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ABC TECHNOLOGY DEVELOPMENT PROGRAM

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CHAPTER 1.0

INTRODUCTION

1.1 Purpose

The Accelerator-Based Conversion (ABC) facility will be designed to 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 [50] tons of weapon's grade plutonium to be disposed of in [20] years.

This mission is supported by four major objectives:

- provide a reliable plutonium disposition capability within the next [15] 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 ABC Technology Development Program defines the technology development activities that are required to accomplish this mission. The technology development tasks are related to the following topics:

- blanket system
- vessel systems
- reactivity control systems
- heat transport system components
- energy conversion systems
- shutdown heat transport systems components
- auxiliary systems
- technology demonstrations - large scale experiments

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Still to be added are tasks on waste management and storage, plant protection and control systems, instrumentation development, and maintenance (especially remote maintenance).

At this time, the ABC system is in the pre-conceptual design stage. The Design Data Needs² that have been identified so far are predominantly generic and based mostly on engineering judgement rather than on detailed design information and evaluation. However, even while the ABC design is still evolving, it is necessary to begin at the earliest possible time with an experimental program that supports the ABC design because even very basic design selections such as the secondary or even tertiary coolant depend on the outcome of experimental work.

The ABC design benefits from a large technology data base obtained from the Molten Salt Reactor Experiment (MSRE) and its supporting technology programs, and the design and technology development programs in support of the Molten Salt Breeder Reactor (MSBR) program³⁻¹⁵. This data base is sufficient to ensure the technical feasibility of the ABC blanket concept and move the focus of the technology development work to the assurance of technical viability.

1.2 Background

Investigations of molten salt reactors started in the late 1940's as part of the U.S. program to develop a nuclear powered airplane¹⁶. In 1950, molten fluoride salts were adopted as fuel in the Aircraft Nuclear Propulsion Program. Fluorides appeared particularly suited for this application because they have:

- a high solubility of uranium,
- are stable chemical compounds,
- have very low vapor pressures even at very high temperatures,
- have good heat transfer properties,
- are not damaged by radiation,
- do not react violently with air or water, and
- are inert to some common structural materials.

To investigate the use of molten fluoride fuels and particularly to study the nuclear stability of a circulating fuel system, the Aircraft Reactor Experiment (ARE) was operated in 1954, using a salt mixture of NaF and ZrF₄ with UF₄ as fuel, BeO as moderator, and Inconel as structural material. The 2.5 MWt experiment was successfully operated; no mechanical or chemical problems were encountered, and the reactor was found to be stable and self-regulating.

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The success of this experiment led to continued molten salt development work considering different molten salt reactor concepts for civilian applications and concluding that the most economical power can be produced in graphite-moderated converter and breeder systems operating in the thorium cycle. In 1959, a USAEC task force concluded that the molten salt reactor had the highest probability of achieving technical feasibility of all fluid reactor concepts. Because many of the features of the civilian power reactors would differ from the ARE, another reactor experiment was proposed to investigate some of the technology that would be used for power reactors. The design of the Molten Salt Reactor Experiment (MSRE) was begun in 1960, construction began in 1962, and the reactor was first made critical in 1965. This reactor used unclad graphite as moderator, and a mixture of LiF, BeF₂, ZrF₄, and UF₄ as fuel. Hastelloy N was the structural material for this 8 MWt reactor. The MSRE operated successfully until 1969 when it was permanently shut down. Earlier, the design of a molten salt breeder reactor had begun, supported by an extensive technology development program.

For other than technical reasons, the molten salt reactor program was discontinued early in 1973, but then resumed in 1974, and finally terminated at the end of FY1976.

The ARE and MSRE together with the supporting technology development programs established not only the technical feasibility of molten salt reactors but provided also a high level of assurance for their economical viability. The behavior of molten salt fuel is probably better known than that of any other fuel type. The molten salt reactor's passive and inherent safety characteristics are equal or better than that of any of the advanced reactors.

In the late 1970s, molten salt reactors, this time in form of denatured molten salt reactors, were again considered, this time under the Nonproliferation Alternative System Assessment Program (NASAP). However, work did not extend beyond the completion of the final NASAP report.

It should be noted that molten salt work did continue in the U.S., albeit at a much lower level of effort, for non-nuclear applications (e.g., waste management). Overseas, especially in the former Soviet Union, molten salt technology work for both nuclear and non-nuclear applications continues until today.

In 1993, LANL decided to use a molten fluoride salt as the basis for a series of accelerator-driven transmutation technologies (ADTT), among them the accelerator-based conversion (ABC) system for the disposition of plutonium. Unlike its predecessors, namely the molten salt breeder or converter reactor designs, the ABC system is subcritical and does not rely on frequent reprocessing of the fuel salt to

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achieve a high breeding gain. Its aim is not the production of fuel but its complete destruction. The high level of fission product accommodation in the fuel salt is utilized to burn over 90% of the fissile plutonium charged to the ABC system without having to reprocess the fuel. Gaseous fission products such as Xe and Kr which are insoluble in the molten salt are continuously removed through an off-gas system. The noble metal fission products can be removed continuously, for example, via electrolytic deposition. If desired, seminoble metals can also be removed through the same process.

The novelty in this concept is the merging of two established technologies, namely molten salt reactor technology and accelerator technology. This ABC technology development program will address only the molten salt technology issues that need to be resolved. There are no basic technology feasibility issues that need to be addressed¹⁷. Whatever feasibility issues exist relate to the particular ABC system design and its capability to achieve the mission objectives.

While there is a broad molten salt reactor technology base, existing data bases are often insufficient to fully support the ABC system design and many of the phenomena observed in the past during MSRE operation and in subsequent loop experiment need to be fully understood.

1.3 Approach

This ABC Technology Development Program is based on design data needs² that (1) were identified in the design of the ABC system but (2) are based largely on engineering judgement and the experience described in numerous reports on molten salt technology development needs and assessments of the technology development status. Design Data Needs are translated into technology data needs and they in turn are translated into near-term, intermediate-term and long-term technology development needs.

Near-term technology development needs identify high-priority issues that need to be addressed at the earliest possible time because of their high impact on design and the assurance of ABC design feasibility and technical viability of the chosen design. They cover also issues that need to be resolved for the early experiments such as the

¹⁷ The Molten Salt Group which was headed by Ebasco Services, Inc. and included five large industrial firms and 15 utilities, concluded in 1971 that the existing molten salt reactor technology was sufficient to justify the construction of a molten salt reactor demonstration plant.

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proof-of-principle experiment. Typical for this near-term technology development category are small-scale experiments.

Intermediate-term technology development needs are those that fall between the long-term and near-term needs. Results from those experiments are not needed in the early phase of the design. However, there are instances where work on those needs has to commence early because of long lead times for equipment and/or the long duration of the experiments (e.g., long-term corrosion tests). Typically, such experimental data are obtained from large-scale experiments.

Long-term technology development needs are those where experimental results are required at a later stage. These data needs will complement existing data bases. Very often, producing those data is routine work using well-established techniques in existing facilities. As in the previous category, work on long-term technology development needs might have to begin early in the project because of long lead times for the procurement of equipment (e.g., steam generator tests) and the long duration of the tests. Typically, such experimental data are obtained from large-scale experiments.

Of particular importance for the ABC Technology Development Program are the proof of principle experiment and the technology demonstration experiment.

In a proof-of-principle experiment the accelerator (e.g., the LAMPF facility at LANL) will be coupled with a molten salt blanket with its associated heating, cooling and control systems, purification systems and a gas sparging system. Such an experiment will start without any fissile material in the blanket. It will allow for target testing in a molten salt environment. System thermal-hydraulics can be studied under both forced and natural convection flow conditions. Operation of the system will also permit to gain experience in the areas of maintenance, repair and replacements. At a later stage, fissile material can be added in small quantities to achieve a certain level of subcriticality. At that time, a series of reactor physics and thermal-hydraulics experiments can be carried out that will provide methods validation data.

The proof-of-principle experiments will be completed by tying this now subcritical facility at a specific eigenvalue to the accelerator to produce nuclear heat and demonstrate that stable, reliable and continuous heat production can be achieved.

The demonstration experiment goes well beyond the proof-of-principle experiment in that its purpose is the demonstration of the capability of the ABC to safely burn plutonium and produce electricity. It might be possible to base this experiment on the proof-of-principle experiment by using all or part of its equipment. At this stage,

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components such as pumps, valves, IHX, steam generator, processing facilities, etc. do not have to be of prototypical size but their design features should closely resemble those of the prototype components.

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References:

1. "Accelerator-Based Conversion (ABC), PU Disposition, Requirement Document", prepared by Barthold & Associates, Inc., August 26, 1994
2. "ABC Design Data Needs", prepared by Barthold & Associates, Inc., September 29, 1994
3. "Program Plan for Development of Molten Salt Breeder Reactor", ORNL-5018 (December 1974)
4. "Conceptual Design Characteristics of a Denatured Molten-Salt Reactor with Once-Through Fueling", ORNL-TM-7207 (July 1980)
5. "Development Status and Potential Program for Development of Proliferation-Resistant Molten-Salt Reactors", ORNL-TM-6415 (March 1979)
6. "Distribution and Behavior of Tritium in the Coolant Salt Technology Facility", ORNL-TM-5759 (April 1977)
7. "Evaluation of Alternate Secondary (and Tertiary) Coolants for the Molten Salt Breeder Reactor", ORNL-TM-5325 (April 1976)
8. "Heat Transfer Salt for High Temperature Steam Generation", ORNL-TM-3777 (December 1972)
9. "Assessment of Molten Salts as Intermediate Coolants for LMFBRs", ORNL-TM-2696 (September 1969)
10. "Conceptual Design Study for a Single-Fluid Molten Salt Breeder Reactor", ORNL-4541 (June 1971)
11. LANL/ORNL viewgraph presentation to JASONS, January 18-19, 1994
12. ADTT Presentations at International ADTT Conference in Las Vegas, July 1994
13. LANL Viewgraph Presentations dated 2/17/94, 5/16-18, 94, and 8/24/94
14. M. W. Rosenthal et al., "Recent Progress in Molten Salt Reactor Development", 1971

ABC TECHNOLOGY DEVELOPMENT PROGRAM

15. "The Status and Technology of Molten Salt Reactors: A review of Molten Salt Reactor Work at the ORNL", the draft of eight papers distributed at the Molten Salt Reactor Program Information Meeting on May 27-28, 1969
16. H. G. MacPherson, "The Molten Salt Reactor Adventure", Nucl. Sci. Eng. 90, 374-380 (1985)

CHAPTER 2.0

CURRENT STATE OF TECHNOLOGY

In the following the state of molten salt technology in the U.S. will be summarized with focus on those topics that are important for the ABC system. Today, molten salt technology for nuclear energy applications is extensively pursued in Russia, and to a lesser extent in Germany, Sweden, China, Japan, France and Italy. The foreign technology base has not been assessed for the development of this technology plan.

2.1. BLANKET SYSTEM

The most important event in the history of molten salt reactors was the successful operation of the 8 MWt Molten Salt Reactor Experiment (MSRE). The operation of this reactor showed that the molten salt reactor is a very practical reactor. It ran for long periods of time, and when maintenance was required, it was accomplished safely and without excessive delays. The MSRE demonstrated the expected flexibility and ease of handling the fuel. The MSRE was the first reactor in the world to run solely on ^{233}U . This highly radioactive fuel would have been very difficult to handle as a solid fuel but posed no problems for the liquid fuel. While the MSRE had control rods, they were never used because the reactor was self-regulating.

Only three problems requiring further development work turned up during and after the operation of the MSRE:

1. Hastelloy N embrittlement due to accumulation of helium at the grain boundaries
2. Hastelloy N embrittlement due to tellurium attack
3. the tritium produced by neutron reactions with lithium penetrated metal walls easily and in a commercial reactor would be expected to appear in steam generators and reach the atmosphere

It was later determined that

- modified alloys that contained small amounts of niobium could reduce significantly the Hastelloy N damage;

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- tellurium attack could also be reduced by controlling the redox potential of the fuel;
- coolant development work showed that a coolant consisting of sodium fluoride and sodium fluoroborate would effectively trap the tritium and that tritium could then be removed and isolated in a gas purge system.

While these remedies showed, in principle, how the observed and anticipated problems could be corrected, the existing data base is rather too sparse to establish a firm technology base.

The fuel salt in the MSRE consisted of $\text{LiF-BeF}_2\text{-ZrF}_4\text{-UF}_4$. The breeder reactor (MSBR) that was the focus for the molten salt development work during the 1960s and 70s, used as fuel a mixture of $\text{LiF-BeF}_2\text{-UF}_4\text{-ThF}_4$. An extensive data base exists for both fuels. The fuel proposed for the ABC system is a mixture of $\text{LiF-BeF}_2\text{-ZrF}_4\text{-PuF}_3$ for which no directly applicable data exists. In the following a brief summary of the state of the art in molten salt technology is presented as it relates to the ABC system.

2.1.1 Molten Salt Development and Characteristics

2.1.1.1 fuel chemistry

The MSRE fuel $\text{LiF-BeF}_2\text{-ZrF}_4\text{-UF}_4$ has performed well throughout the life of the MSRE. However, the neutron flux level and duration of fuel exposure in the MSRE were well below the expected ABC conditions. Furthermore, the PuF_3 present in the MSRE came mainly from the Pu formed in the neutron capture reactions in ^{238}U (about 500 grams) that stayed in the salt. During the last few months of MSRE operation, a total of 183 grams of plutonium instead of ^{233}U was added in form of PuF_3 powder. The reactivity effects of the plutonium began to be seen about 10 minutes after immersion of the fuel-containing capsule in the fuel salt, and the dissolution of all PuF_3 was complete in about 3 hours.

At one time, there was a concern that a contamination of the salt with moisture or oxygen could lead to a uranium oxide precipitation, and the reduction of UF_4 into UF_3 would lead to UF_3 disproportionate into UF_4 and insoluble uranium. However, neither condition was approached in the MSRE. The presence of ZrF_4 in the salt assured that the first oxide to form in the salt would be ZrO_2 that would precipitate. In regard to the reduction of uranium it was found that the valence of the fission products was slightly less than four thus assuring that any excess fluorine would lead to the gradual conversion of UF_3 to UF_4 , and not to the precipitation of uranium. Furthermore,

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maintaining a small fraction of UF_3 by occasionally exposing a rod of beryllium metal to the salt (which led to the formation of BeF_2 and reduction of UF_4 to UF_3) also guaranteed a reducing, non-corrosive environment.

Only sparse data are available on the $\text{Pu}^{3+}/\text{Pu}^{4+}$ equilibrium that is of importance for the ABC system. No direct experimental data are available for the solubility of actinide oxides in the fuel salt. Only very small concentrations of higher actinides were present in the MSRE salt (the fuel was 33% enriched in ^{235}U).

In regard to the corrosion products and their interaction with the fuel, the following was observed in the MSRE. After the fuel processing in September 1968, some unreduced corrosion product fluorides from the fluorination process were apparently left in the salt. The impact of their presence was felt when beryllium was added to the salt to produce the desired UF_3 level. It was observed that the first three additions of beryllium were consumed to reduce these fluorides to metals before UF_3 was built up. Except for this interference with the UF_3 - UF_4 balance just mentioned, corrosion products never amounted to more than 200 ppm with no significant effect on fuel salt chemistry.

A significantly higher buildup of corrosion products can be expected for the ABC design because a salt cleanup is scheduled only after about 10 years of operation. How a high concentration of corrosion products will affect the $\text{Pu}^{3+}/\text{Pu}^{4+}$ balance is unknown.

The MSRE experience showed that the noble gases Kr and Xe are virtually insoluble in the fuel and were easily removed. Fission products with stable fluorides (Rb, Cs, Sr, Ba, the lanthanides and Zr) appeared almost entirely in the fuel as fluorides. More noble fission products (Nb, Mo, Ru, and Te) are virtually absent from the fuel and were deposited on graphite and Hastelloy N. Analyses of MSRE salt samples for five noble-metal nuclides showed that the fuel salt contained only a few tens of percent of the nominally calculated inventory.

No problems were encountered in regard to the solubility of rare earth fluorides. It was known that their solubility limit in the fuel was only slightly in excess of 1 mole %, however, their concentration in the MSRE fuel salt was far below this value.

Because of the differences in fuel salt composition for the MSRE and the ABC system and because of the very high concentration of fission products present in the ABC fuel salt, and the changed redox potential of the fuel, the existing data base on the solubility of the rare earth elements, noble metals as well as the other fission products is of uncertain value and applicability, and new data are required.

