$ as moderator
- using 3000°K hydrogen to directly cool 400 μm fuel particles that are located in hot frits
- a power density of 40 MW/liter, which is a factor 10 higher than for the NERVA project and the highest-flux reactors in the world
- 1000 seconds operating time for the fuel
- 100 seconds lifetime for a polyethylene/aluminum matrix that forms the moderator
- fractional burnup of <0.1%

During the presentation of the ADAPT bead fuel concept, the PBR fuel technology was referred to implicitly as an established fuel technology on which the bead fuel technology is built and similarities to the AVR pebble fuel and HTGR coated particle fuel were claimed.

Firstly, PBR technology is not an established fuel technology. At this time, it is a classified technology and, therefore, is not available for general review. However, whatever information is available does not support the claim that it is a well-established technology. Only very limited fuel testing has been performed. When I asked Jim Powell for the allowable initial defect fraction for this fuel, which is a measure for the state of development of the fabrication process, I was told that no such criterion had been developed.

According to Jim Powell of BNL, a first version of a PBR fuel element had been tested with a fuel for which no specifications had been developed in regard to initial defects or releases of fission products. The only quality requirement on this fuel was that it

be manufactured on a "best effort" basis. It was expected that this fuel could operate at a maximum temperature of 3300°K and an average power density of 60 MW per liter.

The particle diameter for the PBR fuel was in the 0.5 mm range whereas the proposed particle diameter for the ADAPT fuel is in the 2.5-10 mm range. Both fuel designs had inner and outer PyC coating layers. The PBR fuel had an outer coating of ZrC; no such layer is shown for the bead fuel; an optional SiC layer coating is proposed for the ADAPT fuel. Furthermore, BNL proposes a multi-layer high retention fuel particle made in a multistep fabrication process:

- graphite spheres are soaked in uranyl nitrate
- the uranyl nitride is converted into UO_2 in a two-step process. firstly, the spheres containing uranyl salt are heated to 200°C to remove the solvent; and secondly, they are heated up to 550°C to convert the uranium into the oxide form.
- the graphite spheres are recycled five times and all previously stated steps are repeated
- in the next step, the graphite spheres containing UO_2 are heated to 3000°C to form molten UC_2 in the graphite spheres
- the UC_2 impregnated fuel (spheres) are then sent to a fluidized coater to apply various coating layers

The final form of the fuel consists of:

- UC_2 impregnated kernel
- crystalline graphite layer
- glassy carbon layer
- pyrocarbon layer
- Zr or Nb carbide layer
- pyrocarbon layer
- Zr or Nb carbide layer

No protective outer carbon layer has been proposed. The need for seal coats between the coating layers is not mentioned.

The BNL fuel kernel fabrication process is different from the inner or outer gelation process employed for the manufacture of conventional coated particle fuel kernels. Based on the data presented by BNL, fuel impregnations have been made and conversions of the nitride to oxide and later carbide forms have been achieved. No

information on the coating of the fuel balls has been provided. No consideration was given to the stability of the fluidized bed for the bead fuel that has a volume that is 50-100 times larger than that for conventional coated particles, and the evenness with which coatings can be applied.

To appreciate the difficulties that face the new fuel development, I want to highlight some of the fabrication issues for "conventional" coated particle fuel where processes are in use that were developed and refined over the last 30 years and that are still being refined.

A coated particle represents a highly sophisticated, interacting multi-layer fission product containment system. The interactions between different coating layers are the result of temperature, temperature gradients, fluence and fluence gradients, and (energy-dependent) neutron flux effects. The mechanical stability of the coating layers is further affected by the presence of fission products and their chemical interaction with some of the layers and the buildup of fission gas pressure that can lead to the cracking of one or more of the coating layers.

Over 20 years ago, the currently used TRISO coated particles evolved from the BISO particle. Parameters such as:

- kernel dimensions
- kernel and coating density,
- coating layer isotropy/anisotropy (BAF factor),
- microporosity,
- coating layer thicknesses and their variations and bonding/debonding of layers,
- neutron flux and fluence,
- temperature and temperature gradient,

affect particle design and performance in ways that are not fully understood yet for "conventional" coated particle designs. Add to this the effects on fuel quality/performance of fabrication process parameters such as:

- number and type of required coating layers,
- coating temperature,
- coating gas type and impurities,
- coating gas temperature and flow rate,
- coater size,
- coating bed stability,
- temperature gradient across the coater,
- continuity in coating process,
- quality control between coating layer applications,
- U/Pu contamination, etc.
- allowable ranges for all of the above

and this illustrates the complexities and difficulties to (1) manufacture coated particles, and (2) manufacture coated particles to meet certain specifications. The latter issue cannot be emphasized enough because a very high coating quality is required to ensure low releases during normal operation.

There is no data base for the new bead fuel. No information is available on achievable fuel quality. New quality control methods would have to be developed to determine the quality of this new fuel type. There are no models that describe fuel performance and fission product transport.

