Appendix 6.0
Molten Salt Reactor

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Acronyms

AFCI	Advanced Fuel Cycle Initiative
AHTR	Advanced High Temperature Reactor
AMSR	Advanced Molten Salt Reactor
DOE	Department of Energy
FY	fiscal year
GIF	Generation IV International Forum
I-NERI	International Nuclear Energy Research Initiative
kW_e	kilowatt electricity
LS-VHTR	liquid-salt-cooled VHTR
LWR	light water reactor
MSBR	Molten Salt Breeder Reactor
MSR	molten salt reactor
MW_e	megawatt electric
MW_t	megawatt thermal
NE	DOE Office of Nuclear Energy, Science, and Technology
NGNP	Next Generation Nuclear Plant
NHI	National Hydrogen Initiative
ORNL	Oak Ridge National Laboratory
R&D	research and development
SNF	spent nuclear fuel
VHTR	Very-High-Temperature Reactor

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A6.1 INTRODUCTION AND BACKGROUND

A6.1.1 System Description

Molten Salt Reactors (MSRs) are liquid-fueled reactors that can be used for production of electricity, actinide burning, production of hydrogen, and production of fissile fuels (Figure A6.1). Electricity production and waste burndown are envisioned as the primary missions for the MSR. Fissile, fertile, and fission isotopes are dissolved in a high-temperature molten fluoride salt with a very high boiling point (1,400°C) that is both the reactor fuel and the coolant. The near-atmospheric-pressure molten fuel salt flows through the reactor core that contains graphite moderator. In the core, fission occurs within the flowing fuel salt that is heated to ~700ºC, which then flows into a primary heat exchanger where the heat is transferred to a secondary molten salt coolant. The fuel salt then flows back to the reactor core. The clean salt in the secondary heat transport system transfers the heat from the primary heat exchanger to a high-temperature Brayton cycle that converts the heat to electricity. The Brayton cycle (with or without steam bottoming cycle) may use either nitrogen or helium as a working gas.