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The MSRE experience illustrated the importance of knowing the physical properties of the fuel salt in use, and not relying on extrapolating properties from similar salts. For the MSRE, the heat removal capability of the air-cooled secondary heat exchanger was substantially less than expected because the coolant specific heat used for the design had been extrapolated from data for other salts, but was experimentally shown to be 11% higher than the original value. For the ABC design, the physical property data available today are sufficient for conceptual design but are insufficient for preliminary and final design. No data are available that show the change in physical properties of the fuel salt due to fuel salt composition changes which are the result of fuel burnup and fission product buildup.

The MSRE operating experience did not show an attack on the graphite by the salt. Small quantities of fuel salt had penetrated the graphite either through pores or cracks. No information is available on fuel and fission product interaction with graphite at higher fluence levels.

Tellurium was present in the MSRE fuel salt as a fission product. From post-irradiation work on Hastelloy N it was concluded that tellurium is probably responsible for the intergranular attack observed in Hastelloy N. However, little is known about the behavior of tellurium in molten salt environments, especially about the effect of the redox potential of the fuel on the relative concentration of the various tellurium species.

2.1.1.2 fuel-coolant interaction

A eutectic mixture of NaBF_4 and NaF is proposed for the ABC system 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. In case fluoroborate coolant mixes with fuel salt due to leaks in the primary heat exchangers, the reaction of safety significance is the decomposition of NaBF_4 :

which leads to an increase in BF_3 pressure and a partial immiscibility of the resulting phases. A similar reaction occurs when leaks occur in the secondary system (e.g., in the steam generator). Dynamic mixing experiments to investigate non-equilibrium conditions have not been conducted.

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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. How a steam generator tube rupture would affect coolant chemistry is unknown, and how an oxide-laden coolant ingress into the fuel salt would affect the fuel chemistry is unknown, too.

2.1.1.3 coolant chemistry

The MSRE used a mixture of 66 mol% ${}^7\text{LiF}$ - 34 mol% BeF_2 as secondary coolant. However, this salt has a high liquidus temperature of 856°F (458°C) which is not compatible with desired steam generator operating conditions. Furthermore, the cost for this coolant is high because of the required Li enrichment.

For the MSBR, a mixture of 8 mol% NaF - 92 mol% NaBF_4 had been proposed. This coolant has a lower melting point of 725°F (385°C) and is inexpensive. Preliminary investigations showed that this salt seems to be sufficiently stable in a radiation field and compatible with Hastelloy N.

Because of the abundant lithium in the ABC system, the tritium production rate is very high. In a system with LiF-BeF_2 coolant, this tritium is expected to migrate freely from the fuel into the coolant and from there into the steam system. One of the expected advantages of fluoroborates is the effective trapping of tritium thus stopping it from reaching the steam system. However, while effective trapping of tritium has been demonstrated, the trapping mechanism and its limitations are not fully understood.

A series of tritium experiments were conducted in ORNL's Coolant Salt Technology Facility (CSTF) to examine the potential of sodium fluoroborates for limiting tritium transport to the steam system. The results from these experiments demonstrated the effectiveness of sodium fluoroborate for trapping tritium. More than 90% of tritium added to the CSTF at steady-state conditions appeared in the off-gas system in a chemically combined (water-soluble) form. When chemical additions were made to the salt, the trapping efficiency of the salt appeared to be further enhanced.

The vapor pressure of BF_3 due to the thermal decomposition of the coolant at high temperatures was found to be less than 1 atm at the highest operating temperature for compositions containing less than 93 mol% NaBF_4 . It is not known whether the coolant chemistry will change when BF_3 is removed from the coolant system.

It is known from small loop tests that even a small addition of water to the fluoroborate coolant will greatly increase corrosion. It is, therefore, necessary to be

able to purify the salt to remove reaction products and other contaminants. This capability has been demonstrated on a lab-scale batch basis but not as an on-line purification process.

Some physical properties such as viscosity, density and heat capacity are well known for fluoroborates. However, data such as thermal conductivity, surface properties (surface tension, interfacial tension between salt and graphite), vapor pressure and vapor composition are not well known.

2.1.1.4 fission product chemistry

Table 2.1 shows the distribution of fission products found in the MSRE. The noble gases have very low solubilities in the fuel salt and in case of ^{135}Xe , over 80% was removed through helium sparging. More efficient sparging methods had been proposed for the MSBR but they have not been tested yet.

The rare earths and other stable fluorides (e.g., Zr, Ce, Nd, Sr, Cs, Y, Ba, Rb) are expected to stay in the salt.

Most of the iodine stays in the salt and only small or negligible amounts are present on metals, graphite and in the off-gas system. With the focus in the past being on a molten salt breeder reactor there was considerable attention given to the removal of iodine ^{135}I which decays with a half life of 6.7 hours into ^{135}Xe . This isotope is a significant neutron absorber with a half life of 9.1 hours.

Noble metals do generally not react with fluorine and are present in metallic form. They were found mainly in the salt, on metals and on graphite. Their distribution varied widely and could not satisfactorily be explained. Their chemical state is also not well known.

At one time it was thought that a substantial amount of noble metals was present in the gas phase in the pump bowl. However, improved sampling techniques developed during the MSRE operation with $^{233}\text{UF}_4$ showed that previous samples had been contaminated by salt mist and that only a small fraction of the noble metal fission products escaped to the cover gas.

It was observed during MSRE operation that niobium which among the noble metals is the most susceptible to oxidation, could be made to disappear by lowering the redox potential of the fuel, but it subsequently reappeared for reasons that were not always explainable.

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TABLE 1.1.3.1: DISTRIBUTION OF FISSION PRODUCTS FOUND IN THE MSRE*

Fission Product Group	Example Isotopes	Distribution (%)				
		in salt	to metal	to graphite	to off-gas	other
stable salt seekers	^{98}Zr , ^{144}Ce , ^{147}Nd	~99	negligible	< 1 (fission recoils)	negligible	processing
stable salt seekers (noble gas precursors)	^{89}Sr , ^{137}Cs , ^{140}Ba , ^{91}Y	variable	negligible	low	variable	
noble gases	^{89}Kr , ^{91}Kr , ^{135}Xe , ^{137}Xe	low	negligible	low	high	
noble metals	^{95}Nb , ^{99}Mo , ^{106}Ru , ^{111}Ag , ^{129}Te	1-20	5-30	5-30	negligible	processing
iodine	^{131}I , ^{135}I	50-75	< 1	< 1	negligible	processing

* This table was taken from ORNL-5018, "Program Plan for Development of Molten Salt Breeder Reactor", December 1974, p. 5-15

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No information is available on how the noble metals are deposited on graphite or metallic surfaces, i.e., if they are adsorbed or deposited in pores or if they diffuse into surfaces. No information is available on how noble metals are resuspended or lifted off,

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2.1.1.5 tritium behavior

There are four processes that lead to the production of tritium:

- ternary fission
- ${}^6\text{Li}(n,\alpha){}^3\text{H}$ (thermal neutron absorption)
- ${}^7\text{Li}(n,\alpha n){}^3\text{H}$ (fast neutron reaction)
- ${}^{19}\text{F}(n,{}^{17}\text{O}){}^3\text{H}$

The two lithium reactions produce in roughly equal proportions over 98% of the total tritium.

It appeared from MSRE measurements that about 60-70% of the calculated tritium production found its way through the fuel off-gas system. About 12-15% of the tritium diffused through the heat exchanger tubes, and about nine-tenth of this diffused through the tubes into the cooling air. The disposal of tritium was never problematic for the MSRE. It was calculated that operating at 7.25 MW, with ${}^{233}\text{U}$ fuel, the MSRE would produce about 40 Ci/day.

For the 2250 MW MSBR it was estimated that 2420 Ci/day would be produced of which 1670 Ci/day would reach the steam system. It was concluded that this high a tritium concentration would require precautions such as tritium monitoring, good ventilation of work areas, restrictions on handling discharged water and possibly the use of masks when working on steam leaks.

A comparison of measured MSRE tritium concentrations at various locations within the MSRE system, with calculated data showed a very large discrepancy in the prediction of the tritium that escapes into the cell atmosphere. Furthermore, there were indications of a tritium holdup in graphite and also metals, and the formation of hydrocarbons.

Work done after the MSRE shutdown showed that the use of NaF-NaBF₄ as intermediate coolant could trap a very high percentage of the tritium produced in the blanket. If this capability can be confirmed it could greatly mitigate if not eliminate the tritium control issue, provided the fluoroborate coolant is otherwise acceptable as a secondary coolant. Hitec, a heat transfer salt (Na-K, nitride-nitrate mixture) also traps tritium effectively and has a very low melting point (288°F, 142°C), and could be used in a tertiary loop (because of its violent reaction with graphite, it is not suited as a secondary coolant when graphite is used in the blanket region).

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2.1.2 Graphite Development and Characteristics

The compatibility of molten salts with graphite has been extensively demonstrated, especially in the MSRE where after 5 years of operation the graphite retained preinstallation machining marks and documented surface imperfections.

Under irradiation, graphite undergoes dimensional changes. Unlike in metals, radiation damage in graphite consists entirely of displaced atoms due to neutron/atom or atom/atom collisions. When exposed to irradiation, graphite densities rise initially resulting in a volume decrease. For different graphites, dimensional changes of -1% to -3% have been observed. These changes are not isotropic. For example, for N3M graphite at 600°C, the maximum dimensional change in radial direction is about -3% at 20 dpa whereas the change in axial direction is only about -2%. During the shrinkage phase, the bulk graphite starts to fill existing internal voids. The concerns during this phase are the dimensional changes that occur which are closely related to the neutron flux and temperature profiles, i.e., the dimensional changes are uneven.

This period of graphite shrinkage is followed by graphite swelling that results in a volume increase. The concerns during this second phase are the dimensional changes that occur, the rapid change in material properties, and the formation of cracks into which the salt can penetrate. Common to all graphites is the rapid increase in swelling after (approximately) returning to its original dimensions. The common design rule for graphite structures in a neutron field is that the end of lifetime is reached whenever the graphite has returned to its original dimensions after it went initially through a shrinking phase followed by a swelling phase. However, it is conceivable that excessive shrinkage alone could impose constraints on the design.

Salt penetration into the graphite is prevented by surface tension if the entrance pore diameters to the internal void structure are $1\mu\text{m}$ or less. It has been estimated that should fuel salt seepage into the graphite occur, local hot spots could easily attain temperatures of 1100 -1200°C. At those temperature, the graphite damage rate is greatly increased over that at 700°C. For fuel salt seepage to occur the entrance pore diameters would have to be much larger than $1\mu\text{m}$, or cracks would have to occur, or significant pressure surges would have to happen.

For the MSBR, a life-limiting fluence of $3 \times 10^{22} \text{ n/cm}^2$ had been adopted. At this fluence cracks would begin to form in the graphite that would significantly lessen the resistance to xenon diffusion into the graphite. In later molten salt reactor studies on nonproliferating systems where xenon permeability was no a major concern, the life-limiting fluence was established by the formation of cracks that permitted salt intrusion. It was concluded that with improved graphites, the life-limiting fluence

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could be as high as 4×10^{22} n/cm².

To address the dimensional stability of graphites, so-called binderless graphites such as POCO AXF were found to have improved dimensional stability over most of the conventional graphites. Pyrolytic sealing of graphite had been investigated and found to be not very effective. During irradiation of sealed graphite, the injected pyrocarbon reduced the void spaces and swelling began at lower fluences than those in the absence of a coating.

Today, four different graphites seem to be of potential interest to an ABC system, namely type N3M from ORNL, type IG-110 from Japan's Toyo Tanso, type AFX-8Q from POCO, and type 2020 from Stackpole. MHTGR graphite candidates such as H451 need to be considered, too.

Note: During the NP-MHTGR program's graphite development program it was learned that currently, there are no more U.S. suppliers of nuclear grade graphite. It was concluded that graphite needs would have to be met by foreign suppliers. This is an important consideration whenever a particular graphite type has to be qualified in the U.S. for nuclear service.

2.1.3 Metals Development and Characteristics

Hastelloy N was the sole structural material used in constructing the MSRE. This alloy was specifically developed for use in molten salt systems. It contains 16% molybdenum for strengthening, 7% chromium for oxidation resistance in air, as well as 5% iron, 1% manganese, and 1% silicon.

Two problems were noted with Hastelloy N, none of them compromised the MSRE operation, that needed to be addressed before more advanced molten salt reactors could be built.

First, it was noted that Hastelloy N was embrittled by helium produced from tramp ¹⁰B:

and from nickel in a two-step reaction:

The boron reaction saturates and produces about 1-10 at. ppm of helium. The reaction with nickel, however, does not saturate. It was estimated that during 30 years of MSBR operation, this reaction would produce about 40 at. ppm of helium.

The second problem was caused by the fission product tellurium that diffused a short distance into the metal along grain boundaries and caused embrittlement.

Both problems were addressed in the years following the shutdown of the MSRE.

The embrittlement by helium had been addressed first. It was found that the carbide precipitate that normally occurs in Hastelloy N could be modified to obtain resistance to helium embrittlement. It was known that the presence of 16% molybdenum and 1% silicon led to the formation of a coarse carbide that was of little benefit for reducing helium embrittlement. However, reducing the molybdenum concentration to 12% and the silicon content to 0.1% together with the addition of a reactive carbide former like titanium led to the formation of a fine carbide precipitate and an alloy with good resistance to helium embrittlement. This led to a modified Hastelloy N that contained 2% titanium.

The understanding of the intergranular embrittlement of Hastelloy N by tellurium was developed later. Hastelloy N probes taken from the MSRE showed an abundance of tellurium but no other fission products. To simulate this tellurium diffusion in the laboratory was initially unsuccessful. Electrochemical probes were immersed directly in salts that contained tellurium, but there was never any evidence of a soluble tellurium species. It was hypothesized that tellurium moved through salt from one point to another as a low-pressure, pure-metal vapor and not as a reacted species. An experimental system was developed for exposing metal specimens to tellurium. The metal specimens were suspended in a stirred vessel of salt with granules of Cr_3Te_4 and Cr_5Te_6 lying at the bottom of the salt. Tellurium at a very low partial pressure was in equilibrium with these two compounds and it was found that exposure of Hastelloy N specimens to this mixture resulted in crack severities similar to those noted in MSRE samples.

The most important finding was that a modified Hastelloy N with 1-2% Nb had good resistance to tellurium embrittlement. However, it was also found that the presence of titanium (that was added to overcome the helium embrittlement) negated the beneficial effects of niobium. Even though the mechanical properties of such an alloy (containing 2% Ti and 2%Nb) in the irradiated condition were found to be excellent, the tellurium embrittlement problem was not resolved.

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The next step in the development work was the elimination of titanium from the modified Hastelloy N. One irradiation experiment was run that showed promising results. An alloy containing 2% Nb was irradiated at 704°C and found to be 30% stronger than standard Hastelloy N and had a fracture strain of 3% compared with <1% for standard Hastelloy N. Even though alloys with Nb have generally not as good post-irradiation properties as those with Ti, it needs to be determined whether these properties for the Nb-containing Hastelloy N are adequate.

So far, niobium-modified Hastelloy N has only been made in melts of less than 50 lb and a limited number of test welds have been made in the 1/2 inch-thick plates that passed bend and tensile tests required by the ASME Boiler and Pressure Vessel Code.

In a different approach to dealing with the tellurium problem, the fuel salt melt was made reducing in the hope that this would tie up the tellurium in some metal complex. It was found that the U^{4+}/U^{3+} ratio had a significant effect on cracking suggesting that the reactor could possibly be operated in a chemical regime where tellurium would not be embrittling even to standard Hastelloy N.

No information is available on the performance of Hastelloy N or modified Hastelloy N in the fuel melt used for the ABC that does not contain uranium but plutonium, where the fission product concentrations are much higher than in the MSBR and where higher fluences can be expected than those projected for the MSBR.

Unknown is also the effect of the harder neutron spectrum on Hastelloy N. It needs to be determined whether any irradiation-enhanced precipitation causes embrittlement.

MSBR studies included also an assessment of iron-based alloys as alternate containment/vessel materials. The advantages of these alloys are (1) the higher resistance to Te embrittlement and (2) the lower (n, α) cross section and, therefore, lower radiation-induced helium embrittlement. The disadvantages for the use of these alloys are the temperature limitation to <650°C and the unknown effect of the Pu^{4+}/Pu^{3+} redox potential.