A further complication comes from the use of Pu as fuel that will require the use of remotely controlled techniques for fabrication and quality control. I noticed that the BNL fuel process viewgraph referred only to uranium fuel. The fabrication process for Pu-bearing PBR fuel has not even been demonstrated on a lab scale without regard for fuel quality. Because of the presence of Pu, difficulties can also be expected in the scrap handling and processing. These difficulties are enhanced by the use of a significantly different fuel design for the ADAPT concept. To set this issue in perspective, I have included in Appendix A an excerpt from Reference 2 that deals with the technology status for Pu-bearing coated particles and their use in a NP-MHTGR for Pu burning.

Note: Reference 2 states that "costs of approximately \$250M have to be added to cover the additional work required to develop and qualify Pu fuel for the NP-MHTGR. These costs are in addition to the costs already included in the NP-MHTGR Program to qualify HEU fuel".

GA is the only U.S. company with commercial coated particle fabrication experience and the cost estimate of \$250M is probably on the low side because it is an incremental cost to the HEU fuel qualification.

Note: It is mainly for the cost to develop and qualify Pu-bearing coated particles that I did not propose the pebble bed reactor as an option for Pu disposition when I evaluated solid fuel options for Pu disposition. However, if cost is not the deciding factor and the time to develop the Pu destroyer is not a major factor, the pebble bed reactor concept deserves serious considerations (see section 4 in this report).

In my estimation, it will take a minimum of 10 to 15 years of most intensive work to develop a certifiable ADAPT fuel fabrication process, quality control methods and ADAPT fuel beads meeting desired specifications, and understanding the underlying science and technology, at a cost of at least \$350 - 500M.

In parallel would be development work for the fuel element, with in-pile and out-of-pile testing, simulating normal operating and accident conditions. It is not clear where

those tests could be performed due to the space limitations in U.S. test reactors, and the non-prototypical test conditions.

It is not known at which time any assurance can be obtained, if at all, that this fuel is capable of extremely high burnups and high fission product retention rates. The development plan proposed by BNL is overly optimistic and unappreciative of the development issues for coated particle fuel and regulatory requirements (see section 2.6). In my view, there is the possibility that it will not be possible to develop a bead fuel that meets the ADAPT requirements.

The similarities that were claimed to exist between AVR and ADAPT fuel do not exist in any practical manner. AVR fuel is a spherical fuel with a fuel ball diameter of about 50 mm whereas the ADAPT beads have a diameter of 2.5 - 10 mm. Each AVR fuel ball contains > 10,000 TRISO coated particles with a diameter of < 1 mm. AVR fuel operates at a core power density of 2-3 w/cc whereas the ADAPT bead fuel is expected to operate at 500 w/cc which would yield an average core power density of > 100 w/cc. The core power density for an MHTGR is 5.9 w/cc.

One of the major advantages of the AVR fuel is its small temperature gradient across the fuel particle which is achieved by the large distance between the coated particles inside the fuel ball, and the very low power density. This is contrasted by the very high temperature gradients that can be expected across the ADAPT fuel because of the very high power density and the high packing density.

Over 30 years of coated particle technology have underlined time and again the importance of qualitatively and quantitatively understanding the phenomena that govern particle performance and fission product transport. For the proposed bead fuel, we don't even know what all the performance and transport issues are. We have no phase diagrams that would illuminate the chemistry of the kernel during burnup. We know, for example, that in pure carbide kernels, lanthanide and rare earth carbides are believed to be formed. Post- irradiation annealing experiments have shown that these carbides are mobile in the kernel at high temperatures. They attack the SiC layer and cause fuel particle failure. Some of the BNL viewgraphs seem to indicate that the fuel will actually be mixture of both oxide and carbide forms. In this case the formation of CO needs to be considered in addition to the fission gas buildup whenever internal pressures are assessed.

2.2 fuel element

BNL proposed to use Zircaloy as structural material for the pressure tubes. This material choice is not compatible with a 1000°K coolant temperature. In LWRs and CANDU reactors that use Zircaloy, mixed mean outlet temperatures are typically below 600°K. I doubt that Zircaloy can be used for temperatures well in excess of 600°K.

Note: Zircaloy had been proposed by Jim Powell for the cold frit in a PBR Pu burner (Reference 2). However, he had also proposed an inlet temperature of only 100°C. He proposed Inconel for the hot frit that was exposed to 700°C gas temperatures.

During the meeting at DOE-HQ, references were made to both Inconel and Incoloy (Alloy 800H?) as alternate materials, even though I suspect that presenters misspoke when they alternately referred to both materials. Incoloy Alloy 800 is a stainless steel and can withstand high temperatures but has so far been only considered for application in low-fluence environments (e.g., steam generator, IHX). Inconel is a nickel-base alloy with 8-10% molybdenum, and 20-23% chromium. It is a strong material at high temperatures, but also a very strong neutron absorber.

When the particle bed concept was first considered in the early 1980's under the SDI program, one of the concerns was the interaction of the frits with the coated particles. Because the coolant gas flows across the (loose) bead fuel bed, there will be mechanical interaction between the ball coatings and the frits, creating a dust that could clog the frits but also transport activity throughout the primary system. For nuclear rocket propulsion, this might very well be a non-issue because of the extremely short lifetime of the fuel. However, for a reactor with much longer fuel lifetimes, the dust problem could be a limiting factor. To my knowledge, there are no data on long-term dust formation in such an environment. However, it might be possible to design around the dust and clogging problem by drilling larger holes in the frits.