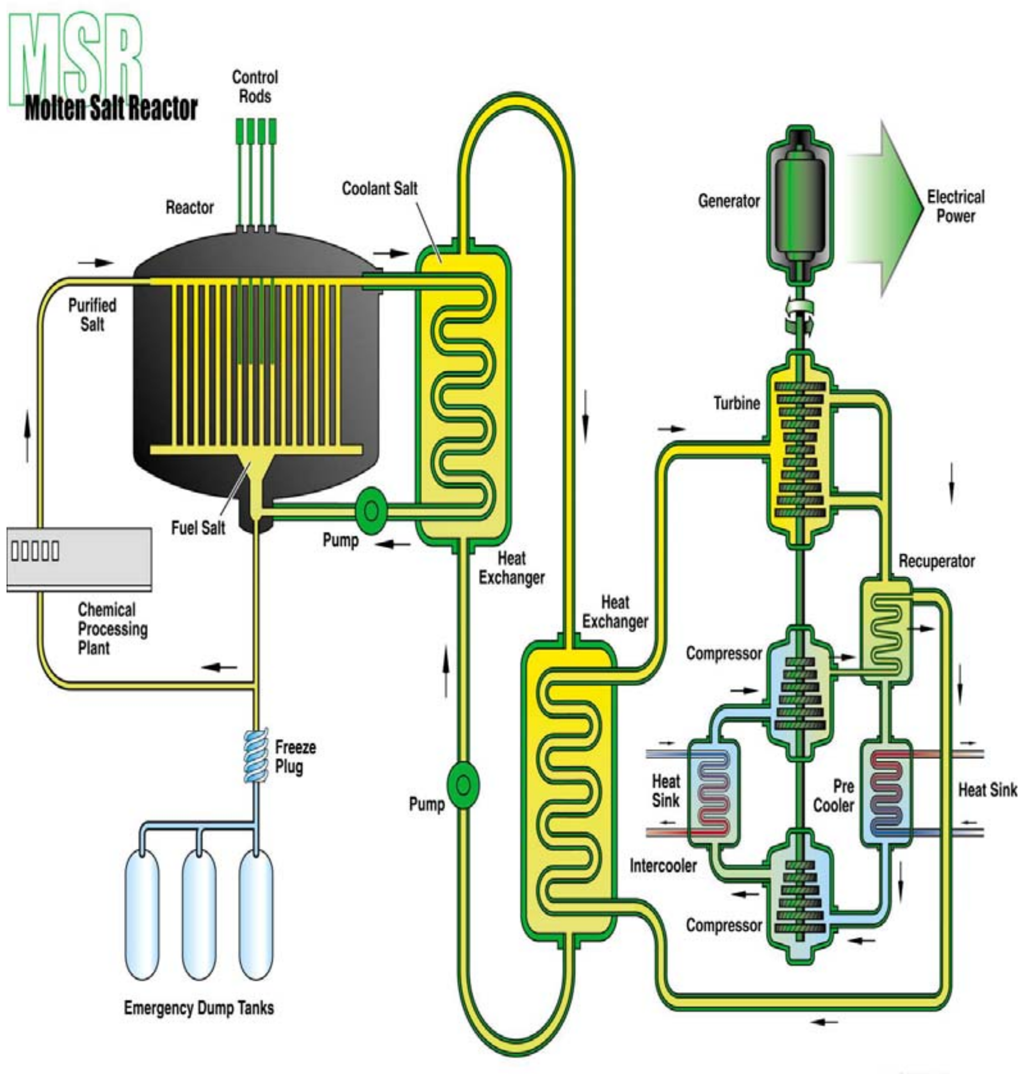


Figure A6.1. MSR with Brayton power cycle.

The use of a liquid fuel, versus the solid fuels of the other Generation IV concepts, creates potentially unique capabilities that are not achievable with solid-fuel reactors, but it also implies a different set of technical challenges than other Generation IV concepts. The unique capabilities include:

- Destruction of long-lived radionuclides without the need to fabricate solid fuels
- A wider choice of fuel cycles (once through, waste burning, fissile fuel production [breeding]) without major changes in the reactor design
- Very low fissile fuel inventory relative to other reactor concepts (fissile inventory may be as low as a tenth of a fast reactor per kW_e) that may create alternative safeguards strategies
- Full passive safety in very large reactors with associated economics of scale (under accident conditions, the fuel is drained to passively cooled, critically safe storage tanks)
- Limiting the radioactivity in the reactor core (accident source term) by on-line removal and solidification of the mobile fission products
- Limited excess reactivity requirements in the core due to on-line fuel management.

Nuclear reactor types can be classified by power output and the peak temperatures of their coolants (Figure A6.2). Light water reactors (LWRs) are low-temperature, high-pressure reactors. Traditional fast reactors cooled with liquid sodium operate at medium temperatures and low pressures. Two options exist for high-temperature reactor coolants: (1) high-pressure gases and (2) low-pressure liquids with boiling points above the peak coolant temperatures. MSRs are a type of high-temperature reactor.