Stainless steel performed well in thermal convection loops as shown in Table 2.2. However, more data (out-of-pile, in-pile) are required to assure the suitability of stainless steel in the ABC system.

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Table 2.2: Operating Conditions for Stainless Steel Thermal Convection Loops¹

Loop Material	Salt Composition (mole %)	Maximum Temperature, (C)	ΔT (C)	Time (hrs)
304L SS	LiF-BeF ₂ -ZrF ₄ -UF ₄ (70-23-5-1-1)	688	100	83,520
316 SS	LiF-BeF ₂ -ThF ₄ -UF ₄ (68-20-11.7-0.3)	650	110	4,490
316 SS	LiF-BeF ₂ (66-34)	650	125	25,100

¹ Data are taken from ORNL viewgraphs for JASON meeting presentation, January 18, 1994

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2.1.4 Monitoring and Surveillance in Molten Salt Systems

Prior to the MSRE operation, extensive experience in the handling and analysis of non-radioactive fluorides was gained under the ANP Program. Ionic or instrumental methods had been developed for most metallic constituents, and methods were available for F⁻ and sulfur.

For MSRE application, it was necessary to adopt these methods to hot-cell operation. The major development of analytical methods for discrete sampling was done under the MSRE program. However, with the exception of in-line analyses of the off-gas and remote gamma spectroscopy, all analyses were performed on batch samples off-line, either in hot cells or bench-top methods.

During the operation of the MSRE and in the subsequent technology development program, the development of discrete sampling techniques was continued and their limitations were identified.

In regard to specific analytical techniques, experience had shown that certain electroanalytical methods, voltammetry and perhaps potentiometry, offer the most convenient transducers for in-line installation.

Spectrophotometric methods were found to be useful for the analysis of fuel, coolant and processing streams to aid in the evaluation of the chemistry of such streams. However, they were used only on a batch-wise basis with only exploratory consideration of in-line use.

Chemical methods have been used in the analysis of samples. However, they have not been extended to in-line analysis. Such an analysis requires the addition of a reagent and the subsequent monitoring of reaction products. It was felt that the applicability of such methods for in-line use will be limited.

Analysis of gas streams was carried out continuously for the MSRE. Commercially available instrumentation was adopted for high-temperature molten salt applications.

During the latter phases of the MSRE operation, gamma scans were extensively used. A variety of fission products were located and identified on surfaces of MSRE components. Much valuable quantitative and qualitative information was obtained on the deposition of fission products on major component and on the distribution of fission gases in the system. Problems with this technique were encountered by the multitude of overlapping photopeaks of short-lived isotopes for which adequate energy characterizations were unavailable and the inability to resolve, using a

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computer, the large number of higher order multiplets that were found in the spectra.

Primary accomplishments in the period from 1974 to 1976 included (1) on-line voltammetric techniques for the determination of UF_3/UF_4 that were refined and successfully and routinely applied in corrosion test loops, (2) measurements of protonated (and tritiated) species in the NaF-NaBF_4 coolant salt, and (3) the successful application of voltammetric and chronopotentiometric techniques to measurements of Fe^{2+} in $\text{LiF-BeF}_2\text{-ThF}_4$.

2.1.5 ABC Thermal-Hydraulics

The state of the art can be summarized as follows:

1. thermal-hydraulic data

A sufficient data base had been developed for the MSRE and also the MSBR. At this time, there is no consistent data base available for the ABC system that takes into consideration the changed fuel salt composition, the considerable changes during life of the molten fuel salt composition, mainly due to the accumulation of fission products and the rise in Pu concentration.²

The thermal-hydraulics data feed into blanket and ABC systems thermal-hydraulics codes that are available only to the extent that general systems codes have been adapted to the ABC design.

2. pressure drop data

A sufficient data base had been developed for the MSRE and also the MSBR. No such data base has been developed for the ABC molten salt fuel and coolant.

3. flow mixing data

The flow mixing issue (hot streaks, cold streaks) has not been addressed in any depth in the past. It is known that the radial power profile in the ABC blanket region is not flat, and that enrichment grading commonly used to flatten the radial power

² There are many property data available that can be used in thermal-hydraulic analyses. However, it is important to develop ABC-salt specific property data and not to rely on extrapolations as in case of the MSRE where such extrapolations led to a significant reduction in power output due to the use of incorrect specific heat data and air properties.

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distribution is not possible in a fluid fuel reactor. The temperature distribution at the blanket outlet will show a fairly large variation. It is unknown to what extent normal flow mixing is sufficient or whether enhanced mixing is called for to avoid cold and hot streaks in the blanket outlet plenum and in the IHX inlet plenum that can cause component damage.

4. cross flow data

It is known from Ft. St. Vrain HTGR operation that cross flow has to be taken into consideration for core that use stacked graphite blocks as moderator. Thermal and irradiation effects lead to uneven swelling/shrinkage of the graphite blocks which in turn will open gaps between the blocks within one column and also gaps between columns. Cross flow tests for HTGR configurations had been performed at the Commissariat a L'Energie Atomique (CEA) and also at GA. The results were considered insufficient as a base for cross flow assessments.

No data exist for molten salt cross flow in an internally moderated system using stacked graphite blocks as moderator.

5. graphite column vibration data

In support of the HTGR program, multi-column flow tests were conducted at GA in 1980 on an eight-block high core design. The tests showed that the columns were susceptible to flow-induced vibrations.

No data exist on flow-induced vibrations for graphite columns in a molten salt environment.

6. control rod vibration data

Control rod vibration tests were conducted in 1975 under the HTGR program on the Ft. St. Vrain control rod design. The tests showed that the control rods were susceptible to flow-induced vibrations.

The role of the control rod for the ABC system has not been identified yet. Reactivity control during operation is provided through measured fuel additions. Startup and shutdown control is provided through the accelerator controls. It might be necessary to have a control rod system as a backup shutdown system to the accelerator. In this case, and also depending on the design of the control rods or reactivity control devices, control rod vibration data might have to be obtained.

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7. blanket fluctuation data due to mechanical/thermal movements

Power fluctuations were initially observed in the Ft. St. Vrain reactor and attributed to mechanical/thermal movements of the graphite blocks/columns. This problem was corrected through the introduction of region constraint devices that eliminated the problem.

No information is available on ABC blanket fluctuation and, in case they can occur, how they could be eliminated.

8. cold bypass flow

For the MHTGR that uses a graphite block configuration similar to that proposed for the ABC blanket, cold bypass flow can range from 15 - 25% of the total flow. The major concern here is with fuel flowing in the gap between blocks where the gap has a much larger hydraulic diameter than the coolant (fuel) holes. The cold bypass flow issue is somewhat different for fluid fuel reactors than for solid fuel reactors because the heat is produced in the fluid fuel itself rather than being conducted into the coolant as in a solid fuel reactor. Assuming the same local power for fuel salt travelling in a small-diameter hole as compared to a large-diameter hole, the energy deposited in the fuel is proportional to the travel time through the holes - which is less for the large-diameter holes.

Neither analytical nor experimental assessments of this issue have been made for a system similar to the ABC system. Flow through gaps was not an issue for the MSRE and the MSBR.

2.1.6 Methods Validation

The MSRE data are the best source today for any method validation efforts because of the many measurements that have been made during the operation of the MSRE. However, those data are over 20 years old and not well documented; furthermore, they need to be reviewed for compliance with NQA-1 and -2 data requirements.

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2.1.7 Graphite Structures Performance

1. graphite block failure mode data

There is a limited data base on failure mode and failure loads of virgin HTGR graphite blocks, but no correlations for crack progression analysis were developed. Limited cracking in two Ft. St. Vrain fuel (graphite) blocks were observed after discharge from the reactor; however, the cracking was too minor to affect the functions of these blocks.

No data exist on combined mechanical and thermal/irradiation loads; no data exist for graphite block failure modes in a molten salt environment. It is also unknown whether fuel salt trapped in pores/cracks in the graphite can lead to local overheating and subsequent crack formation.

2. graphite blanket support strength data

Full-scale testing on Ft. St. Vrain core support structure mockups confirmed adequate safety margins against vertical loads. However a series of tests done on core support posts for the Large HTGR (2240 MW) showed premature failures of the seats. Later design changes eliminated those failures but while the discrepancies between experiment and prediction seem to have been corrected, the correlations were never validated.

There are no graphite strength data in a molten salt environment.

3. graphite structure dynamic strength data

In 1976, a series of dynamic tests on unirradiated graphite elements were performed in a pendulum rig. The results showed that the dynamic strength can reasonably well be predicted with static finite element codes. However, the tests were done on graphite elements not used in the MHTGR and ABC systems, made of H-327 graphite that is not on the list of graphite types under consideration for the ABC system.

No data exist on the failure of graphite structures under either mechanical loads or combined mechanical and thermal/irradiation loads.

4. seismic loads on graphite structure

Tests run in 1974 on a scaled model of the large HTGR to validate seismic load predictions were unsuccessful. No additional tests were ever run.

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2.2 VESSEL SYSTEM

The MSRE fuel inlet temperature was 1170°F (632°C), and the fuel outlet temperature was 1210°F (654°C). The MSRE vessel and all components that came in contact with molten salt were made of Hastelloy N and performed without any problems. Corrosion of the metal took the form of chromium removal from the surface. This process is limited by the diffusion rate of chromium in the alloy. Throughout the 22,000 hours that molten salt was in the fuel loop at 650°C, the chromium leaching was equivalent to the removal of all chromium from a layer only 0.01 mm deep. The leached chromium appeared in the salt where the concentration built up to about 100 ppm.

Corrosion was practically nil in the secondary salt system where LiF-BeF₂ circulated in Hastelloy N for more than 26,000 hours. The chromium concentration remained constant at about 37 ppm and post-operation examination of tubes from the air-cooled heat exchanger in this system showed no evidence of mass transport.

Hastelloy N in and around the reactor core did suffer irradiation damage referred to above. However, this damage did not affect the MSRE operation because the original design had made allowances for such effects.

The design operating conditions for the MSBR were advanced compared to the MSRE operating conditions in that the inlet temperature was lowered to 1050°F (565°C) but the outlet temperature was raised to 1300°F (704°C). Even further advanced are the inlet and outlet temperatures for the ABC system that call for a 577°C fuel inlet temperature and a 727°C fuel mixed mean outlet temperature³. At this time, there is no data base to support these operating conditions.

2.3 REACTIVITY CONTROL SYSTEMS

2.3.1 MSRE fueling system

Capsules containing fissile material were periodically immersed through the sampler tubes, located on the fuel pump (bowl), into the molten salt to release the fissile material as make-up fuel for the primary system. A capsule contained up to 85 grams of ²³⁵U. No problems were encountered except in two cases when the capsules being lowered in the fuel sampler hung up, causing the driver cable to tangle. Both times, the capsules were accidentally dropped into a cage in the pump bowl. Efforts at

³ Data are based on LANL viewgraphs presented August 24, 1994.

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retrieval were unsuccessful.

Altogether, the three salt samplers were used for over 550 sampling operations and the addition of 141 uranium enriching capsules.

2.3.2 MSRE control rod system

There has never been a failure of the MSRE control system to scram when requested.

During normal reactor operation, the MSRE system reactivity was monitored by a calculation executed every five minutes by an on-line digital computer. The computer was programmed to take signals of control rod position, neutron flux, and fuel temperature and compute a reactivity balance based on present conditions and the power history. The precision of the results was excellent, with the variation between successive calculations being generally less than 0.02% $\delta k/k$.

2.4 HEAT TRANSPORT SYSTEMS

2.4.1 heat exchangers

The MSRE employed an 8 MWt intermediate heat exchanger (IHX) where fuel salt transferred the heat to coolant salt, and an air-cooled secondary heat exchanger where the coolant salt transferred the heat to air. The MSRE operation has shown that conventional design calculations adequately predict heat transfer in the salt-salt heat exchanger and that there is no change in heat transfer over an extended period of operation. Heat transfer coefficients that were measured over a period of 15,000 hours did not show any change in the coefficient.

In the air-cooled heat exchanger, the measured salt-air coefficient proved to be 27% lower than the design value. This discrepancy was attributed to errors in the choice of air properties used in the design calculation. However, even after this correction had been made, the design coefficient was still higher than the observed value. This discrepancy was attributed to the inapplicability of the heat transfer correlations used in the design for the particular geometry of the air flow through the heat exchanger.

The intermediate heat exchanger for the MSBR was designed for 556 MWt which is a factor 70 larger than the MSRE IHX. The current ABC system design calls for three IHX, each designed for 168 MWt.

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2.4.2 pumps

Centrifugal pumps of conventional hydraulic design with a capacity of 1200 gpm and 850 gpm, respectively, were used in the MSRE for the circulation of fuel and coolant salts. The vertical shafts had oil-lubricated bearings. Oil leakage past the lower seal was drained out of the pump housing and a split purge of helium into the shaft annulus prevented oil vapors from going down into the pump bowl or gaseous fission products from coming up the shaft. The fuel pump had circulated salt for more than 18,000 hours, and the coolant pump for more than 22,000 hours.

The only problem with the salt pumps had been oil leakage into the pump bowl. A small amount of oil (about 1 cc per day) leaked from the seal drainage passage past a gasketed joint into the pump bowl. The oil thermally decomposed (it did not react chemically with the salt) and its products were drawn into the off-gas lines causing plugging of lines.

The fuel pump design for the MSBR had a capacity of 16,000 gpm (2,955 kg/s). This design had not been tested. The ABC system design calls for a fuel pump capable of circulating 567 kg/s.

2.4.3 valves

Freeze valves were used throughout the MSRE system taking the place of mechanical valves, in all piping containing salt. They operated as intended after the cooling-air controls were properly adjusted during the pre-nuclear startup of the reactor. It was learned that the valves in the salt transfer lines were slow to use (it took sometimes several hours to set them up). Fuel and coolant drain valves operated reliably, with thaw times between 10-15 minutes.

The MSRE freeze valves consisted of flattened 2 inch piping that was externally cooled. The MSBR had 6 inch drainage piping and it was realized that an extensive cooling system would be required to keep the salt frozen. A final conclusion had not been reached on the MSBR drainage valve design.

2.4.4 steam generator

The MSRE did not employ a steam generator. This is an as of yet unproven components although a detailed design had been developed under the MSBR program.

2.5 SHUTDOWN COOLING SYSTEM

The MSRE did not employ a separate shutdown cooling system. When fuel was drained from the primary system into the fuel drain tank, decay heat was removed via water-cooled bayonet tubes that extended down into thimbles in the drain tanks. The steam produced in the tubes was condensed in a steam drum and returned by gravity.

A different cooling system for the drain tank was developed for the MSBR. The drain tank was cooled by natural circulation of a ${}^7\text{LiF-BeF}_2$ salt mixture through drain tank tubes, and then cooling this salt by radiative heat transfer to boiling water.

2.6 AUXILIARY SYSTEMS

2.6.1 drainage system

The MSRE had two fuel drain tanks that were cooled by water as described in section 2.5. Fuel drainage in the MSRE was initiated by not cooling the freeze valves anymore and allowing it to thaw. This also led to fuel drainage in case of an extended power failure because the cooling air required to keep the valves frozen is interrupted.

The MSRE had also a separate coolant drainage tank that was also freeze-valve actuated.

The conceptual MSBR design shows a freeze plug between the reactor vessel and the drainage tank that is expected to allow fuel drainage to be completed within 7 minutes. The drain tank also serves as a 2 hr holdup volume for highly radioactive fission product gases after they are separated from the fuel salt; i.e., the fuel drain tank is part of the MSBR off-gas system.

Located under the steam generators of the MSBR are coolant drain tanks that are connected through freeze valves.

2.6.2 cover- and off-gas systems (incl. bubble generator)

For the MSRE, the bowl of the fuel pump was the surge space for the fuel loop. Dry, deoxygenated helium at 5 psig blanketed the salt. About 50 gpm of the 1200 gpm discharged by the pump was sprayed into the gas space to provide contact between salt and cover gas to allow ${}^{135}\text{Xe}$ to escape the salt. A flow of 4 l/min STP of helium carried the xenon and krypton out of the pump bowl, through a holdup volume

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providing about 40-minute delay, a filter station, and a pressure-control valve to charcoal beds. The charcoal beds consisted of pipes filled with charcoal, submerged in a water-filled pit at about 90°F. The beds were operated on a continuous-flow basis and delayed xenon for about 90 days and krypton for about 7 days. Thus only stable or long-lived gaseous nuclides were present in the helium which was discharged through a stack after passing through the beds.