It should also be noted that the AVR had a high amount of graphite dust in the primary loop and dealt with it successfully. However, this dust did not contain any fission products because the AVR fuel balls are surrounded by a 5 mm carbon layer that is pure graphite except for an extremely low level of impurities, none of them uranium.

Because of the frequent references made to the use of conventional HTGR technology for the ADAPT concept a clarification is in order. The reason why the HTGR core is an all-graphite structure was the necessity for not having any metallic structures or components in the core. The melting points for Incoloy and Inconel are in the 1300 - 1400°C range. In a depressurized conduction cooldown event in a MHTGR with its huge heat capacity and extremely low power density, core temperatures can increase to 1600°C, which is well above the melting point for these metals. I would expect that the temperature in the ADAPT concept under similar circumstances, including pressurized conduction cooldowns, can increase even more.

Note: It is for these reasons that carbon-carbon composites have been under consideration for the MHTGR as containers for control material both in the center and outer reflector regions.

2.3 refueling

Periodic refueling is necessary because of the very high fuel burnup rate and the absence of a sufficiently large excess reactivity. BNL recognizes this as an important issue. How continuous or very fast batch refueling is accomplished is not clear. BNL claimed that beads from one fuel element would be drained into another element, perhaps several times back and forth, to control power and burnup distribution and reactivity.

To design such a refueling system that is safe and reliable is extremely difficult and requires a substantial engineering and testing program. In addition to the engineering of such a system, there are feasibility issues that need to be addressed. The BNL design shows a bed height of 1900 mm with bead discharge ports below the particle bed. It is quite conceivable that the weight of the bead column will not only cause bead bridging but, because of frictional forces, close off the discharge ports. When in 1985 prismatic fuel was selected over pebble bed fuel for the U.S. 350 MWt MHTGR design, the inability to withdraw pebble bed fuel from the very long core for refueling was one of the reasons why pebble bed fuel was not selected as the reference fuel.

There is also the need for a bead level indicator in each fuel element, and a counting process for the beads that are moved. There is a need for assuring that no bead bridging occurs that could allow the formation of voids below a bridge inside the elements (e.g., after a certain amount of beads in the lower section have been withdrawn during refueling), with the potential of a sudden collapse. This would also create substantial hot spots inside the element because such a void would not impede coolant flow, create a very strong cold bypass flow and leave fuel beads in other locations with less flow than required. It is not clear how this refueling process could be tested in a reactor environment except in a demonstration plant.

Note: This refueling system is different from that proposed by BNL in 1992 for the PBR Pu burner. That system used 0.5 mm diameter coated particles that were hydraulically unloaded from the bottom of each fuel element by entrainment in the helium coolant that flows into a hydraulic transfer load/unload unit. This unit is shut off during reactor operation. For this type of refueling it was essential that the small diameter, lightweight spent particles are readily entrained by and carried along with the outflowing helium. The proposed ADAPT fuel design is not suited for this type of refueling.

In one of the presentations it was claimed that the pebble reload and shuffling in the ADAPT blanket is based on the German AVR system. However, there are no similarities between the proposed bead fuel withdrawal for the ADAPT and the proven refueling method used in the AVR (and later also in the THTR). In the AVR, fuel balls are continuously withdrawn from the bottom of the core, their burnup is measured and

those pebbles with low burnup are then outside the core pneumatically lifted, in a closed loop, to an elevation above the upper layer of pebbles in the core, and just dropped down into the core. The refueling in the ADAPT system relies on lining up neighboring fuel elements so that the loading port of one assembly is aligned with the discharge port for the assembly next to it. This would be done during shutdown (depressurized?) and somehow signals would have to be given to open and close the refueling ports. Furthermore, BNL proposes to remove beads from some elements after a high burnup has been achieved leaving unanswered how the burnup status of all the beads will be determined.

2.4 fuel burnup

The claim that the ADAPT system, operating in a criticality range of 0.94 - 0.96, will achieve nearly 100% Pu burnup is impossible. Even if all fission products had suddenly disappeared, fissile material is required for (1) maintaining a $k_{eff} = 0.94 - 0.96$, and (2) compensating for parasitic absorptions and neutron leakage. Having no fissionable material left in the ADAPT blanket, i.e., 100% destruction of plutonium, means $k_{eff} \approx 0$.

To compensate for the significant reactivity swing BNL proposes to use:

- burnable poison
- movable poison
- moderator/control rods
- beam current

Burnable poison can compensate for a significant change in reactivity. However, the neutrons absorbed in the poison will not be available for the disposition of plutonium. How effective burnable poison will be in an unmoderated configuration still needs to be determined. How the burnable poison is introduced is also unknown. As excess reactivity is built into the system, the initial fissile inventory is increased.

Because of the high fission product inventory, the worth of control poisons will be greatly reduced. While the burnable poison inventory is reduced as fuel is burned, movable control is required to control the mismatch between fuel and poison burnup in terms of reactivity worth. It is not known how this can reliably be accomplished for a very high burnup blanket, in either a thermal or unmoderated configuration.