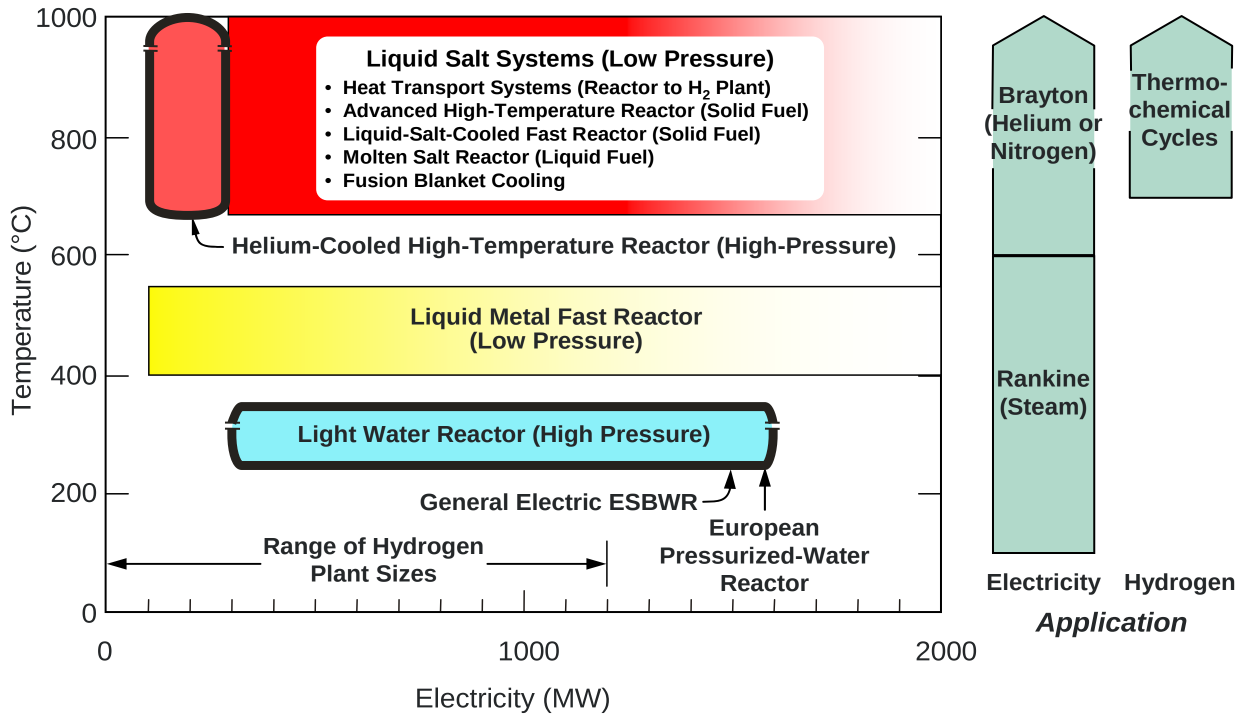


Figure A6.2. Reactor type versus temperature and power output.

The MSR was the high-temperature reactor developed to provide high-temperature heat for aircraft propulsion in the 1950s. It was then developed as a breeder reactor in the 1960s and early 1970s. Many of the technical challenges were a direct or indirect consequence of the limits of high-temperature technologies at that time. The Next Generation Nuclear Plant (NGNP) baseline concept is the modular, Very-High-Temperature Reactor (VHTR) using helium cooling. Because the NGNP is a high-temperature reactor, the development of the NGNP provides multiple key technologies for an Advanced Molten Salt Reactor (AMSR) such as Brayton power cycles (to replace earlier MSR steam cycles), compact heat exchangers (to replace tube-and-shell heat exchangers), and carbon-carbon composite materials (to replace some metallic components). The new technologies developed for the NGNP potentially imply major reductions in capital cost and reduce or eliminate about half of the technical challenges identified with MSRs.

One example can illustrate some of the implications of NGNP technology for MSRs. The traditional MSR had a steam power cycle. Steam-cycle peak temperatures are limited to ~550ºC, but the requirements for good physical properties for the molten salts imply operating temperatures at or above 700ºC. Temperature limits in the steam cycle prevented efficient conversion of MSR high-temperature heat to electricity. The cold-water temperatures required special design features to avoid freezing of the salt. The NGNP program is developing higher-temperature helium Brayton power cycles. Adoption of this new technology for an AMSR design simultaneously improves plant efficiency (with major improvements in economics) and eliminates multiple technical challenges.

Two experimental MSRs built at Oak Ridge National Laboratory (ORNL) established the basic technology for the MSR. The first reactor was the 2.5 MW_t Aircraft Reactor Experiment that in 1954 demonstrated peak operating temperatures up to 860°C. This was part of an effort to build a nuclear-powered military aircraft with the jet engines receiving heat from the MSR via an intermediate heat transport loop. This was followed in the 1960s by the Molten Salt Reactor Experiment, an 8 MW_t reactor, to demonstrate key features required for a Molten Salt Breeder Reactor (MSBR).