The MSRE did not employ a separate bubble generator as part of a sparging system.

The main difference between the off-gas systems for the MSRE and the MSBR, besides the use of the drain tank for a 2 hr holdup of off-gases, is the addition of a gas cleanup system for the latter. Instead of discharging the helium through a stack it is cleaned and recycled.

As part of the MSBR development work, a venturi device was selected in which gas is injected into the venturi throat and bubbles are generated by the fluid turbulence in the diffuser section. A model of the bubble generator was tested under a variety of conditions of air flow and water flow rates, teardrop shapes and different throat widths.

A pipeline bubble separator was chosen to remove the gas-rich bubbles from the fuel salt. Tests were conducted in demineralized water and two modelling fluids, a 41% glycerin-water mixture and a 31% aqueous calcium chloride solution. Bubble separation efficiencies of greater than 80% were achieved.

Tests in salt have not been conducted, and a bubble generation and removal bypass system to the fuel pump have not been tested either.

2.6.3 salt purification system

Adjacent to the drain tank cell, the MSRE had a simple facility for processing the fuel or flush salt. The purpose of this facility was to (1) remove oxide contamination from the salt should this be necessary, and (2) to recover uranium from the salt at the end of the experiment.

The MSBR used continuous reprocessing that included also fuel cleanup. The MSBR design does not show facilities for coolants salt purification even though this should not be a difficult task.

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2.6.4 fuel processing

Fluorination to recover the original charge of uranium was accomplished for the MSRE. Mixtures of fluorine and helium were bubbled at rates up to 33 std liters/minute of fluorine through a 1 in. dip tube immersed 64 in. in the tank of salt. At the beginning of the processing there was a high utilization of fluorine as the UF_4 was raised to UF_5 . Then volatile UF_6 began to be formed and the fluorine utilization decreased to about 30%. The effluent gas stream passed through a bed of NaF pellets at 750°F to remove volatile impurities and then into a series of canisters filled with NaF pellets at 200 - 300°F where the UF_6 was absorbed.

The uranium concentration in the flush salt was reduced to 7 ppm in 7 hours of fluorination. In the fuel salt processing, 47 hours of fluorination removed 218 kg of uranium from the 4730 kg of salt, leaving a concentration of 26 ppm U.

Decontamination of uranium from the fission products was very effective; the processing tank read over 2000 r/hr while the canisters containing 12 kg U read less than 0.002 r/hr. The measured decontamination factors for gross beta and gamma activities were 1.2×10^9 and 8.6×10^8 , respectively.

At the end of the fluorination, the fuel salt contained 850 ppm Ni, 410 ppm Fe, and 435 ppm Cr as fluorides. These fluorides were reduced by treatment with hydrogen and finely divided zirconium. This treatment reduced the corrosion products to 60 ppm Ni, 110 ppm Fe and 25 ppm Cr.

An extensive fuel processing system for the isolation of ^{233}Pa and the removal of fission products was designed for the MSBR. All processing steps had been demonstrated in lab experiments, but the overall system was never assembled and tested.

2.6.5 containment

The MSRE design aimed at zero leakage from the system of piping and vessels that formed the primary containment for the fission products. In addition, a secondary containment was provided to limit the release of fission products to the environment in the event of a failure in the primary containment. Stringent leakage criteria were imposed on the secondary containment. For example, the reactor cell and drain tank cell were required to leak at a rate of less than 1% per day at 39 psig.

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The primary containment was achieved by use of welded construction with a minimum of gasketed joints (that were pressure-buffered). Zero leakage had been attained during all periods of operation.

The MSBR employed an arrangement where the reactor and steam generator facilities are located inside a reactor building and in steam cells, located on one reinforced concrete pad, and the remaining structures on another. This provides for a seismic isolation because no piping or connections containing radioactive material would cross the boundary between the pads.

2.6.6 cell and component heating

All salt piping and vessels of the MSRE were electrically heated to prepare for salt filling and to keep the salt molten when there is no nuclear power. In the reactor and drain tank cells, where radiation levels make remote maintenance necessary, heater elements and reflective metal insulation were combined in removable units.

The salt-air heat exchanger was equipped with doors that dropped to block the air duct and seal the radiator enclosure if the coolant salt circulation stopped and there was danger of freezing salt in the tubes.

Movable heaters were used to unplug lines (for example, in the off-gas lines).

An elaborate electric cell heating system was designed for the MSBR. Heat losses from the equipment were considered sufficient to maintain most cells at 1000 F - 1100 F. During initial warmup, downtime, or possibly at low power levels, electric space heaters were proposed for cell heating. The overall heatup power requirements were estimated to about 1.3 MW.

2.6.7 electric power systems

The MSRE used as auxiliary power supply for instruments and controls a system of storage batteries kept charged by an ac-dc motor-generator set. A static dc-ac inverter changed the power from the batteries into the ac required for the instruments and control circuits. Without the charger, the batteries could deliver 100 kW of power at 250 V for at least an hour.

No additional standby power (e.g., for salt circulation pumps, freeze valve coolant pumps, cell cooling system, etc.) was supplied.

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For the MSBR, the same instrument power supplies were proposed. In addition standby power of about 10 MW was requested for salt circulation pumps, freeze valve cooling pumps, cell heating systems, etc.

2.6.8 coolant processing

The MSRE had no direct provision for secondary coolant processing. However, the coolant pump bowl provided for sampling and the coolant could be drained for storage. If deemed necessary, off-site coolant processing could have been performed.

The same arrangement for coolant processing can be applied to the MSBR.

CHAPTER 3

TECHNOLOGY DEVELOPMENT NEEDS

The ABC Design Data Needs referred to earlier (see Chapter 1) are summary statements of ABC technology development needs in a prescribed format. In addition to the ABC technology needs they also identify designer alternatives in case a particular design data need cannot be addressed, and identify fallback positions in case of an uncertain or unfavorable outcome of the technology development work.

The ABC technology development tasks addressed here were extracted from the ABC Design Data Needs report and classified either as near-term, intermediate-term, or long-term technology development needs. Only experimental work is addressed here. Any design development work would have to be addressed under a separate design activities task.

3.1 Relationship Between the MSBR/MSRE and the ABC System

3.1.1 unique MSRE/MSBR issues

It is important to remember that in 1970 the molten salt reactor technology had already been advanced enough, after the successful operation of the Molten Salt Reactor Experiment (MSRE) and years of comprehensive technology development work, to call for the building of a demonstration plant. Many of the problems addressed then dealt with the use of molten salt fuel in a thermal breeder reactor (MSBR). Keeping the fission product inventory low was a necessity for successful breeding and required fast and efficient fuel reprocessing on a continuing basis with very low losses in fissile material. The continuous and efficient removal of Xe and Kr to avoid severe neutronics penalties when these fission products would be absorbed by the core graphite, was another important requirement. The development of a high-quality graphite with a long life was necessary to avoid long downtimes for the graphite replacement. Because the MSBR worked in the Th cycle, the core power density was constrained by the maximum allowable number of absorptions in Pa, the precursor for ^{233}U . Furthermore, particular attention had to be given to the separation of Pa during the reprocessing.

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3.1.2 significance of MSBR issues for the ABC system

The ABC system does not face the problems described in section 3.1.1, or they are far less important. Using an externally moderated or unmoderated ABC configuration would eliminate the problems and complications coming from graphite use as an internal moderator, ranging from graphite lifetime limits and adsorptions of gaseous fission products in graphite to the possible use of Hitec as a secondary coolant. For the ABC system, there is no need for continuous reprocessing; if any fuel processing besides possibly the removal of rare earth trifluorides is required at all, it could be done in about 10 or more year intervals. Because the mission of the ABC system is the effective destruction of weapons grade plutonium fuel, breeding of fuel is not even desired. Not using thorium in the fuel eliminates any problems in regard to Pa and the resulting power density constraints. While the stripping of the noble gases is still desirable, it is not necessary to aim for very high removal efficiencies in internally moderated systems. Furthermore, by not using an internally moderated ABC blanket configuration, any removal of noble gases would be greatly simplified because there is no more competition for the noble gases between the pores in graphite and the helium gas bubbles from the gas sparging system, that could detrimentally affect the neutronics.

3.1.3 common issues MSBR-ABC system

Many of the technology issues are the same for the MSBR and the ABC system. The MSRE operation pointed to some problems with the use of standard Hastelloy N in the core and its embrittlement due to helium production and interaction with tellurium. The embrittlement issue needs to be resolved, most likely through the addition of small amounts of niobium to standard Hastelloy N. The use of iron-based alloys is another possibility at least for the secondary circuit.

While the MSRE coolant (LiF-BeF_2) performed without any problems, this coolant is not well suited for a steam cycle electricity-generating plant because of its high cost and the requirement for extensive feedwater preheating due to its high melting point. For this reason, a mixture of sodium fluoride and sodium fluoroborate was proposed for the MSBR as coolant. In addition to a much lower melting point, this coolant has the ability to trap tritium and prevent it to be released to the environment through the steam system. However, this coolant can react with the fuel in case of a leak. Furthermore, even under normal operating conditions it slowly thermally decomposes and produces BF_3 which is in equilibrium with the coolant salt. The gaseous BF_3 is swept out of the pump bowl and if no makeup BF_3 is added, this would slowly change the coolant composition and increase its melting point. The coolant chemistry and the

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design implications need to be better understood.

The MSRE operating experience showed that the noble metal fission product behavior could not be well predicted and more insight has to be gained in the noble metal fission product transport. A clarification of this issue is also required for the ABC system.

There is only limited experience on in-line monitoring and surveillance of fuel and coolant salts available. For both the MSBR and the ABC system, extensive technology development needs have been identified. However, the resolution of most of these issues requires only small-scale experiments.

Considerable efforts are required in the component development area. In most instances, the issue is the proper scaling of equipment that performed well in the past (e.g., pumps, valves). However, there are also new developments required such as the steam generator.

3.1.4 differences between ABC system and MSBR

The most significant difference between the MSBR and the ABC system is the operation of the ABC system in the subcritical state. This has significant design implications. However, it is not a controlling issue for the base technology development program.

What most distinguishes the base technology for the MSBR from the ABC system fuel is (1) the use of plutonium trifluoride as fissile material, (2) the absence of fertile material, and (3) the very high level of fission product accumulation in the fuel salt due to the absence of continuous reprocessing. Of particular concern here are the concentrations of actinide and rare earth trifluorides whose solubility limits constrain fuel salt life. How fuel properties change due to the increasing amount of plutonium and fission products in the fuel salt is unknown.

Another important difference between the MSBR and ABC system is the change in fuel redox potential due to the use of plutonium trifluoride fuel which would tend to lead to a reducing rather than oxidizing environment with as of yet unquantifiable consequences.

A significant difference between the two molten salt system concepts is the absence of fertile material for the ABC system, that will tend to lead to a prompt positive prompt temperature coefficient. Adding resonance absorbers (such as ^{167}Er) can lead

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to a strong negative prompt temperature coefficient as shown in the TRIGA reactor. However, the performance of those absorbers in a molten salt environment still has to be assessed.

3.1.5 new ABC issues

An ABC system can be expected to be much simpler than a MSBR for the reasons cited above. However, the ABC system introduces new issues because of its linkage to an accelerator, the difference in mission and the use of weapons grade plutonium as fuel that is dissolved in a different salt.

Accelerator issues and blanket-accelerator interface issues will not be addressed in this technology development plan. This will be done in a separate technology development plan when the design has progressed to the conceptual level.

Much of the ABC technology development work aims at the creation of data bases that characterize fuel, coolant, graphite, and structural metals performance under both normal operating and accident conditions. Because of the large accumulation of fission products in the fuel salt, fission product interactions among themselves and with fuel salt constituents, with graphite and structural metals are potentially of greater significance than for the MSBR with its low level of fission products. Higher actinides will be present in much higher concentrations than in the MSBR, and their chemistry is not well known.

Because plutonium fission release only three fluorine atoms compared to four in uranium fissions, the redox potential of the fuel is different and has to be characterized with all its implications.

One of the advantages of the subcritical operation of the ABC system is the possibility to control its fission rate through an external neutron source, the accelerator-driven target. This potentially eliminates one of the problems common to all "deep burn" reactor concepts, namely the reactivity control in a core that contains a large amount of fission products which greatly reduce any control poison worth. How effective and reliable the external neutron source control is has to be determined.

Mentioned earlier was the temperature coefficient issue because cores which contain only ^{239}Pu as fuel without any fertile material tend to have a strong prompt positive temperature coefficient. How configuration modifications and the addition of resonance absorbers can yield a prompt negative temperature coefficient in the presence of high fission product concentrations has to be determined and demon-

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strated.

3.1.6 changes in regulatory requirements

Many of the topics addressed in this technology development plan are in response to state of the art requirements for licensing that are more stringent today than 25 years ago. For example, many of the existing data on molten salt and material performance might not be readily applicable in support of licensing because of the lack of quality assurance detail available today for those data.

Computer code verification and analysis method validation are required following highly formalized guidelines, which will require significant efforts.

Regulatory requirements today demand a clearly demonstrated defense in depth capability without mostly relying on a single barrier for radionuclide releases. This will tend to favor a pool concept over a loop concept (like the MSBR) and will guide the design of the vessel systems.

More than in the past are passive and inherent safety features emphasized. These features are particularly important for safe and reliable shutdown, reliable decay heat removal and the availability of a reliable heat sink.

Functional separations of systems, structures and components and clearly identifiable separations of safety-related from non-safety-related systems, structures, and components will also affect the design and some of the issues that need to be addressed under a technology development program plan.

3.2 Technology Demonstration

The demonstration of the readiness of the ABC technology is done in three steps:

1. proof of principle demonstration
2. demonstration plant operation
3. prototype operation

The proof of principles experiment has to demonstrate the following:

- an accelerator can be coupled to a subcritical blanket and produce fission heat in a controllable manner, using a separately tested target design

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- use of accelerator controls to control the ABC blanket operation
- the heat produced in the target and blanket can be rejected through heat exchanger - pump - piping systems in a controlled manner
- effective gas sparging for an internally moderated configuration can be demonstrated
- salt chemistry control can be demonstrated
- analytical and diagnostic techniques can be demonstrated
- the principles for remote maintenance, repair, and replacements can be demonstrated

Initially, the proof of principle experiment does not use fissile material. However, a still to be determined amount of fissile material will be added later to achieve a predetermined level of subcriticality and continued energy production. It might be possible to use the same facility for the proof-of-principle experiment and the demonstration experiment. However, the issues of configuration prototypicality and containment need to be addressed.

In a demonstration plant all systems, structures and components of the full-size plant design are represented, albeit on a smaller scale, to demonstrate system integration, safe and reliable operation, including maintenance, repair and replacement operations. The intent of the demonstration plant is to demonstrate that the ABC system is capable of safely, reliably and economically achieving the mission objective of destroying plutonium at a predicted rate.

The operation of the prototype precedes the full-scale operation. It is the operation of a single module for several years and, when successfully concluded, the addition of the remaining modules to the full-size plant.

3.3 near-term technology development needs

Near-term technology development needs identify high-priority issues that need to be addressed at the earliest possible time because of their high impact on design and the assurance of ABC design feasibility and technical viability of the chosen design. They cover also issues that need to be resolved for the early experiments such as the proof-of-principle experiment. Typical for this near-term technology development category are small-scale experiments.

Near-term technology needs are summarized in Table 3-1.

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3.4 intermediate-term technology development needs

Intermediate-term technology development needs are those that fall between the long-term and near-term needs. Results from those experiments are not needed in the early phase of the design. However, there are instances where work on those needs has to commence early because of long lead times for equipment and/or the long duration of the experiments (e.g., long-term corrosion tests). Typically, such experimental data are obtained from large-scale experiments.

Intermediate-term technology needs are summarized in Table 3-2.

3.5 long-term technology development needs

Long-term technology development needs are those where experimental results are required at a later stage. These data needs will complement existing data bases. Very often, producing those data is routine work using well-established techniques in existing facilities. As in the previous category, work on long-term technology development needs might have to begin early in the project because of long lead times for the procurement of equipment (e.g., steam generator tests) and the long duration of the tests. Typically, such experimental data are obtained from large-scale experiments.

Long-term technology needs are summarized in Table 3-3.