2.5 safety

One of the reasons why the MHTGR operates at 350 MWt to 450 MWt at a core average power density of 5.9 w/cc (which is less than for DRAGON and Ft. St. Vrain) is the avoidance of fuel temperatures in excess of 1600°C in case of a depressurization accident when the only heat removal path is through the vessel wall into the

cavity, and from there into the environment via a passive reactor cavity cooling system. In case coolant flow stops or is lost in an ADAPT core, massive radionuclide releases must be expected because of the sudden fuel heatup due to the absence of a significant heat sink in the blanket and the high power density. By necessity, this will introduce the need for an intermediate heat exchanger that is not shown on the BNL flowsheets. Furthermore, it raises questions about the feasibility of the concept when any flow stoppage can cause fuel overheating and increased fission product release.

The ADAPT design does not exhibit the safety characteristics that the NRC Advanced Reactor Policy Statement calls for. The ADAPT concept relies on engineered safety rather than passive and inherent safety features.

Any water ingress could have a very serious safety impact. During the presentation of the concept was mentioned that the eigenvalue could be raised by 10 points due to the addition of a moderator. This has serious implications for the selection of the eigenvalue operating range and the sizing of the accelerator and will be discussed in the next section.

The use of an ECCS was mentioned where highly borated water is injected. Considering the very high core fuel temperatures under accident conditions that can easily be in excess of 2000°K, it is not clear how water can cool the fuel. Most likely, the injection of water will cause new and very severe safety problems coming from the production of hydrogen, the exothermal reactions of graphite with water (should graphite be used as a moderator), and the burning of the graphite surrounding the fuel kernels leading to a complete fission product release.

During the presentations a flowsheet for an emergency decay heat cooling system was shown. However, this system flowsheet shows only steam dumps, an emergency condenser and an emergency feedwater tank. These are all measures to remove heat from the steam generator. They are the equivalents to the feedwater and auxiliary feedwater systems in LWRs and are expected to work well in case normal feedwater flow is lost or the turbo-generator system is unavailable. More important under accident conditions, however, are those systems that remove the heat in case of a coolant flow stoppage and/or a depressurization of the primary loop. In both instances the heat generated in the blanket is not transported anymore to the steam generator. Furthermore, in case of a steam generator tube failure, the steam generator would have to be isolated and how this is done was not shown. It should be mentioned that for the other two accident initiators just cited (coolant flow stoppage, depressurization) the steam generator would have to be isolated, too. For the MHTGR this achieved by installing the steam generator below the level of the core so that very hot gases could not enter the steam generator (investment protection goal). The price for this protection, however, is the loss of a natural circulation loop that could remove decay heat. For this reason, heat rejection through the vessel wall is necessary for

the MHTGR and this introduces limits on core size and power density.

Not addressed are accidents that could destroy the target. Among them are both the pressurized and depressurized conduction cooldown accidents. How a melting (and slumping) of the target would affect reactivity is not known. However, neutronically, the (shut-down) target is a neutron absorber where the loss of absorptions would have to be offset by an even greater loss of neutrons due to increased leakage to produce a net negative reactivity feedback (what I doubt).

On one of the viewgraphs a neutron flux of 2×10^{15} n/cm²-sec was listed which would translate into an annual fluence of 6×10^{22} n/cm². This very high flux/fluence is a direct consequence of the extremely high power density selected for the ADAPT blanket. Not mentioned in the BNL presentations were material problems under those conditions that could be substantial. It is even possible that the feasibility of this system is questionable based on material concerns unless huge shielding spaces are provided to lower vessel fluences. Again, this is not an issue for rocket propulsion application, but it is a serious concern for a reactor.

2.6 ADAPT Technology Development Plan

The development plan for the ADAPT concept leading up to Title I design work that was presented at the DOE-HQ meeting is unrealistic in terms of the very few tasks listed, schedule and the required level of funding. Not included in the Plan is the preparation of a Technology Development Plan that can easily consume a year in time and \$1-2M in funding. Also not included are the efforts to develop design and safety requirements documents which can easily absorb a few million dollars. There also does not seem to be any allowance for capital cost.

The Plan calls for the completion of a feasibility study within nine months at a cost of \$1.17M, leading to a point design. If ADAPT were to be based largely on existing technology, this task could be completed within nine months provided experienced design staff is available and only minor extrapolations from existing technology would be required. In my view, the best that can be accomplished in nine months is an identification of the key issues that need to be resolved and that affect the concept feasibility. During this time, a (highly speculative) ADAPT design can be developed that is based on the hypothesized fuel behavior and whose value is uncertain.

The cost for the development of a preconceptual design of \$28.8M seems realistic even though the time schedule of 12 months is extremely tight. It would mean that about 120 FTE would be working on this project. Missing from the task list for the preconceptual design is any work on the accelerator/blanket interface, and design work outside the conventional NSSS area. The schedule shows that in addition to the preconceptual design work, about one-third of the fuel development work is to be completed. This is totally unrealistic.