The renewed interest in MSRs is a consequence of changing goals and new technologies. Russian and Organization for Economic Cooperation and Development studies have identified the MSR as a potential component of a closed fuel cycle to efficiently burn actinides because it offers the potential to reduce the long-term radiotoxicity of the wastes produced from production of electricity in other types of reactors. The use of liquid fuels avoids some of the technical difficulties (such as fuel fabrication) for burning actinides—especially the intensely radioactive higher actinides. There is a secondary interest in the MSR's use for hydrogen production because of the high-temperature capability. In Europe, there is the traditional interest in the MSR as a thermal-neutron breeder reactor.

A6.1.2 Overall Systems Timeline

The overall systems timeline is shown in Figure A6.3 with viability determined by 2015. Because the basic technology of the MSR has been demonstrated, viability is defined as sufficient information to make a credible determination on the commercial feasibility of an MSR for power generation or viability to meet one of the new missions proposed for the MSR such as actinide burning.

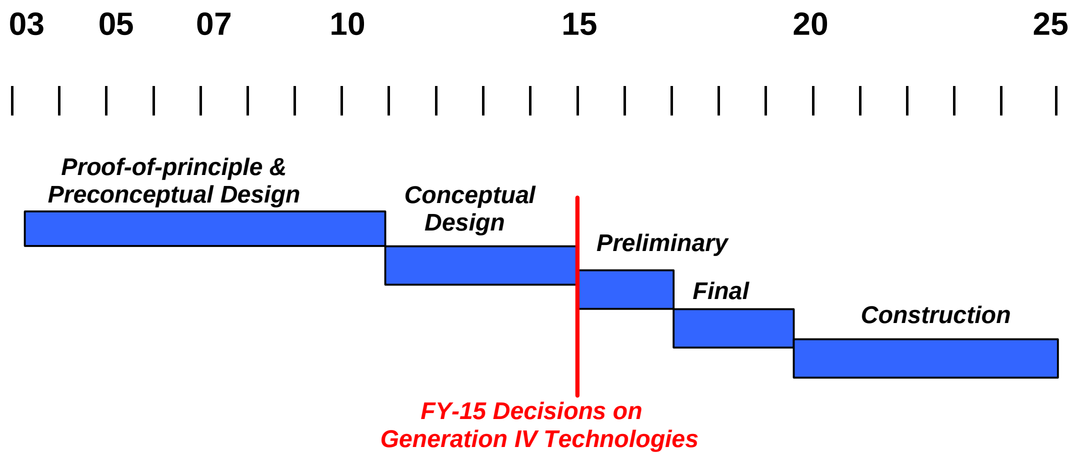


Figure A6.3. Ten-year project schedule.

A6.2 RESEARCH AND DEVELOPMENT STRATEGY

The research and development (R&D) strategy for the MSR is driven by three factors: (1) the billion-dollar MSR programs of the 1950s and 1960s that provided the technological foundation, (2) the technological overlap between the development needs for the MSR and other Department of Energy (DOE) programs, particularly the Generation IV NGNP (based on the VHTR) and the Nuclear Hydrogen Initiative (NHI), and (3) the European Community MSR programs. The technologies being developed for the NGNP provide the basis for an AMSR with major improvements in economics and reductions in research and development requirements for the MSR.

Molten fluoride salts, the base technology for the MSR, are being considered in multiple nuclear applications (see Figure A6.2). In the other applications, the salts are clean salts without dissolved fuel (referred herein as liquid salts). Much of the ongoing R&D for other applications is directly applicable to the MSR. Three of these applications are listed below:

- ***Liquid-Salt Heat-Transport Systems:*** Liquid salts are being investigated by the DOE Office of Nuclear Energy, Science, and Technology (NE) NHI Program for transport of heat from the NGNP to hydrogen production systems. Liquid salts are one of two candidates for this task. The technology is the same basic technology required for the MSR intermediate heat transport loop. These high-temperature heat-transport loops are also candidates for use in solar power towers and the in-situ recovery of shale oil.
- ***Advanced High-Temperature Reactor (AHTR):*** The AHTR is a solid fuel reactor that uses a clean liquid salt coolant to transfer heat from the solid reactor core to an intermediate heat exchanger. The AHTR uses the same high-temperature reactor fuel as the VHTR. The intermediate heat-transport loop transfers the heat to a Brayton power cycle or a hydrogen production facility. The high-temperature variant of the AHTR is called the Liquid-Salt-Cooled VHTR (LS-VHTR) and is a coolant variant within the NGNP program.
- ***Fusion Reactors:*** Liquid salts are major candidates for cooling inertial and magnetic fusion energy systems.