Table 3-1: Near-Term Technology Development Needs

1. Chemical Research and Development

1.1 Fuel and Fission Product Chemistry

PuF₃ phase diagrams
solubility of rare earth trifluorides
solubility of actinide fluorides
fuel-coolant interaction
physical properties of fuel salt
use of Ce to simulate Pu

1.2 Coolant Chemistry

chemistry of sodium fluoroborate and trapping process for tritium
corrosion chemistry of fluoroborate
BF₃ effects
solubility of elemental hydrogen (tritium) in fluoroborate

1.3 Tritium Behavior

techniques for measured additions of tritium
tritium solubility in fuel and coolant salt
solubility of tritium fluoride in fuel salt
tritium permeability for Hastelloy N
TF/T₂ ratio as a function of redox potential
tritium removal efficiency through gas sparging

2. Graphite Research and Development

graphite stock development and qualification
special graphite treatment for property improvement
sealing graphite to reduce graphite permeability
physical properties

Table 3-1: Near-Term Technology Development Needs (Cont'd)

3. Metals Research and Development

applicability of existing Hastelloy N information
modification of Hastelloy N
material and irradiation properties of modified Hastelloy N candidates
material specs for modified Hastelloy N
vendor qualification
irradiation of commercial heats and material characterization
material selection for auxiliary systems
tests of double-walled piping
basic metal studies
stainless steel applications
material specifications for stainless steel candidates

4. Monitoring and Surveillance Research and Development

begin with selection and/or development of methods and devices for near-term experiments
redox conditions of fuel
plutonium and other transuranics concentrations
rare earth fission product concentrations
corrosion product ions
valence state and concentrations of some fission products
O² content of fuel
free and combined hydrogen and tritium content in fuel and coolant
oxygenated, protonated and tritiated species in fuel and coolant
gas stream monitoring
design of in-line analytical test facility

5. Thermal-Hydraulics Design Development

thermal-hydraulics data
pressure drop data

6. Methods Validation

no activity

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Table 3-1: Near-Term Technology Development Needs (Cont'd)

7. Components, Systems and Structures Technology Development

electrolytic deposition of semi-noble metals
electrolytic deposition of noble metals

8. Large Scale Experiments

begin with experiments in support of design of major test loops
begin with experiments in support of design of in-line analytical test facility
begin with experiments in support of design of proof of principle experiment
begin with experiments in support of design of demonstration experiment

Table 3-2: Intermediate-Term Technology Development Needs

1. Chemical Research and Development

1.1 Fuel and Fission Product Chemistry

bubble generator and gas separator design and effectiveness
removal of gaseous fission products
noble metals chemistry
solubility of actinide oxides
 $\text{Pu}^{3+}/\text{Pu}^{4+}$ equilibrium
solubility of corrosion products
tellurium chemistry
fundamental studies on molten salts
freezing and thawing behavior of molten salts

1.2 Coolant Chemistry

gas-liquid equilibrium in the pump bowl
coolant purification
irradiation stability
freezing behavior

1.3 Tritium Behavior

tritium fluoride corrosion behavior
tritium and tritium fluoride diffusivities in fuel and coolant salt
tritium permeation through other structural materials
equilibria between important species containing tritium in molten salt systems
isotopic exchange between tritium and chemically bound hydrogen
tritium absorption on carbon in contact with coolant
metal coating effect on tritium confinement

2. Graphite Research and Development

basic graphite studies
physical properties
fracture mechanics data
graphite destructive and non-destructive examination data
corrosion data

**Table 3-2: Intermediate-Term Technology Development Needs
(Cont'd)**

3. Metals Research and Development

- basic metal studies
- corrosion loops
- fabricability studies
- determination of limiting conditions
- ASME Boiler Code case
- material optimization
- effect of irradiation acceleration
- interpretation of small-sample irradiation tests

4. Monitoring and Surveillance Research and Development

- evaluation of methods and devices
- determine achievable accuracies
- develop calibration standards
- develop quality assurance and control procedures
- out-of-pile and in-pile testing of methods and devices

5. Thermal-Hydraulics Design Development

- flow mixing data
- cross flow data
- cold bypass flow
- graphite column vibration data
- control rod vibration data
- blanket fluctuation data

6. Methods Validation Data

- use of proof of principle experiment to validate analysis methods

7. Components, Systems and Structures Technology Development

- graphite structure tests

**Table 3-2: Intermediate-Term Technology Development Needs
(Cont'd)**

8. Large Scale experiments

- operate major tests loops
- operate in-line analytical test facility
- operate proof of principle experiment
- begin to operate demonstration plant design

Table 3-3: Long-Term Technology Development Needs

1. Chemical Research and Development

1.1 Fuel and Fission Product Chemistry

- completion of fundamental studies on molten salts
- completion of tellurium chemistry work
- freezing of molten salts containing fission products
- completion of large-scale experimental work

1.2 Coolant Chemistry

- long-term test loop operation
- resolution of anomalies observed in ORNL's Coolant Salt Technology Facility (CSTF)
- impact of coolant leaks

1.3 Tritium behavior

- oxide stability in steam system

2. Graphite Research and Development

- graphite-molten salt interactions studies
- corrosion data

3. Metals Research and Development

- completion of work on:

- corrosion loops
- mockup tests
- completion of data bases
- qualification of reference alloys

Table 3-3: Long-Term Technology Development Needs (Cont'd)

4. Monitoring and Surveillance Research and Development

- evaluation of monitoring and surveillance performance
- selection of reference methods and devices
- develop techniques for maintenance, repair, and replacement of monitoring and surveillance devices

5. Thermal-Hydraulics Design Development

- activities will depend on design selections made

6. Methods Validation

- obtain validation data from large-scale experiments

7. Components, Systems and Structures Technology Development

- TBD

8. Large Scale Experiments

- complete work on:

- major test loops
 - plant protection and control simulation facility
 - in-line analytical test facility
 - proof-of-principle experiment
 - demonstration experiment

- build and operate prototype

CHAPTER 4.0

ABC TECHNOLOGY DEVELOPMENT PROGRAM

For each of the ABC technology development tasks, near-term, intermediate-term and long-term activities have been identified.

Near-term technology development needs identify high-priority technology issues that need to be addressed at the earliest possible time because of their high impact on the selection of basic design features and the assurance of design feasibility. They cover also issues that need to be resolved for the early experiments in support of the technical viability of the ABC concept such as the proof-of-principle experiment. Typical for this category are small-scale experiments.

Intermediate-term technology development needs are those that fall between the long-term and near-term needs. Results from those experiments are not needed in the early phases of the design. However, there are instances where work on those needs has to commence early because, for example, of long lead times for equipment procurement and installation and the long duration of experiments (e.g., long-term corrosion tests). Typically, such experimental data are obtained from large-scale experiments.

Long-term technology development needs are those where experimental results are required at a later stage. These data needs complement existing data bases and very often producing those data is routine work using well-established techniques in existing facilities. Furthermore, technology demonstrations such as the operation of the demonstration plant and the prototype fall into this category. As in the previous category, work on long-term technology needs might have to begin early in the project because of long lead times for the procurement of equipment (e.g., steam generator tests) and the long duration of the tests. Typically, such experimental data are obtained from large-scale experiments.

The ABC technology development areas are grouped as follows:

4.1 chemical research and development

fuel and fission products
coolant
tritium

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- 4.2 graphite research and development
- 4.3 metals research and development
- 4.4 monitoring and surveillance research and development
- 4.5 thermal hydraulics design development
- 4.6 methods validation
- 4.7 components, systems and structures technology development

- fuel and coolant salt
- reactivity control
- molten salt pumps
- valves
- vessels
- graphite structures
- salt-salt heat exchangers
- active and passive decay heat removal systems
- steam technology
- cover- and off-gas systems
- purification systems
- salt processing systems
- instrumentation and control
- cell heating and insulation
- containment
- drainage and storage
- remote maintenance and repair
- systems and components replacements

4.8 large scale experiments

- major test loops
- plant protection and control simulation
- in-line analytical test facility
- proof-of-principle experiment
- demonstration experiment
- prototype operation

4.1 CHEMICAL RESEARCH AND DEVELOPMENT

The objective of this activity is to obtain the chemical information necessary for the design of the ABC system. Work in this activity will include studies of fuel and fission product chemistry, coolant chemistry, and tritium behavior.

4.1.1 FUEL AND FISSION PRODUCT CHEMISTRY

The fuel and fission product chemistry studies include solubility studies for rare earth fluorides and actinide fluorides and oxides, measurement of required physical properties, fuel-coolant interaction, noble metal behavior, and gas sparging.

4.1.1.1 near-term

- PuF₃ phase diagrams

Phase diagrams for a system consisting of LiF-BeF₂-PuF₃-ZrF₄ have to be determined to provide the basis for the optimal ABC fuel salt composition. The current choice for the ABC system is 65 mole % ⁷LiF - 29.99 mole % BeF₂ - 5 mole % ZrF₄ in which 0.01 mole % PuF₃ is dissolved. In the early stages of this task, Ce will be substituted for Pu. The validity of substituting Ce for Pu has to be assessed later.

Of interest is also the effect of large amounts of fission products in the fuel salt melt on the phase diagram.

- solubility of rare earth fluorides

Because the concentration of rare earth fluorides is potentially limiting the fuel melt lifetime, the solubility of rare earth fluorides in the reference fuel melt has to be determined. Furthermore, because of the buildup of fission products in the fuel melt and the buildup of the plutonium inventory causing the change in the fuel melt composition, the solubility of rare earth fluorides in a fuel melt with a changed composition has to be determined, too.

- solubility of actinide fluorides

While MSRE operating experience did not indicate any difficulties in regard to actinide trifluoride solubility, the ABC system will operate at much higher actinide concentrations. The solubility of actinide fluorides, including higher actinides, has to be determined. Furthermore, support has to be provided for the assumption that all rare

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earth and actinide trifluorides act essentially as a single element.

- fuel-coolant interaction

These experiments are predicated of the use of sodium fluoroborate as a secondary coolant.

Areas that need to be investigated are:

- coolant composition stability vs BF_3 partial pressure,
- tritium trapping and release mechanism when coolant and fuel salt mix,
- behavior of oxide and hydroxide ions in fluoroborate melts and when mixed with fuel salt,
- behavior of corrosion products in fluoroborate melts when fuel salt and coolant salt mix;
- BF_3 pressure under static and dynamic conditions including the possibility of pressure surges
- determination of high-melting phases when fuel salt and fluoroborates mix

Two types of experiments are required. One set of experiments deals with gaining an understanding of the basic chemistry of the fuel-coolant interaction. Of particular interest are non-equilibrium conditions. A second set of experiments studies of these interaction phenomena will be conducted 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.

Experiments are also required to determine the possibility of forming phases with high melting points when fuel and coolant mix.

The possibility of freezing a phase having a higher melting point in the vicinity of a leak due to non-equilibrium concentrations near the leak will be investigated.

- physical properties of fuel salt with and without fission product elements present

This task consists of a review of physical properties that were compiled last in 1968. In the meantime many additional measurements were made that need to be evaluated together with the old data. Based on this critical review those physical properties of the fuel salt will be identified that are either too uncertain to be useful for design, or

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have not been measured at all.

Because fuel salt properties change when the salt composition changes due to burnup, this effect has to be assessed experimentally.

- use of Ce to simulate Pu

It is common practice to use Ce as a substitute for Pu. The uncertainties coming from this substitution need to be assessed. This issue is already partly addressed during the determination of the plutonium fluoride phase diagrams in this section.

4.1.1.2 intermediate term

- bubble generator and gas separator design and effectiveness

The bubble generator will produce small helium bubbles in the salt which will be circulated with the salt. Noble fission gases are insoluble in the salt. The presence of the small helium bubbles will "soak up" the noble fission gases and greatly reduce their migration to the pores in graphite.

The activities under this task will focus on the selection of a bubble generator concept (e.g., a venturi device), the design of the bubble generator which will produce bubbles of a certain size and amount, the design of the bubble separator which will separate the dissolved gas from the molten salt, and the integration of the bubble addition and removal system in form of, for example, a pump bypass system.

Initial tests will be conducted in water using air injections. Bubble generation will be characterized and bubble removal effectiveness will be measured.

At a later stage, experiments have to be conducted in molten salt using helium for the bubble generation.

- removal of gaseous fission products through gas bubbles with or without graphite present

It is expected that the noble gases in a molten salt migrate from the molten salt where they originated to sinks in accordance with the laws of mass transfer where mass transfer coefficients control the migration rate. The sinks for the noble gases are any salt-gas interfaces such as the voids in the graphite, the gas space in the pump bowl, and helium bubbles. An analytical model will be developed and validated that describes the migration of noble gases in an ABC primary system molten salt

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environment. The results of this work will be constraints on the graphite porosity, the required magnitude of the fuel bypass flow, the stripping efficiency for the noble gases per cycle, the steady-state gas fraction in the molten salt and guide the design of the off-gas system.

Both argon and helium will be investigated as sparging gases.

The various mass transfer coefficients need to be critically evaluated. Particular attention has to be given to the mass transfer coefficient to bubbles suspended in a turbulent fluid.

- noble metals chemistry

The chemical behavior of the noble metal fission products (Nb, Mo, Ru, Ag, Pd, Cd, and Tc) in the MSRE had not been predictable. Some of the metals are present in metallic form, some as soluble but instable fluorides, and some possibly as volatile gas species. To better understand the noble metal chemistry, the following topics need to be addressed:

- determination of possible oxidation states and the nature of the species which these elements form in molten salt fluoride solution
- equilibrium of these species under controlled redox potentials to determine their thermodynamic stability
- redox chemistry of noble metals by reactions involving trace concentrations of oxide
- reactions with graphite
- kinetics of noble metals agglomeration and deposition as a function of flow rate, chemical environment, their diffusivities, physical and chemical condition of surfaces available for deposition, presence of colloidal (metallic, graphite, oxide, etc.) particles
- resuspension (liftoff) of deposited particles bearing noble metals
- distribution of noble metals between molten salt and cover gas

The proposed electrolytic deposition of noble metals as a means for removing them from the fuel salt has to be assessed. The effectiveness of such a process depends in large part of the chemical state in which the noble metals are present in the fuel salt. Furthermore, the effectiveness of providing large metallic surfaces (e.g., in the heat exchanger) for noble metal deposition has to be investigated, considering both the deposition effects and reentrainment effects.

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- solubility of higher and lower actinide oxides over the range of possible redox potential values

The ABC fuel must be protected from oxide contamination to avoid the precipitation of actinides whose solubility in the fuel salt is very low. In the past, the main focus was on the solubility of the oxide forms of Th, Pa, U, and Np with little attention paid to Pu because it existed only in small concentrations. Higher actinides had received no attention. The U^{3+}/U^{4+} ratio as a measure of the oxidizing conditions was found to be an important parameter.

The ABC system is fueled with Pu and, particularly late in life, significant levels of higher actinides such as americium and curium have been produced. Further, the oxidizing conditions will be different.

The solubility of lower and higher actinide oxides over the range of possible redox potential values has to be determined.

- Pu^{3+}/Pu^{4+} equilibrium

The control of the U^{3+}/U^{4+} ratio was shown to be an important parameter for the MSRE. To maintain a non-corrosive environment, beryllium had been added periodically to create a deficiency in fluorine by forming BeF_2 and reducing U^{4+} to U^{3+} .

The predominant fluoride form for Pu is PuF_3 . Because the valence of the fission products is greater than 3, it seems that corrosion caused by excess fluorine is not a concern. However, it has to be known to what extent Pu^{3+} can be oxidized to Pu^{4+} under equilibrium conditions and how this is affected (and affects) the formation of fission product fluorides. The upper and lower limits for the Pu^{3+}/Pu^{4+} ratio have to be known.

- solubility of corrosion products

While corrosion rates are expected to be low in the ABC system using a modified Hastelloy N, during the long salt life without reprocessing these corrosion products accumulate and their solubility has to be determined.

- tellurium chemistry

The reason for singling out tellurium from the rest of the fission products is the effect this fission product has on the intergranular attack on Hastelloy N that has been observed in the MSRE experiment. Tellurium is expected to be present in the fuel melt

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as Te_2 , and to a lesser extent as Te , and/or in a reduced state. Information on this equilibrium must be developed to understand the tellurium chemistry in regard to the cracking of Hastelloy N. The next three subtasks will provide this basis.

1. solubility of tellurium gas and of the solid tellurides under various redox conditions

The solubility of elemental tellurium in molten salts is expected to be very low. To obtain reliable solubility measurements requires a sophisticated experimental design (e.g., the use of radiochemical-tracer tellurium that would require glovebox operation).

The tellurides LiTe_3 and Li_2Te are expected to be much more soluble and a different experimental setup is required for solubility measurements.