To allow for three years to develop a conceptual design, as shown on the BNL viewgraph, is realistic, provided the proper staff is assembled.

The development plan is unsatisfactory in the technology development area where the emphasis in the BNL Plan is on demonstration and not on development first and demonstration later. I will illustrate these shortcomings by reviewing "in-pile testing" as part of fuel particle development.

The BNL development plan allocates 1.25 years for in-pile testing, from beginning to end. During this time the following tasks have to be performed:

- the capsule test, including on-line monitoring and instrumentation, would have to be designed and fabricated in accordance with accepted QA and QC standards,
- the proper irradiation facility has to be selected,
- safety analyses have to be done and reports written,
- test plans have to be developed,
- preanalysis has to be performed,
- access has to be secured to hot cells after the test is completed.

These tasks can easily take 2 years. They include the review by the appropriate committees/panels, responses to comments and modifications of the design and experiment plan. During all this time, no experimental work has been done. The next tasks are:

- the irradiation in a reactor that will typically take at least several months;
- the capsule(s) has to be removed from the reactor and stored before it goes to the hot cells. This step can take a year.
- in the hot cells, disassembly will begin, metrology data will be taken, and then metallography examinations begin. This can easily take another year.
- to properly analyze the experiment and provide feedback to other experiments will take another year, including the reporting of the test and the experimental and analytical results.

This stretches the time for one experiment to at least four years which is quite different from the 1.25 years in the BNL plan. Furthermore, it is unrealistic to assume that only one capsule experiment would be required. More realistically, one should assume that a dozen or more tests need to be performed. This would stretch the fuel particle development easily over 10 years. Another year or two can easily be added for out-of-pile heatup tests on irradiated fuel.

Add to this the fabrication development efforts and the annual expenses in the early years of the project of \$30-50M can easily double. They can be kept lower when the Plan is stretched out over more years.

Similar lack of realism exists in the fuel element development that is only shown as a demonstration effort and not a development effort, with the times off easily by a factor 3-4.

I think it is possible to begin with preliminary design five years after the program started, as proposed by BNL, but an aggressive technology development program is needed to support such a schedule. A preliminary design is typically used to obtain a construction permit and it is not required that all technology development program work has been completed. However, at this stage no feasibility problems are allowed to exist anymore.

The BNL promises a technology demonstration by 1998. However, nowhere is explained what such a demonstration will encompass.

2.7 other issues

Not addressed in the BNL presentation is the temperature coefficient for the all-Pu core that I expected to be strongly positive.

Not addressed are instrumentation and control issues that could be very challenging. The refueling will require extensive instrumentation and monitoring.

The overall ADAPT technology status is in its infancy at best. A new fuel fabrication process is proposed that totally differs from established fuel kernel manufacturing processes. There is no experience base for coating in a fluidized bed consisting of the large-diameter fuel beads. It is conceivable that such a process will not work for large beads. No information was provided on achievable coating qualities.

In case a moderator is added to the system, which seems a necessity considering the large reactivity increase due to water ingress, the scope of the required development program is widened considerably.

At this time it is not known what auxiliary systems are required. It is conceivable that

extensive coolant cleanup system are required.

Extensive model development for fuel performance and fission product transport is required. Confirmation testing alone (or "best efforts" fuel fabrication) are insufficient to enter into any form of licensing review.

It is not clear to me why a gas turbine cycle has not been selected that could eliminate some of the problems (e.g., water ingress) and would also help in achieving higher thermal efficiencies.

I doubt that the ADAPT could ever be built without an IHX unless the current design is substantially modified. Among the parameters that would require changes are power density and fuel design.

Because of the many novel features for an ADAPT system a demonstration plant is a necessity. Most likely such a system would have to be preceded by tests in smaller modules.

Water ingress is expected to increase blanket reactivity by 0.1 which could render the blanket superprompt critical when it is operated at a $k = 0.95$. In a steam cycle system, the possibility for water ingress always exists and has to be accommodated by the design. This would require to lower the subcriticality to below 0.9 which would require a beam current of 250 mA or more. This is well outside of the capability of the LINAC proposed by BNL that would produce 1 GeV protons with a 100 mA current.

Large resonant cross

3.0 CONCLUSIONS

The ADAPT system is an interesting concept that, in my view, has not progressed well beyond the brainstorming stage. Details presented on the accelerator and the results of a few fuel tests and analyses cannot detract from the fact that the currently proposed system will be extremely expensive to develop and might even pose feasibility questions.

Not stated yet by BNL are the key requirements/parameters for a successful ADAPT system.

While the fuel concept proposed by BNL is a very interesting concept, for all practical purposes it is a fuel that still needs to be developed, both in terms of fabrication technology and performance requirements vs achievable performance characteristics. ZrC coated fuel has so far been produced only on a "best effort" basis, i.e., there were no requirements in terms of design and quality to achieve a certain performance characteristics. No reason whatsoever has been presented by BNL as to why this fuel has to be developed and why existing coated particle fuel is unacceptable. In my view, BNL grossly underestimates the efforts required to qualify a new fuel type. The development of a suitable bead fuel for the ADAPT concept is, in my view, a feasibility issue.