Although the DOE MSR program is a small program, it is organized to develop an AMSR through the utilization of technology from major well-established DOE programs such as NGNP and NHI, combined with historical ORNL MSR technology and current European fuel cycle technology.

A6.2.1 Objectives

The high-level objectives of the MSR R&D program within the Generation IV programs are to:

- Establish a pre-conceptual point design for a modern, economic MSR.
- Assess tradeoffs between the reactor design and potential fuel cycle missions such as transmutation.
- Develop a cost estimate for an MSR. Economic performance is an absolute requirement for large-scale deployment; thus, a preliminary understanding of MSR economics is required by 2010 when preliminary decisions on advanced reactors for fuel production are made.
- Establish the potential of energy conversion systems to use molten salts as heat transfer agents and the ability to couple the MSR with energy conversion devices.
- Coordinate with Advanced Fuel Cycle Initiative (AFCI) Program to develop an integrated fuel cycle that couples with other reactors for actinide burning.
- Interface with Generation IV International Forum (GIF) to optimize effectiveness of R&D plan with the larger French and European Community MSR programs.

A6.2.2 Scope

The scope of the MSR includes: (1) developing a conceptual design of an AMSR to provide an understanding of the economics, (2) developing the technologies to the point that there is reasonable confidence that an MSR could be fully developed, and (3) assessing and developing the associate fuel-cycle technologies to understand the capabilities of MSRs for multiple missions, such as actinide burning.

A6.2.3 Viability Issues

The top-level viability issue is economics—what are the economics of a large AMSR? Because MSR test reactors have been successfully operated, many technical viability issues have been addressed. Second level viability issues include optimum choice of the power cycle, determination of limits of actinide burning capabilities (neutronics and selection of optimum salt for actinide burning), salt processing options, design life of reactor materials, noble metal fission product management, licensing strategies, and safeguard strategies.

A6.2.4 Research Interfaces

A6.2.4.1 Relationship to Generation IV International Forum Research and Development Projects

Significant MSR R&D programs are being sponsored by the European Commission, various organizations in France, and in the Czech Republic. Since 2001, France has coordinated a European review and reevaluation of the MSR technology that has involved 13 participants. Under the sponsorship of the GIF, France is organizing a steering committee to prepare an international R&D plan that will couple various efforts worldwide. This will be the basis for an integrated Generation IV R&D program.

A6.2.4.2 University Collaborations

The MSR program is a joint effort between ORNL and the University of California at Berkeley.

A6.2.4.3 Industry Interactions

Industry interactions are through the NGNP and NHI programs where there is a common interest in development of liquid-salt heat transport systems.

A6.2.4.4 International Nuclear Energy Research Initiative

There is one International Nuclear Energy Research Initiative (I-NERI) activity with the European Community that is funded by the DOE NGNP program under the LS-VHTR activity. Within the European community, it is part of their examination of MSRs. The activities associated with this specific effort are measurement of selected salt properties and better methods to measure salt properties on line, a cross cutting activity for all salt-related activities.

A6.3 HIGHLIGHTS OF RESEARCH AND DEVELOPMENT

Because of ongoing synergistic programs, major advances in development and understanding of MSRs are expected to occur within the next decade with a modest investment of resources. This should enable the program to develop a credible understanding of the economics, capabilities to perform alternative missions (electricity, hydrogen, actinide burning, and fuel production), and issues associated with a modern MSR and, thus, provide the basis for a decision on whether to initiate a large-scale developmental program with the goal of deployment.