Because the redox potential affects the solubility, its effect has to be determined.

2. behavior of Te_3^- in the presence of strong metallic reductants

Te^{3-} is unstable in the presence of strong metallic reductants such as Cr and Be. It needs to be determined whether more noble metals and alloys (Ni, Hastelloy N) can decompose Te^{3-} .

3. oxidation-reduction chemistry of tellurium in fuel salt containing Pu

Experiments need to be designed to measure the effect of the redox potential in the fuel salt on the species of tellurium present and on their distribution between gas, liquid and solid phases.

- fundamental studies on molten salts

Any chemical reaction that is thermodynamically favored can be expected to occur in a molten salt environment. The objective of this task is to accumulate information on:

- free formation energies
- activity coefficients

for all important substances present in the fuel with sufficient accuracies to define the course of all relevant chemical reactions that can occur in the temperature range of interest and the salt composition that has been selected.

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- freezing and thawing behavior of molten salts

While the ABC design provides assurance that salt freezing does not occur (except in cases where it is desired such as in freeze valves), there are accident scenarios where salt freezing cannot be excluded under all circumstances. Of interest are possible separations of salt constituents in the process of freezing, for both ternary and quaternary melts. Furthermore, it also needs to be understood how the thawing process proceeds.

4.1.1.3 long-term

- completion of fundamental studies on molten salts

The fundamental studies on molten salts will be completed.

- completion of tellurium chemistry work

The tellurium chemistry work will be completed using information obtained from post-irradiation examinations of irradiated Hastelloy N.

- freezing of molten salts containing fission products

In case of a spill, molten salt will come in contact with colder surfaces and freeze. The freezing process of molten salts containing fission products (that produce decay heat) and the fission product retention capability of frozen salts will be determined.

- completions of large-scale experimental work

The large-scale experiments such as the proof-of-principle and demonstration experiments provide a link from laboratory scale experiments to large-scale experiments in prototypical molten salt environments which represents a validation of laboratory-scale experiments and a source of data that cannot be obtained in lab-scale experiments.

4.1.2 COOLANT CHEMISTRY

A different technology development program is required for this task in case the reference secondary coolant is not sodium fluoroborate. Fluoroborate salt had been proposed as secondary coolant for the MSBR but its shortcomings were also recognized.

The reference coolant composition is $\text{NaBF}_4\text{-NaF}$ (92-8 mole %). The major advantages of this coolant over the ${}^7\text{LiF-BeF}_2$ coolant used in the MSRE are its low melting point, its low cost and its ability to trap tritium that has been demonstrated in ORNL's Coolant Salt Technology Facility (CSTF). The following list of tasks describe coolant chemistry issues that were not addressed in the CSTF and also issues that were raised during the operation of the CSTF and left unanswered.

4.1.2.1 near-term

- chemistry of sodium fluoroborates and trapping process for tritium (oxide and hydroxide chemistry of fluoroborates)

The presence of oxide and hydroxide species in the coolant, even at very low concentrations, affects corrosion of Hastelloy N and reactions with tritium. At high concentrations (e.g., due to a steam leak into the coolant) the corrosion will increase but the vapor over the coolant would change from all BF_3 to a mixture of BF_3 , oxygen and hydrogen. The chemical formula and structure of oxide and hydroxide species both in the melt and in the vapor are important information to address the corrosion, tritium control and coolant purification issues.

Similarly important is the equilibrium in the reaction:

that occurs when water leaks into the coolant. The study of this reaction is made difficult by the presence of gaseous BF_3 and B-O-F compounds and other compounds. Hydrolysis equilibria will be investigated and the composition of solid, liquid and vapor phases will be studied.

At this time, no firm conclusions can be drawn in regard to the mechanism by which tritium is trapped. Some results indicate that $\text{BF}_3 \cdot 2\text{H}_2\text{O}$ may exist as a molecular compound in the vapor and could be responsible for the tritium trapping. However, this hypothesis need confirmation. It was also concluded that tritium trapping would be favored by relatively high concentrations of oxide and/or metal fluorides in the salt

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along with a relatively high value for the partial pressure of water.¹

- corrosion chemistry of fluoroborates

The free energy of formation of corrosion products such as nickel, iron and chromium and titanium are required data for the study of the corrosion thermodynamics of Hastelloy N as a containment metal for the sodium fluoroborate coolant. These free energies must be known not only to characterize the extent of corrosion but also to predict the redox reactions that can reverse or counter the metal oxidation.

Similar studies have to be conducted for stainless steel as containment metal.

The potential for boride formation has to be investigated. Should boride formation occur and should boron diffuse into the Hastelloy N, embrittlement might result from the production of helium in the (n, α) in 10B.

- BF₃ effects

The thermal decomposition of NaBF₄ will yield BF₃ and NaF. Part of the BF₃ is removed with the coolant pump sweep gas stream because helium or some other gas is swept downshaft toward the pump bowl and in the process, this sweep gas carries the BF₃. To maintain the coolant composition, BF₃ has to be added to the coolant. By ensuring the ratio of NaF to NaBF₄, assurance is also provided that the freezing point for the coolant is not increased. This equilibrium between BF₃ and the coolant salt has to be determined under various normal operating conditions.

In case fuel reacts with the coolant, additional BF₃ is formed. The potential for the formation of BF₃ pressure surges under dynamic conditions has to be investigated.

Present designs of coolant pumps call for helium or some other inert gas to sweep downshaft toward the pump bowl. The He-BF₃ gas mixture cannot be continuously discharged from the plant because of the toxicity and chemical reactivity of BF₃. It needs to be determined whether this gas mixture could be fed into the gas cleanup portion of the off-gas system or any other portion of the off-gas system past the charcoal beds to convert the BF₃ into chemically benign constituents, or to design a separate BF₃ recirculation system.

¹ ORNL/TM-5759: "Distribution and Behavior of Tritium in the Coolant-Salt Technology Facility", by G.T. Mays et al. (April 1977)

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- solubility data for the dissolution of elemental hydrogen (tritium) in sodium fluoroborates

Tritium can exist in a variety of chemical forms in the coolant salt, among them the elemental form. The solubility of tritium in sodium fluoroborates has to be determined.

4.1.2.2 intermediate term

- gas-liquid equilibrium in the coolant pump bowl (effect of off-gas flow rate on equilibrium between gas and liquid)

This task was identified in response to observations in the CSTF. In experiment T4, the equilibrium conditions in the pump bowl between the gas and liquid were drastically changed when the off-gas flow rate was increased from 2 liters/minute to 4 liters/minute. The increase in purge gas flow by a factor 2 decreased the tritium concentration in the off-gas by significantly more than the expected factor 2. This suggests that at the increased purge gas flow rate the gas-stripping efficiency in the pump bowl is greatly reduced.

Note: lowering the purge gas flow rate in experiment T4 by a factor 2 showed the expected increase by a factor 2 in tritium concentration.

- coolant purification

Because of the high corrosivity of sodium fluoroborates in the presence of even small amounts of oxides (e.g., due to the addition of water) it is necessary for the salt to be as dry as practicable. Other impurities that cause corrosion are metallic ions, especially ferrous and ferric ions. They can be introduced when steel vessel equipment is used in the salt preparation.

The initial purity requirements need to be addressed with the vendor to determine the impurities in commercially available NaBF_4 and how they can be reduced by the vendor and how bench-scale purification processes can be scaled up.

During ABC system operation, moisture can enter the coolant through various means. For example, the helium cover gas contains always small amounts of moisture and in case of a steam leak (an anticipated operational occurrence) water can enter the coolant loop in large quantities. Practical on-line purification is necessary to keep corrosion at low levels to maintain the integrity of the metallic structures and to prevent entrained solids (such as Na_3CrF_6) from clogging coolant passages in steam generators and heat exchangers. This might require bypass facilities where one or

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more processing operations can be carried out.

In case of a reaction of coolant with fuel salt, fission products will also be present in the coolant and methods have to be employed to remove those fission products.

- irradiation stability

Radiation stability of the sodium fluoroborate coolant has been shown albeit in a single test run. Additional radiation testing of this material in a flowing system is necessary.

- freezing behavior

Even though the freezing point of sodium fluoroborate is much lower than for the MSRE coolant, it is not low enough to eliminate the need for preheating the feedwater. It needs to be determined which damage can be caused by coolant salt freezing caused by, for example, feedwater preheater failure.

In case sodium fluoroborate is accidentally released to the environment its freezing behavior has to be known as well as its reactions with materials such as concrete.

4.1.2.3 long-term

During the course of the ABC technology development program, the following activities have to be carried out during the later phases of the program:

- long-term test loop operation (similar to ORNL's Coolant Salt Technology Facility [CSTF])
- resolution of anomalies observed in the CSTF such as:
 - identification of the sink that required saturation before steady-state conditions can be established
 - determination of the existence of an extraneous source of hydrogen in the off-gas systems and its effect (if present) on the behavior and distribution of tritium

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- impact of coolant leaks

- steam generators

In case of a steam generator tube break coolant salt will be introduced into the steam-feedwater system. The pH of the water will be changed due to the formation of HF with fluoride-containing salts and the water will also contain concentrations of fluorides. The potential for stress corrosion cracking has to be investigated.

- cell atmosphere

Small amounts of coolant entering a cell will react with the moisture present in the atmosphere and generate acide vapor that could cause minor corrosion of the metal lining of the concrete walls and create (conventional) hazards due to toxicity of the vapors.

4.1.3 TRITIUM BEHAVIOR

4.1.3.1 near-term

- techniques for measured additions of tritium

For out-of-pile tritium experiments it is important to devise a method that permits the addition of measured quantities of tritium to the molten salt. Typically, tritium is contained with hydrogen in a Hastelloy N tube that is immersed in the molten salt stream and diffuses/permeates through the wall into the molten salt. The preparation of this gas mixture requires the mixing of tritium with hydrogen of known volumes and pressures, purification, and then a charging apparatus that loads the addition tube with the gas mixture. Modifications of such an arrangement are required for short term experiments in contrast to long-term experiments.

- tritium solubility in fuel salt

The difficulties in measuring solubilities of tritium are similar to those for measuring hydrogen solubilities and arise from the leakage and permeation of the hydrogen/tritium from the apparatus coupled with the very small quantities of tritium used in the measurements. Furthermore, in the presence of excess hydrogen, tritium permeates metals at a lower rate. However, tritium can be detected with greater

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sensitivity.

The solubility of tritium has to be determined with a without sparge gas bubbles present.

- tritium solubility in coolant salt

The same apparatus used for the tritium solubility measurements in fuel salt can be used for coolant salt. Because chemically bound tritium can be present in NaF-NaBF₄, details on these experiments can be developed only after the coolant chemistry task has been addressed.

- solubility of tritium fluoride (TF) in fuel salt

The problems associated with the measurement of the solubility of tritium trifluoride in fuel salt comes from the chemical interaction between PuF₃ and other constituents and the difficulty of obtaining saturation conditions. For a UF₄ system it had been proposed to use a HF-H₂ mixture of appropriately chosen composition to maintain constant oxidation conditions.

- tritium permeability for Hastelloy N

A modified Hastelloy N is the preferred structural material choice in a molten salt environment. It is known that tritium diffuses through Hastelloy N. For the design and interpretation of tritium experiments and the ABC system design, the tritium permeability for Hastelloy N has to be known.

- TF/T₂ ratio as a function of redox potential

It is likely that both T₂ and TF will be present in the fuel melt. The TF/T₂ ratio have to be determined as a function of the Pu³⁺/Pu⁴⁺ ratio.

- tritium removal efficiency from fuel salt via gas sparging

While the primary purpose of the gas sparging is the removal of noble gases from the fuel melt, this process also removes some of the tritium. The tritium removal efficiency from the fuel salt through gas sparging has to be determined.

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4.1.3.2 intermediate-term

- tritium fluoride corrosion behavior

Depending on the redox potential of the fuel salt, a certain amount of TF can be present in the fuel. This compound could react with metal surfaces in the blanket and/or, after removal from the blanket by sparge gas, with metals in the off-gas system. Furthermore, chemical reactions can occur in charcoal beds and limit the life of those beds. This corrosion behavior has to be characterized.

- tritium and tritium fluoride diffusivities in fuel and coolant salt

For the characterization of tritium and tritium fluoride behavior in molten salt systems, their diffusivities in fuel and coolant salt have to be measured.

- tritium permeation through other structural materials

In case stainless steel or another structural material is used in a molten salt environment, tritium permeation through this material has to be measured.

- equilibria between important species containing tritium in salt-gas systems

Methods need to be investigated that would prevent tritium from reaching the steam system of the reactor. The following chemical interactions are available for trapping tritium and have to be evaluated:

1. oxidation of molecular and atomic tritium
2. isotope exchange with hydrogen
3. sorption onto graphite or carbon

- isotopic exchange between tritium and chemically bound hydrogen

While the exchange of deuterium for protium in the species BF_3OH^- has not proved successful, isotopic exchange reactions might be possible in fluoroborate coolant. However, before any capsule experiments are designed, a thorough literature study should be performed to assess the feasibility of this process.

- tritium absorption on carbon in contact with coolant

Some of the tritium produced in the MSRE was found in the graphite moderator. How effective graphite can be in adsorbing tritium needs to be determined. The catalytic

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effect of, for example, thin nickel coatings has to be investigated.

- effect of tungsten, molybdenum and other coatings of structural metals on tritium confinement

It is conceivable that thin coatings of tungsten, molybdenum and other coatings increase tritium confinement. How effective those coatings are has to be determined.

4.1.3.3 long-term

- high-pressure, high-temperature steam loop operation to investigate oxide stability in steam system

It is conceivable that oxide films formed on potential steam generator materials in the presence of high-pressure steam retard tritium permeation. If it is shown that the oxide films are stable and that they retard tritium permeation, the role of steam chemistry on the oxide formation and effectiveness has to be determined.

4.2 GRAPHITE RESEARCH AND DEVELOPMENT

At this time, a decision has not been made in regard to the use of graphite in the ABC blanket region. Internally moderated, externally moderated and unmoderated ABC system configurations are being considered. Whatever the decision in regard to the use of graphite in the blanket region, it will affect and guide the work that has to be done under this activity.

Note: It was learned during the NP-MHTGR Program that there are no domestic suppliers left for nuclear-grade graphites.

Graphite in an ABC blanket has to perform two major functions, (1) moderate neutrons to achieve a well-thermalized neutron spectrum, and (2) protect metallic structures from high neutron fluences and the resulting neutron-induced damage. The graphite does not serve a structural purpose other than to define a blanket region and to support forces resulting from its own weight.

The primary objectives for this activity are the development and characterization of one or more graphite types that have the following design features in an ABC system environment:

- achievement of a long lifetime
- dimensional stability under irradiation
- structural integrity under irradiation
- non-penetrability by fuel salt
- suitable property data

4.2.1 near-term

- graphite stock development and qualification

As part of the development and qualification task the following topics need to be addressed:

- graphite coke source qualification
- basis for coke qualification
- basis for graphite specifications

Because of the shrinking domestic manufacturing base for nuclear-grade graphites it is important to identify and contact domestic and foreign vendors to ascertain which

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graphites they are capable of supplying, in which sizes, at what cost and schedule. Graphite types such as N3M from ORNL, type IG-110 from Japan's Toyo Tanso, type AFX-8Q from POCO and type 2020 from Stackpole have been identified in the past as promising candidates. Their availability needs to be confirmed.

The use of graphites developed under the HTGR programs needs to be evaluated including U.S., German and Japanese graphites.

The graphite evaluation will include measuring several physical properties such as pore size spectrum, gas permeability, crystalline structure and anisotropy, density, electrical and thermal conductivity and coefficient of thermal expansion.

The most attractive materials will be irradiated and the dimensional changes measured as a function of temperature and fluence.

- special graphite treatment for property improvements

Special graphite treatments need to be investigated for the improvement of properties such as dimensional stability, reduced shrinkage and swelling, and improved structural integrity.

- sealing graphite to reduce graphite permeability

The objective for this activity is to develop a process for decreasing the permeability of graphite in the blanket region with respect to ^{135}Xe to acceptably low levels (about $10^{-8} \text{ cm}^2/\text{sec}$). The following three techniques have been proposed in the past:

- surface coating of graphite with pyrocarbon
- impregnation with carbon
- impregnation with salt

These techniques have not been very successful in the past and need to be carefully evaluated before they are pursued further.