Significant development efforts are also required to qualify the fuel element design, consisting of a frit containing bead-type fuel. Such a fuel element has never been irradiated or even built.

I estimate the fuel development efforts to cost between \$350M and \$500M.

It is highly questionable if goal fuel burnups of close to 100% and a sustained bead fuel power density of 500 kW per liter can ever be achieved. HTGR coated particle fuel made in the U.S. and in Germany has failed on a large scale when irradiated in a high flux (similar to HFIR) at temperatures of 1250°C. When irradiated at lower fluxes and lower temperatures coated particle fuel performed extremely well, both in the U.S. and in Germany.

Note: The HFIR fluxes are comparable to the projected ADAPT fluxes.

The current unmoderated ADAPT design with its extremely high power density is, in my view, technically not feasible for safety reasons because in case of a flow stoppage or loss of coolant, the frits will melt and excessively high fuel temperatures can be expected that would lead to massive fission product releases.

The design feasibility is questionable also because of its response to water ingress. Water ingress is expected to increase blanket reactivity by 0.1 which could render the

blanket superprompt critical when it is operated under normal operating conditions at about $k = 0.95$. In a steam cycle system, the possibility for water ingress always exists and has to be accommodated by the design. This would require to lower the subcriticality to below 0.9 which would require a beam current of 250 mA or more which is at least a factor 2.5 above the BNL LINAC capability .

Average blanket fluxes in excess of 10^{15} n/cm²-sec are projected which, unless proper shielding is provided, would severely limit blanket tank lifetimes. Projected plant lifetimes in the range of 25 to 50 years were shown by BNL. Based on existing structural and shielding materials those lifetimes do not seem to be feasible. No data exist on frit lifetimes under those irradiation conditions. Expected lifetimes of 6 months to 2 years do not seem to be feasible under the projected flux levels.

Many of the problems of the current ADAPT concept can be designed out by relying more on the state of the art in technology. I think that a much better design can be developed using a more conventional gas-cooled reactor design rather than the design proposed here. As I alluded to earlier, the case has not been made for developing a new fuel type to meet Pu disposition goals.

The development schedule and funding that were presented up to Title I design are unrealistic in terms of the limited number of tasks, time and funding. It will take much longer and cost much more.

No reason has been presented why a pebble bed reactor would not be better for this application than the untried PBR concept. It is expected that the PBR allows much higher power densities because the frit arrangement allows for high coolant flow rates due to the short coolant travel distances (several inches) resulting in low pressure drops. Pressure drops in pebble bed reactors are notoriously high because the coolant has to travel 20-30 ft through a tightly packed pebble bed. However, in the ADAPT and PBR configuration, the lack of heat capacity will lead to very high fuel temperatures in case of a flow stoppage or loss of coolant accident. The use of metal frits further limits the potential of the PBR because of the possibility of melting under accident conditions.

re: The PBR

4.0 RECOMMENDATIONS

4.1 Pebble Bed Blanket

It seems that DOE wants at least two options for long-term Pu disposition. One option would be the molten salt blanket, and another option could be a *pebble bed blanket (PBB)*. I had not proposed this system earlier to LANL because it requires considerable fuel development work to remotely manufacture Pu-bearing coated particles and to test them. However, if DOE is willing to consider far-out concepts like the ADAPT system, a PBB is just around the corner. You might want to propose this to DOE.

The PBB would have continuous refueling and can be designed to be passively and inherently safe. The German AVR worked reliably and safely from the late 1960's to the late 1980's and even though it was mostly a test reactor, it has a lifetime availability of 70%+. For a PBB, a net thermal efficiency of 45% can be achieved in an indirect gas turbine cycle.

Clearly, a 100% burnup cannot be achieved with such a system (or any other system). However, 70-80% burnup or even higher should be achievable. The external fuel inventory can be kept much smaller than that for a molten salt reactor. The pebbles might also be in a suitable form for waste disposal without reprocessing. They can also be used in a transmuter.

It is very easy to adjust fuel enrichment in the core to account for the fission product buildup by dropping balls with a higher enrichment and/or withdrawing graphite balls from the core.

The reprocessing of coated particle fuel is very difficult. This is bad for recycle but good for proliferation resistance.

Note: The MARR One concept is based on a small (~150 MWt) pebble bed reactor using AVR technology. The Chinese are building a 10 MWt pebble bed reactor, based on AVR technology. The Russians, Chinese and Japanese know how to fabricate pebble fuel. Using the LANL accelerator, a very small demonstration plant could be built that demonstrates the essential features of the reactor (blanket) module used later.

In a PBB, the power density could be increased somewhat by employing a modified frit concept where the frits are larger and made of graphite thus greatly reducing the pressure drop. There are other possibilities to use cross flow as the heat transport path. Using ZrC-coated fuel would allow to raise maximum allowable accident temperatures from about 1600°C to possibly 1900°C.

The pebble bed reactor concept with cross flow that I described in a recent memo to

you (reference 3) does not use any frits but directs the helium inlet flow into the space between the reactor pressure vessel and the core barrel/inner reactor vessel, from where it enters the core through holes in the core barrel/inner reactor vessel. The coolant flows radially across the pebble bed and is collected in coolant channels in the central reflector region.