- physical properties

To properly characterize the graphites the following physical properties have to be measured:

- thermal property data (irradiated and unirradiated graphite)
- irradiation-induced dimensional change data

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- irradiation-induced creep data

4.2.2 intermediate term

- basic graphite studies

The objective for these basic graphite studies is to determine basic radiation damage mechanisms in graphite. This can be achieved by using single crystals to determine the types and numbers of defects produced by electron irradiation. Of particular interest are:

- irradiation damage mechanisms
- annealing kinetics

Next, polycrystalline graphite will be irradiated to assess the presence of grain boundaries which complicate irradiation damage mechanisms over those in single crystals. The importance of grain size and angular mismatch between adjacent grains has to be determined.

Part of the basic studies are also the following investigations:

- effect of accelerated irradiation

Material irradiations are often performed in high flux reactors where the equivalent of a lifetime fluence can be accumulated in a year or less of irradiation. It needs to be determined whether the balance is disturbed between changes in structure caused by the bombardment with neutrons (i.e., related to the neutron flux level) and annealing effects that might be related to real time. Both real time and accelerated irradiation experiments have to be performed.

- extrapolation of graphite performance data obtained from small sample experiments to the design of large graphite structures

Irradiation samples are mostly small pieces of graphite that are exposed to neutron flux and temperatures distributions that are flat over the irradiation sample. In reality, graphite structures are exposed to flux and temperature gradients, and creep effects, for example, affects graphite properties and graphite behavior. It needs to be determine how small sample irradiation experience can be translated to large graphite component design.

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- very high energy neutron damage

In a thermal reactor, most of the neutrons are at low energies (eV to keV range). In an ABC system, the majority of the target neutrons is in the upper keV and lower MeV range. The damage mechanisms for these very high energy neutrons needs to be determined.

- graphite irradiation in a molten salt environment

Graphite samples are typically irradiated in an inert environment rather than in a molten salt environment. Any effects of fuel entrainment leading to local hot spots that affect graphite swelling, shrinkage and cracking, need to be assessed.

- physical properties for unirradiated and irradiated graphite

For the reference graphite type(s) the following physical properties have to be determined:

- fatigue data
 - multiaxial strength data
 - mechanical and thermal property data
 - permanent reflector graphite property data
- fracture mechanics data

A data base is needed to define the critical stress intensity factors and strain energy release rates for crack initiation, stable crack growth, and crack arrest for the reference graphite type(s).

- graphite destructive and non-destructive examination data

Data are needed to validate NDE techniques and write material procurement specifications for the procurement of graphite for blanket components.

- corrosion data

Graphite components may be corroded by impurities in the fuel salt, coolant salt, reactions with fission products, metallic corrosion products, and other reaction products. It is unknown whether graphite shrinkage or swelling affects graphite corrosion. While graphite corrosion is not expected to be a concern for short graphite

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residence times, correlations have to be developed that can describe the long-term corrosion behavior of graphite.

4.2.3 long-term

- graphite-molten salt interactions studies

Interaction studies between graphite and molten salt have to be carried out to characterize the long-term behavior of molten salt - graphite systems.

- corrosion data

The corrosion effects on graphite design properties have to be determined.

Corrosion data for methods validation have to be obtained.

4.3 METALS RESEARCH AND DEVELOPMENT

The technology development tasks listed below are based on the assumption that a modified Hastelloy N is the prime candidate structural material. Should a stainless steel (e.g., SS304 or HT-9) become the prime candidate this technology development program would have to be revised.

Note: Iron-based alloys are expected to be more resistant to Te embrittlement and have lower (n,a) cross sections. However, their use is limited to temperatures of less than 600 - 650°C which is well below the normal operating condition outlet temperature for the salt. Furthermore, the impact of salt impurities and fission products and the salt redox potential on the performance of those alloys is largely unknown.

4.3.1 near-term

- assessment of applicability of existing modified Hastelloy N information to ABC system

Two types of damage were found in the Hastelloy N used in the MSRE, (1) embrittlement by helium produced in ^{10}B and in a two-step reaction in nickel, and (2) embrittlement by tellurium that diffused a short distance into the metal along the grain boundaries. The proposed remedies at that time were the addition of 2% titanium to enhance the resistance to irradiation embrittlement, and to add up to 2% niobium to provide resistance to intergranular attack by tellurium. However, it was found that the presence of titanium negated the beneficial effects of niobium. One irradiation experiment was carried out in 1976 that showed adequate resistance to irradiation embrittlement for niobium-modified Hastelloy N without titanium.

Experiments were carried out to determine the effects of oxidation states on the tendency for cracks to be formed in the presence of tellurium in the salt. It was found that in a reducing environment very little cracking occurred suggesting that a reactor could be operated in a chemical regime where tellurium might not even embrittle standard Hastelloy N.

The existing data base on Hastelloy N and its modifications need to be assessed in regard to their applicability to the ABC system which differs in many respects from the MSRE environment, especially in regard to the use of plutonium as fuel.

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- modification of Hastelloy N

Upon conclusion of the data base evaluation, Hastelloy N has to be modified to achieve the desired level of resistance to helium and tellurium embrittlement. Modified alloys need to be prepared for irradiation, fabrication studies, corrosion studies, tellurium exposure and mechanical property determination.

- material and irradiation properties of modified Hastelloy N candidates

The modified Hastelloy N candidates have to be characterized in regard to material and irradiation properties.

- material specifications for modified Hastelloy N

In preparation of the testing of commercial heats of modified Hastelloy N, material specifications have to be developed.

- vendor qualification

A vendor has to be identified that can supply modified Hastelloy N at the desired quality, quantity and in suitable structural shapes.

- irradiation of commercial heats and material characterization

Commercially obtained heats of modified Hastelloy N will be tested both in-pile and out-of-pile and fully characterized.

- material selection for auxiliary systems (e.g., coolant salt-steam system, off-gas system, processing systems)

Hastelloy N seems fully qualified for application in auxiliary systems. The use of less expensive materials such as stainless steel has to be evaluated.

- tests on double-walled piping

Among the structural shapes that have to be tested is double-walled piping made of Hastelloy N or other suitable materials.

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- basic metals studies

Basic metals studies have to be carried out with focus on:

- physical properties
- irradiation damage mechanism
 - o tellurium embrittlement

This task is directed toward an understanding of the interaction of tellurium, Hastelloy N and salt to define the phenomena that lead to attack in standard Hastelloy N and to determine how such an attack is prevented in potential materials of construction.

The interaction of Hastelloy N with tellurium vapor has to be investigated and the reacted metal will be used to identify reaction products, evaluate effects on mechanical properties, and determine reaction kinetics.

Even though the presence of salt is not likely to influence the interaction of Hastelloy N and tellurium, experiments are required to confirm this.

- o helium production (boron, nickel)

Various heats of modified Hastelloy N will be irradiated to study their embrittlement during thermal neutron irradiation. Irradiation temperature is an important variable. Post-irradiation creep tests and tube burst tests will be performed; the effect of thermal annealing will be assessed; the microstructure of the modified Hastelloy N has to be determined.

- o other mechanisms

While the embrittlement of standard Hastelloy N is mainly due to the presence of tellurium, assurance has to be provided that other (fission product) elements do not have similar adverse effects which could be of significance for the ABC salt with its high fission product content.

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- properties most affected by irradiation (damage)

Small heats of Hastelloy N containing small amounts of fission products (e.g., Se, Te, Sr, Tc, Ru, Sn, and Sb) have to be prepared and tested under creep conditions to determine how these elements influence the tendency for crack formation.

Tube burst tests with standard and modified Hastelloy N have to be run in a salt-tellurium environment to show the influence of tellurium on rupture life of a material.

Strain cycle tests have to be performed with alternating strains imposed on a tubular sample while the outside surface of the sample is exposed to salt containing tellurium. The time required for cracks to propagate through the tube wall will be determined as a function of alloy composition, concentration of tellurium, strain limits and temperature.

- effect of niobium and titanium (and other) additions

Both out-of-pile and in-pile experiments are required to assess the effects of adding niobium and titanium (and other elements) to Hastelloy N.

- effect of redox potential on cracking

It was learned for uranium systems, that cracking is far less frequent in a reducing than in an oxidizing environment. The cracking behavior has to be similarly assessed for a plutonium system.

In many instances a good understanding of damage mechanisms has already been developed for uranium systems with low levels of fission products present. It has to be determined whether this experience is applicable also to plutonium fuels with very large concentrations of fission products present.

- evaluation of data base for stainless steel applications in molten salt environments

Initial results on the use of stainless steels instead of Hastelloy N (or modified Hastelloy N) have been encouraging because of the greatly reduced embrittlement problems that were encountered with the nickel-based alloys. The data base on stainless steel use in molten salt environments (fuel salt, coolant salt) has to be evaluated. Exploratory experiments with stainless steels in molten salt environments

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have to be conducted.

Areas of concern are metal corrosion and change in mechanical properties due to corrosion and irradiation effects.

- material specifications for stainless steel candidates

Preliminary material specifications for stainless steel candidates have to be developed.

4.3.2 intermediate term

- basic metals studies

The basic metals studies have to be continued with attention given to:

- irradiation damage mechanism
 - properties most affected by irradiation
 - corrosion loops
- Both natural convection and forced flow corrosion loops have to be operated. In designing these loops consideration has to be given to:
- in-pile
 - out-of-pile
 - environments:
 - fuel salt
 - coolant salt
 - other
 - effect of impurities in salt on metals corrosion
 - effect of corrosion on physical properties

- fabricability studies

For promising candidates fabricability studies have to be performed with focus on:

- welding, joining
- component fabrication

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- determination of limiting conditions

Based on the development and characterization of the reference structural alloy(s) limiting conditions have to be identified for normal and off-normal operating conditions. Examples are limiting fluences, temperatures, pressures, temperature gradients, heatup and cooldown rates, etc.

- ASME Boiler Code case

The data obtained will be analyzed and expressed in analytical format suitable for use by designers. Model tests will be carried out to develop high-temperature design methods. These methods will most likely be patterned after the methods developed for type 304 and 316 stainless steel. Among the methods are those for the calculation of multiaxial stresses, creep behavior under those stresses, and methods for summing up creep and fatigue damage.

The data and design methods will be used for preparing a high-temperature ASME Pressure Vessel Code Case for submission. This case will officially establish the design rules as being acceptable for use of modified Hastelloy N in nuclear service.

- material optimization

Work will continue on the development of less expensive structural materials (such as stainless steels).

- effect of irradiation acceleration

Real time irradiation results have to be compared with accelerated irradiation results to determine the validity and applicability of accelerated irradiation results.

- interpretation of small sample irradiation tests for application to large structures

The issue needs to be addressed how the results from small sample experiments can be used in the design of large structures (similar to the graphite work described in section 4.2.2).

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4.3.3 long-term

Work will be completed in the following areas:

- corrosion loops
- mockup tests
- completion of data bases
- qualification of reference alloys

4.4 MONITORING AND SURVEILLANCE RESEARCH AND DEVELOPMENT

To fully utilize the unique features of molten salt systems and ensure safe and reliable operation it is necessary to maintain adequate surveillance and control of the salt chemistry by monitoring the composition in the various loops and material streams in the ABC system.

4.4.1 near-term

- begin with selection and/or development of methods and devices to be used in near-term experiments for monitoring and surveillance in support of blanket operation and the experiments performed under the ABC Technology Development Program

Methods and devices for monitoring and surveillance are not only required for proof-of-principle and demonstration experiments but also for loop operation (e.g., corrosion loops) and irradiation capsule operation. To avoid delays in the commencement of these experiments existing and proven methods should be used first.

The following parameters should be continuously monitored and appropriate devices and methods have to be adapted and/or developed:

- redox conditions of the fuel (only in-line)

The fuel salt redox potential has a significant influence on the chemical and perhaps even physical properties of the fuel. It determines the corrosiveness of the fuel with respect to structural materials, influences the disposition of certain fission products and tritium, and may affect surface tension and wettability of salt melts. For the measurement of U^{3+}/U^{4+} ratios, the voltammetric determination of this ratio has been successfully applied to in-line measurements. No electroanalytical studies for plutonium were conducted under the MSBR Program. Plutonium is expected to be present predominantly in the trivalent state. If another soluble oxidation state can be generated reversibly within the solvent electrolysis limits, voltammetric methods should be applicable also to plutonium.

The applicability of voltammetric methods to plutonium fuel has to be assessed and alternate methods have to be investigated should voltammetric methods be either not sufficiently accurate or not feasible.

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- plutonium and other transuranics concentrations in fuel and processing/waste streams

The electrochemistry of plutonium and higher actinides has to be studied in ABC-composition melts. There is no technology base that had been developed under past molten salt reactor programs. Spectrophotometrical methods for the determination of plutonium have to be studied. Based on MSBR experience with these methods it was concluded that absorptiometric methods should be feasible for the precise measurement of plutonium concentrations.

- rare earth fission product concentration

The rare earth fission product concentration has to be accurately monitored because the solubility of these trifluorides is limited to about 1 mole %.

- corrosion product ions (particularly Cr^{2+} and Fe^{2+}) in the fuel, coolant and some process streams

Methods will be required for the continuous determination of ppm concentrations of corrosion product ions, particularly chromous ions, in fuel streams to monitor corrosion in the ABC fuel system. Complete electrochemical characterizations have been carried out in MSRE solvents for Fe^{2+} , Cr^{2+} , Ni^{2+} , and Ti^{3+} and Ti^{4+} . The electroanalytical chemistry of these ions has to be characterized in ABC fuel salt.

In-line methods for the measurement of corrosion product ions, particularly Cr^{3+} , are needed to monitor the corrosion within the coolant system. The measurement of chromic ions in fluoroborates will be of great importance because of the low solubility of CrF_3 in this coolant. The concentration of Cr^{3+} must be limited to a value that does not permit the deposition of Na_3CrF_6 at the coolest region of the heat exchanger.

- valence state and concentration of some fission products (e.g., Te, Nb, Zr, Nd, and Eu)

It would be desirable to know the valence state and concentration of some important fission products to be able to continually monitor the fission product chemistry in the fuel melt. This is particularly important because of the very high fission product concentrations expected later in fuel life.

In addition it needs to be known whether the presence of fission products can interfere with electroanalytical methods used for surveillance. For example, noble fission products could deposit on electrodes and alter their characteristics or possibly

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cause electrode damage.

- O^{2-} content of fuel

The in-line determination of oxide is of prime importance because oxide could cause precipitation of plutonium from the fuel. The oxide tolerance of the fuel has to be accurately known. Electroanalytical methods at oxide concentrations below the tolerance level of the fuel as well as for higher oxide concentrations have to be developed.

- free and combined hydrogen and tritium content in fuel and coolant

One means of controlling tritium emissions from the ABC system is to operate, if possible, the ABC system at redox conditions that will retain a major fraction of the tritium in a combined form and prevent it from diffusing into the coolant. Methods have to be developed to measure the ratio of free to combined hydrogen and tritium in fuel and coolant salt.

- oxygenated, protonated and tritiated species in fuel and particularly in the coolant

This activity aims at a monitoring of the trapping of tritium in the fuel and particularly the coolant. After the trapping mechanism has been fully understood, methods for the determination of oxygenated, protonated and tritiated species in fuel and coolant have to be developed and demonstrated. It has been shown that molecular spectroscopy, using infrared or near-infrared measurements, can be used to measure complexes of anions including hydroxides with di- and trivalent cations at ppb concentrations.

- gas stream monitoring (fuel cover gas, coolant cover gas, off-gas system, process gas streams) shall include: permanent gas contaminants, moisture, tritium, hydrocarbon, fission product gases, BF_3 , HF, hydrogen, hydrolysis products, F_2 , HT, HTO (trace constituents can only be measured in-line).

Many commercial instruments are available that can be used directly or with modifications for these determinations. The first activity under this task is to survey the market to determine the available instruments and the need for modifications. The development of new instruments should be the last resort.

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- employ methods and devices in small-scale experiments

Electroanalytical and spectrophotometric methods as well as analysis by chemical methods need to be employed in small-scale experiments to test those methods.

- determine which of those control parameters have to be tracked in-line and off-line

A determination has to be made in regard to the parameters that have to be monitored continuously, whether they have to be tracked in-line or off-line. It is conceivable that some parameters can be monitored initially in-line but that either because of deterioration of the measuring devices or because of interferences in-line measurements are not possible anymore.

- design of in-line analytical test facility

A facility will be designed that will demonstrate the capabilities of various in-line methods in terms of reliability, reproducibility, time constants, and the effect of operating parameters such as temperature, flow rate, etc.

4.4.2 intermediate term

- evaluation of the methods and devices for monitoring and surveillance, considering reliability, independence, diversity, and redundancy as criteria

The operation of the in-line analytical facility will address reliability, independence, diversity and required redundancy of the methods and devices required for monitoring and surveillance and help in the development of quality control procedures..

- determine achievable accuracies

The accuracies of the methods and devices for monitoring and surveillance have to be determined.

- develop calibration standards

Before the various devices can be used calibration standards have to be developed.

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- develop quality assurance and control procedures

QA and QC procedures have to be developed for both the devices to be used and how they have to be used.

- out-of-pile and in-pile testing of the methods and devices

Out-of-pile testing in the in-line analytical test facility has to be complemented by in-pile testing.

- employ devices and methods in large-scale experiments

Monitoring and surveillance devices and methods have to be installed in the proof-of-principle and demonstration experiments.

4.4.3 long-term

- evaluate monitoring and surveillance performance based on experience gained in monitoring and surveillance operations in large-scale experiments

The experience gained in operating large-scale experiments (proof-of-principle and demonstration experiments) has to be evaluated for use in a prototype plant.

- select reference methods and devices for monitoring and surveillance

Based on the evaluation of the large-scale experiments, criteria for the selection of reference methods and devices for monitoring and surveillance have to be developed and applied.

- develop techniques for maintenance, repair and replacement of monitoring and surveillance devices

One of the criteria for the suitability of in-line devices will be the ease with which they can be replaced. Techniques have to be developed for maintenance, repair and replacement of those devices.

4.5 THERMAL HYDRAULICS DESIGN DEVELOPMENT

To a large extent, the thermal-hydraulics design development is driven by basic design choices (e.g., internally or externally moderated blanket) and design detail (inlet and outlet plenum designs). The following list of activities is more generic than design-data-needs driven. Many of the parameters addressed below lose greatly in significance in case an unmoderated or externally moderated blanket is selected.

4.5.1 near-term

The following flow parameters will be measured in heat transfer loops:

- thermal-hydraulics data

Basic physical properties need to be determined such as thermal conductivity, specific heat, density, as well as friction factor correlations and Nusselt number correlations, over the appropriate range of Reynolds numbers for fuel and coolant salts.

- pressure drop data

Loss coefficients for different sections of the primary and secondary circuits need to be determined, as a function of Reynolds number.

4.5.2 intermediate term

The following flow parameters will be measured in heat transfer loops:

- flow mixing data

Flow mixing in particular in the outlet plenum has to be investigated. Parameters of interest are pressure drop, flow distribution, hot/cold attenuation data as a function of Reynolds number and location of hot/cold streaks.

- cross flow data

In case stacked HTGR-type graphite blocks are used in the blanket region as moderator containing coolant holes, cross flow data have to be determined, among them loss coefficients as a function of expected cross flow pressure differentials, cross flow gaps, and flow bypass gaps.

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- cold bypass flow

The flow in the gaps between the graphite columns has to be characterized.

In a vibration test facility, the following data will be determined experimentally:

- graphite column vibration data

Data have to be produced to show that blanket columns have no significant flow-induced vibrations under any blanket operating conditions. The vibratory characteristics of the blanket columns has to be determined with and without cross flow.

- control rod vibration data

In case control rods are used, data have to be obtained to confirm that the control rods have no significant flow-induced vibrations under any blanket operating conditions. The vibratory characteristics of the control rods has to be determined with and without cross flow.

- blanket fluctuation data due to mechanical/thermal movements

Data have to be obtained to show that the graphite block configuration will be stable under expected heating, flow and pressure drop conditions.

4.5.3 long-term

The long-term activities under this task depend on design selections made.

4.6 METHODS VALIDATION

4.6.1 near-term

No specific experimental activities are planned as near-term activities. Validation plans need to be developed and existing data bases need to be evaluated with emphasis on assurance of quality of the data (i.e., they must meet applicable standards), data accuracies and applicability.

4.6.2 intermediate term

The proof of principle experiment plays a very important role for methods validation in the following areas:

- neutron physics data at elevated temperatures
- shielding
- thermal hydraulics
- transients and accidents
- graphite and metal performance
- mechanical performance

4.6.3 long-term

Additional validation data can be obtained from the demonstration experiment and the prototype operation.

4.7 COMPONENTS, SYSTEMS AND STRUCTURES TECHNOLOGY DEVELOPMENT

Any technology development work on ABC systems, structures, and components is design-specific. In the absence of any design specificity at this time, the following is a list of topics that need to be addressed.

- 4.7.1 fuel and coolant salt
- 4.7.2 reactivity control
- 4.7.3 molten salt pumps
- 4.7.4 molten salt valves
- 4.7.5 vessels
- 4.7.6 graphite structures
- 4.7.7 salt-salt heat exchangers
- 4.7.8 active and passive decay heat removal systems
- 4.7.9 steam technology
- 4.7.10 cover- and off-gas systems
- 4.7.11 purifications systems
- 4.7.12 salt processing systems
- 4.7.13 instrumentation and control
- 4.7.14 cell heating and insulation
- 4.7.15 containment
- 4.7.16 drainage and storage
- 4.7.17 remote maintenance and repair
- 4.7.18 component and system replacement

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4.7.6 Graphite Structures

4.7.6.1 near-term

No work is planned in the near-term.

4.7.6.2 intermediate term

Experimental work is proposed to address the following topics:

1. graphite block failure mode data
2. graphite blanket support strength data
3. graphite structure dynamic strength data
4. seismic loads on graphite structure

In the absence of a blanket design a description of required experimental work has to be deferred.

4.7.6.3 long-term

TBD

4.7.11 salt processing systems

4.7.11.1 near-term

- electrolytic deposition of semi-noble metals (Zn, Ga, Ge, As)
- electrolytic deposition of noble metals (Nb, Mo, Tc, Ru, Rh, Pd, Ag, Cd, In, Sn, Sb)

4.7.11.2 intermediate-term

TBD

4.7.11.3 long-term

TBD

4.8 LARGE SCALE EXPERIMENTS

Large-scale experiments play a major role in the development and demonstration of the technology base for the ABC system. They are design-specific and, at this time, only broad outlines for those experiments can be given due to the absence of a pre-conceptual design.

To better visualize the duration of the technology development program, it is assumed that the near-term phase will last about [3] years, the intermediate-term phase about [4] years, and the long-term phase about [3+] years.

1. major test loops

Among the major test loops is a (low pressure) salt technology test facility that permits experiments on tritium behavior, corrosion behavior, thermal-hydraulics parameters, monitoring and surveillance devices and methods, gas sparging, off-gas system performance, as well as basic filling and draining procedures.

For the investigation of steam-salt interactions a high-pressure steam loop is required.

Construction of the test loops and operation of some of them are near-term objectives.

2. plant protection and control simulation facility

The complexities in the plant protection and control system have to be tested in a simulation facility that allows for a simulation of accelerator, blanket and auxiliary system controls, the definition of set points, a simulation of both normal operation and upset events, and the monitoring and surveillance system.

Design, construction, and operation of this facility are intermediate-term objectives.

3. in-line analytical test facility

This facility will demonstrate the capabilities of various in-line monitoring and surveillance methods in terms of reliability, reproducibility, time constants, and the effect of operating parameters such as temperature, flow rate, etc.

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Design of this facility is a near-term objective. Construction is an intermediate objective, and operation an intermediate- to long-term objective.

4. proof-of-principle experiment

In a proof-of-principle experiment the accelerator will be coupled with a blanket with its associated heating, cooling and control systems, purification systems and a gas sparging system. Such an experiment will start without any fissile material in the blanket. It has to demonstrate the following:

- molten salt technology whose development was interrupted in the mid-1970's has been fully recovered
- the ABC target assembly design is technically viable
- an accelerator can be coupled to a subcritical blanket and produce fission heat in a controllable manner, using a separately tested target design
- use of accelerator control system to control the ABC blanket operation can be accomplished
- the heat produced in the target and blanket can be rejected through heat exchanger - pump - piping systems in a controlled manner
- effective gas sparging for an internally moderated configuration can be demonstrated
- salt chemistry control can be demonstrated
- analytical and diagnostic techniques can be demonstrated
- the principles for remote maintenance, repair, and replacements can be demonstrated

System thermal-hydraulics can be studied under both forced and natural convection flow conditions. Initially, the proof of principle experiment does not use fissile material. However, a still to be determined amount of fissile material will be added later to achieve a predetermined level of subcriticality and continued energy production. At that time, a series of reactor physics and thermal-hydraulics experiments can be carried out that will provide methods validation data.

The proof-of-principle experiments will be completed by tying this now subcritical facility at a specific eigenvalue to the accelerator to produce nuclear heat and demonstrate that stable, reliable and continuous heat production can be achieved.

It might be possible to use the same facility for the proof-of-principle experiment and the demonstration experiment. However, the issues of configuration prototypicality and containment need to be addressed.

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Design and construction for this experiment begin during the near-term phase, operation begins late in the near-term phase and extends into the intermediate-term phase.

5. demonstration experiment

The demonstration experiment goes well beyond the proof-of-principle experiment in that its purpose is the demonstration of the capability of the ABC to safely, reliably, predictably and economically burn weapons grade plutonium and produce electricity. This experiment has to demonstrate successful systems integration, licensability for the prototype, remote maintenance, repair and replacements. It might be possible to base this experiment on the proof-of-principle experiment by using all or part of its equipment. For this experiment, components such as pumps, valves, IHX, steam generator, processing facilities, etc. do not have to be of prototypical size but their design features should closely resemble those of the prototype components.

Design work for this plant will commence late during the near-term phase and be completed during the intermediate term phase. Construction will commence late during the intermediate phase and be completed early in the long-term phase. Operation will start during the long-term phase.

6. prototype operation

The prototype experiment is the operation of a single ABC module demonstrating that such a plant can be built on schedule and within the allowed budget; that it can be operated reliably and safely and meets the mission objectives. Over a period of five years, all features of the ABC system, such as off-gas treatment, fission product removal, fuel and coolant drainage and storage, fuel and coolant processing, plant protection and control systems, tritium handling, remote maintenance, repair and replacement will be demonstrated. Furthermore, the prototype operation is expected to confirm the projected economics of the operation.

The prototype design will begin and be completed during the long-term phase with begin of construction coinciding with the end of the technology development activities.

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Table 4-1: Proposed Schedule for Large-Scale Experiments

YEAR	1	2	3	4	5	6	7	8	9	10	11 -15
CORROSION LOOPS											
DESIGN											
CONSTRUCTION											
OPERATION											
ANT PROTECTION SIMULATION FACILITY											
DESIGN											
CONSTRUCTION											
OPERATION											
IN-LINE ANALYTICAL TEST FACILITY											
DESIGN											
CONSTRUCTION											
OPERATION											
PROOF-OF-PRINCIPLE EXPERIMENT											
DESIGN											
CONSTRUCTION											
OPERATION											
MONSTRATION EXPERIMENT											
DESIGN											
CONSTRUCTION											
OPERATION											
PROTOTYPE											
DESIGN											
CONSTRUCTION											
OPERATION											

CHAPTER 5.0

PRIORITIZATION OF TECHNOLOGY DEVELOPMENT TASKS

The ABC Technology Development Program identifies near-term, intermediate-term and long-term technology development needs. Because of budgetary constraints, not all technology development tasks can be carried out simultaneously and the ABC project has to prioritize the development tasks. Of particular importance is the prioritization of the short-term technology development needs because they have to be dealt with immediately.

For the ABC project there are two major objectives for the near-term technology development work:

1. development of a conceptual ABC design
2. perform proof of principle experiment

To determine the significance of technology development tasks for the *development of a conceptual design*, the following questions have to be asked:

1. does this technology development task support basic ABC system design selections, such as:
 - fuel salt
 - coolant salt
 - structural material
 - operating parameters
 - blanket configuration
 - basic energy conversion system (e.g., steam vs gas-turbine cycle, adding a tertiary loop)
 - assurance of prompt negative temperature coefficient
 - defense in depth approach

Any task related to the selection of these basic design selections has a high priority. Without being able to make those selections, the ABC system design cannot be developed.

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2. does this technology development task support conceptual design optimization such as:

- blanket power level
- blanket power density
- blanket dimensions
- refinement of operating parameters
- fuel life extension
- reduction in the number of auxiliary systems through shared functions
- reduction in external fuel inventory

Design refinement and optimization are important tasks that have to be addressed all through the design process. However, they are of lower priority than the selection of basic design features. In a constrained budget environment these tasks will have a medium priority.

3. how much does it cost to perform this development task?

- cost vs benefits
- affordability
- cost and impact of delay

There are limits to the extent to which a design can be further refined and optimized. There is a cost attached to design refinement and optimization and often more insight can be gained by operating, for example, a proof-of-principle experiment rather than conducting a broad-based technology program, especially when data have to be obtained in a molten salt environment.

Under constrained budget conditions, experience has shown that desired experiments could not be conducted because of funding shortfalls, and use had to be made of fallback positions. The issue of affordability has to be addressed in relation to other experiments of similar importance.

While the immediate cost is an important consideration in deciding priorities, there are also long-term considerations of importance. For example, work on a steam-generator experiment has to commence early and will be very expensive. However, any delay in this work will cause a delay for the overall project that will result in overall cost increases. Similar considerations apply to all component tests that have long lead

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times.

The conduct of the *proof of principle experiment* is important because it establishes both technical feasibility and technical viability of the accelerator driven transmutation technologies, in particular for plutonium disposition. To establish priorities, the following questions have to be asked:

1. does this development task support the design of the proof-of-principle experiment facility? Examples are the following:
 - necessary LAMPF modifications
 - selection of basic design features and operating parameters
 - selection of fuel and coolant salt, structural material
 - selection of basic experiment facility layout
 - selection of blanket configuration
 - target design support
 - assurance of target performance
 - instrumentation
 - enclosure/containment
 - systems and components (e.g., specifications, vendor qualification)
 - maintenance, repair and replacements

All work in support of the design of this experiment has a high priority. It is the most important experiment in support of the ABC design that will not only provide a confirmation of the technical viability of the ABC concept but also a wealth of technology data.

2. does this technology development task support the planned experiments for this facility? Examples are the following:
 - adaptation of the LAMPF facility for this experiment
 - bubble generator and separator
 - experiment monitoring and surveillance
 - transmutation of Pu

The purpose of this facility is to conduct experiments. Work related to those experiments has, therefore, a high priority.

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3. does this technology development task aim at the optimization of the proof of principle experiment facility? Examples are the following:

- size of facility
- reduction in number of auxiliary systems
- refinement of operating parameters
- operating flexibility

A certain level of optimization work for this facility is needed but more important is operating flexibility. The priority for this work lies between high and medium.

4. how much does it cost to conduct this technology development task?

- cost vs benefits
- affordability
- cost of delay

The proof of principle experiment is needed to establish a solid footing for the ABC system design. To ensure the viability of the ABC project, this experiment has to be done at an early time. The benefits are very high, and so is the cost for a delay. Affordability is not a criterion in this prioritization because without this demonstration of concept viability it will be very difficult to conclude successfully the ABC project.

The Tables 1 and 2 show the priorities for the eight different technology tasks for the development of the conceptual design and the proof of principle experiment, respectively.

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Development of Conceptual Design

Development Task	Priority		
	High	Me- dium	Low
1. Chemical Research and Development			
1.1 Fuel and Fission Product Chemistry	X		
1.2 Coolant Chemistry	X		
1.3 Tritium Behavior		X	
2. Graphite Research and Development	(X)*		(X)*
3. Metals Research and Development	X		
4. Monitoring and Surveillance Research and Development	X		
5. Thermal-Hydraulics Design Development	X		
6. Methods Validation			X
7. Components, Systems, and Structures Technology Development			X
8. Large Scale Experiments	X		

* whether a "high" or "low" priority has to be assigned depends on the design concept chose (i.e., an internally moderated configuration requires graphite development work at a high priority; externally or unmoderated configurations require graphite development work at a low priority).

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Proof-of-Principle-Experiment

Development Task	Priority		
	High	Medium	Low
1. Chemical Research and Development			
1.1 Fuel and Fission Product Chemistry	X		
1.2 Coolant Chemistry	X		
1.3 Tritium Behavior		X	
2. Graphite Research and Development			X
3. Metals Research and Development		X	
4. Monitoring and Surveillance Research and Development	X		
5. Thermal-Hydraulics Design Development		X	
6. Methods Validation			X
7. Components, Systems, and Structures Technology Development			X
8. Large Scale Experiments	X		