I think it is worthwhile to study this concept further. It relies largely on existing technology and only the Pu-bearing fuel would have to be developed which will take some time, but time did not seem to be a critical factor in DOE's third phase planning. This system would employ continuous refueling that has been demonstrated for several decades. It is based on a coated particle technology that is well developed. By using graphite frits one of the major limiting factors for pebble bed reactors could be eliminated, namely the need for low power densities to keep pressure drops low. Such a system would also allow the use of a gas turbine cycle with thermal efficiencies in the 45-50% range.

4.2 DOE-DP Briefing on MSR Technology

It might be worth exploring the possibility for providing a more detailed briefing to Howard Cantor and his staff on molten salt technology. Emphasis should be placed on what is known already and what are the gaps that need to be filled. For such a presentation it would be good to have available besides the MRS technology issues the following:

- clear statement of objectives
- crisp statement for need of accelerator/molten salt blanket
- overall design approach (design and safety requirements, integrated approach)
- rationale for technology development program (technology data needs, required experiments and facilities)
- schedule and cost for normal pace and accelerated development
- industrial partnership and collaborating labs

A MSR technology briefing had been given in January 1994 to DOE-NE by ORNL. It might not hurt to present a somewhat more optimistic outlook to DOE.

4.3 LAMPF Demonstration Experiment

I know that a demonstration experiment at LAMPF is very dear to LANL. I suggest, however, to change the emphasis in presenting the experiment from it being the centerpiece of the ABC program at LANL to being one important step on the way to an ABC molten salt system. No rationale has yet been presented for such an experiment. The rationale has to come from an overall technology development program that identifies technology data needs, how, when and at which cost to

provide the data, and what the facility requirements are.

It requires a substantial effort to identify exactly what the role of such a demonstration experiment is. Ultimately, an experiment is required that links an accelerator with a blanket. However, if it is decided to use the ATP to produce tritium such a demonstration might already be required under the ATP program leaving open the question as to why a second demonstration is needed to show that this coupling must be demonstrated again.

As I explained during the JASON dryruns, an experiment such as that proposed by LANL has merits only if and when a clear rationale for it can be presented.

TABLE 1

IMPORTANT DATES AND STATISTICS IN OPERATION OF MSRE

Salt first loaded into tanks	October 24, 1964
Salt first circulated through core	January 12, 1965
First criticality	June 1, 1965
First operation in megawatt range	January 24, 1966
Reach full power	May 23, 1966
Complete 30-day run	January 14, 1967
Complete 3-month run	April 28, 1967
Complete 6-month run	March 20, 1968
End nuclear operation with 235U	March 26, 1968
Strip U from fuel carrier salt	August 23-29, 1968
First criticality with 233U	October 2, 1968
First operation at significant power with 233U	October 8, 1968
Reach full power with 233U	January 28, 1969
Reactor Operation terminated	December 12, 1969
Hours critical	17,655
Fuel loop circulation	21,788
Equivalent full-power hours with 235U	9,005
Equivalent full-power hours with 233U	4,167

TABLE 2: SUMMARY OF LANL ABC AND BNL ADAPT TECHNOLOGY STATUS

TOPIC	LANL ABC	BNL ADAPT
Demonstration Plant	● basic feasibility of circulating molten salt fuel reactors was demonstrated in 1954 in 2.5 MWt Aircraft Reactor Experiment (ARE) ● 8 MWt Molten Salt Reactor Experiment (MSRE) was successfully operated 1964-1969 ● all components, systems, and structures performed well and compatibility with molten salts was demonstrated ● successful control and passive safety were demonstrated ● surveillance and diagnostics methods were demonstrated ● remote maintenance, repair and replacement were demonstrated	none
Fuel Technology	● molten salt fuel is the best characterized fuel ● fuel performance has been demonstrated ● no fabrication development required ● only confirmatory work for PuF₃-bearing salts is required	● fabrication technology does not exist ● fuel design does not exist ● fuel requirements have not been established ● unrealistic 100% burnup goal has been set ● fuel performance and fission product transport unknown ● fuel bead interaction in loose bed unknown
Fuel Element	● not applicable because fuel is liquid	● frits have been tested for nuclear rocket propulsion with lifetimes in the second to minute range ● frit behavior during fuel life unknown ● frit/bead interaction unknown ● fluence effects unknown

Refueling	● continuous and simple ● has been demonstrated	● not even conceptually developed ● refueling ideas presented might not be feasible ● fuel bead burnup determination on which removal is based, is unknown
Materials	● compatibility of graphite with molten salt has been demonstrated out-of-pile and in MSRE ● graphite does not have a structural function ● any graphite development aims at optimizing graphite performance to extend life and decrease porosity ● compatibility of Hastelloy N with molten salt has been demonstrated ● any structural material development work aims at optimizing structure life	● frit material not selected yet ● use of moderator is contemplated but material has not been selected yet ● should graphite be selected, the ADAPT program could benefit from MHTGR program provided it is continued ● no vessel material has been selected
Components and Systems	● pump performance has been demonstrated ● freeze valve performance has been demonstrated ● IHX performance has been demonstrated ● pump, valve and IHX sizes need to be scaled up ● fuel drainage system has been demonstrated ● off-gas system performance has been demonstrated ● steam generator needs to be developed	● ADAPT technology development could benefit from MHTGR program provided it is continued (currently, there is no funding requested for FY95) ● IHX is most likely required and needs to be developed ● He cleanup system required

Reactivity Control	● no excess reactivity control required ● MSRE reactivity was continuously monitored; control rod for reactivity control was present but never used because MSRE could be load demand controlled	● no concept has been selected that would control reactivity in a core where fuel is burned 100% ● excess reactivity control uncertain
Safety Issues	● assurance of a negative Doppler coefficient ● no other safety issues has been identified that cannot be handled through prudent design	● assurance of a negative Doppler coefficient ● reliance on engineered safety rather than passive and inherent safety ● extensive radionuclide releases can be expected during pressurized and depressurized conduction cool-down events due to fuel overheating ● during pressurized and depressurized conduction cooldown events, metallic structures in the blanket region, including the target zone, can be expected to melt ● fuel overheating due to bead bridging resulting in increased cold bypass flow ● sudden reactivity insertion due to bridge collapse

Feasibility Issues	none				● it is uncertain whether the fuel performance in terms of very high burnups and no radionuclide releases can ever be achieved ● during depressurized and pressurized conduction cooldown events it is likely that: - fuel overheats and possibly melts - metallic structures in the blanket region melt ● very high vessel fluences may severely limit vessel life ● the refueling concept might not work safely and reliably ● the ADAPT design approach does not emphasize inherent and passive safety but relies on engineered safety, which does not comply with NRC's Advanced Reactor Policy Statement

5.0 LIST OF REFERENCES

1. U.S. Department of Energy Plutonium Disposition Task Force, Draft Report Input NP-MHTGR, Rev. 0, September 10, 1992
2. Fax, J. Powell to J. Herczeg, "Description of the Particle Bed Reactor Plutonium Burner", September 14, 1992
3. Fax, W. P. Barthold to E. D. Arthur, " Plutonium Disposition Via Pebble Bed Reactor with Cross Flow", April 8, 1994

APPENDIX A: PLUTONIUM COATED PARTICLE FUEL IRRADIATION EXPERIENCE

Reference 1 provides a brief summary of the technology status on plutonium coated particle fuel that is copied below.

"Six irradiation tests have been conducted using high-enriched HTGR plutonium fuel. Five of these tests were performed as part of the Dragon HTGR Project (United Kingdom) in the late 1960s, and the sixth was performed by General Atomics in the Peach Bottom HTGR in the early 1970s. For each of these tests, the fuel consisted of kernels encapsulated by the standard HTGR pyrocarbon and SiC coating layers. The Dragon tests consisted of loose particles in graphite holders. The fuel for the Dragon tests was fabricated by Belonucleaire in Europe. For the Peach Bottom tests, standard HTGR fuel compacts were fabricated at Oak Ridge National Laboratory. For all of these tests, plutonium performed well, establishing the feasibility of high burnup, TRISO-coated, plutonium fuel for the NP-MHTGR."

"The first four Dragon tests were performed in the R2 reactor of AB Atomenergi in Studsvik, Sweden. The fuel was irradiated at peak temperatures of 1450° to 1850°C and burnups of about 280,000 to 320,000 MWd/t. Fission product release for the two capsules irradiated at 1450°C was very low. For example, fractional release for Kr-85m was about 5×10^{-6} . The two tests irradiated at 1850°C had higher fission product release, but they were intended to simulate accident temperatures which were believed to be credible for a reactor of the Dragon design (but are 400°C higher than those reached under NP-MHTGR design basis accident conditions)."

"The last Dragon plutonium test consisted of loose particles in two fuel elements in the dragon reactor. The fuel was irradiated at peak temperatures up to 1275°C and a peak burnup of 700,000 MWd/t. The fast neutron fluence was 1.5×10^{25} n/cm² (>0.18 MeV). Following the irradiation, about 500,000 out of 2,000,000 fuel particles were visually examined. No failed particles were observed. The mean fractional release of Kr-85m was less than 4×10^{-5} ."

"The Peach Bottom plutonium irradiation test consisted of 56 standard size fuel compacts in a graphite body fuel test element. The fuel was irradiated at an average temperature of 1150°C and an average burnup of 710,000 MWd/t. The fast fluence was 2.2×10^{22} n/cm² ($E > 0.18$ MeV). Two different highly enriched (88% fissile) plutonium oxide compositions were tested. Post irradiation examination of the fuel found good performance for the fuel with a lower oxygen to plutonium (O/Pu) ratio (1.68/1). Poor fuel performance was observed for the fuel with a higher O/Pu ratio (1.81/1). This poor fuel performance resulted from the well-understood "kernel

migration" phenomenon. In this case, the kernel migration was the result of the higher CO formation in the fuel particle during irradiation. Kernel migration was not observed in the fuel with the lower O/Pu ratio."