A6.3.1 System Design and Evaluation

The goal of the system design and evaluation studies is to optimize system design for a modern MSR. Since development of detailed conceptual designs for large MSRs in the 1960's, major changes in goals, regulatory requirements, and technologies have occurred that have not yet been integrated into the conceptual design approach for a next generation MSR. Coordination with the NGNP and NHI, which use common technologies and international partners, is critical to optimizing the resources available for MSR development.

A6.3.1.1 Design Optimization

The objective of this work is to determine the characteristics and design parameters of a modern, optimized MSR that update the 30-year-old design established for the 1000+ MW_e MSR. Three major changes must be incorporated into a modern MSR design. First, the new high-temperature NGNP technologies (such as Brayton cycles) must be incorporated into the MSR design because they simultaneously eliminate previously identified technical challenges associated with earlier MSR designs, improve plant efficiency, and reduce the capital cost per kW_e. Second, advances in non-NGNP technologies—such as remote operations, robotics, and controls—must be incorporated into the conceptual design. Last, changing mission requirements will simplify plant design. Early MSRs were designed to maximize fuel production; that mission, in turn, required complex, high-capacity, on-line salt processing. For actinide burner and hydrogen missions, there is the potential to eliminate most on-line fuel processing systems and greatly simplify the plant design.

A6.3.1.2 Regulation

Liquid fueled reactors use different approaches to reactor safety than solid fueled reactors. These include: (1) draining the fuel into critically safe, passively cooled tanks if off-normal conditions occur, (2) limiting excess reactivity by online fuel processing and/or continuous fueling, and (3) limiting fission product source terms by on-line processing. The current regulatory structure was developed with the concept of solid-fuel reactors. The comparable regulatory requirements for this system must be defined. Using current tools, appropriate safety analysis is required followed by appropriate research on the key safety issues.

A6.3.1.3 Safety

The objective of this work is to obtain the information required to assure MSR safety. The critical safety requirement for an MSR is that the radionuclides remain dissolved in the molten salt under all conditions. The reactor size, design, and safety systems are dependent upon this property. There are two basic R&D tasks: (1) determine the limits of the solubility of trivalent actinides in candidate molten salts and (2) assure control of noble metal fission products in the primary system. New applications for MSRs, such as actinide burning, imply higher concentrations of trivalent actinides and noble metals in the salt than were used in the past and may require modification of the salt composition to assure solubility under all conditions. R&D is required to determine the trivalent solubility limits under these different conditions. Similarly, the behavior of noble metal fission products in the salt and their ultimate disposition is required. Under some conditions, fission product noble metals may plate out on heat exchangers resulting in high decay heat loads and limited equipment lifetimes.

A6.3.2 Fuels and Fuel Cycles

Molten salt fluorides are stable under irradiation; thus, there is no need for a classical solid-fuel development program. However, there are multiple fuel cycle challenges. Some are common to other reactors and their associated fuel cycles, and some are unique to the MSR. Specifically, because the system is a molten fluoride salt system, there are unique chemical issues not associated with other reactors. There is a need to develop a fluoride high-level waste form and an integrated fuel recycle strategy. Since earlier MSR efforts, there have been major advances in separation technologies and proposals for highly innovative separation systems unique to fluoride salts. Preliminary exploration of these systems is appropriate because of their potential. This activity is currently being coordinated in the AFCI program at the systems level. More detailed efforts will be required in the future.

A6.3.3 Energy Conversion

The goal of the energy conversion R&D is to establish the technical basis for coupling Brayton cycles for electricity production and thermochemical water cracking cycles for hydrogen production to MSRs. These activities are expected to take place as part of an effort on crosscutting energy conversion R&D.

A6.3.3.1 Development of Heat Exchangers for Coupling to Energy Conversion Systems

Fluoride salts are leading heat transfer fluid candidates to transfer heat from the NGNP to hydrogen production facilities in the NHI program. This requires the development of high-pressure helium to low-pressure salt heat exchangers. The same technology is required to transfer heat from the MSR to a helium power cycle—except the heat is transferred from the molten salt to the helium in the power cycle. Consequently, the R&D will be coupled to that of the crosscutting energy conversion R&D.

A6.3.3.2 Development of Multi-Reheat Brayton Power Cycles

The proposed MSR power cycle is an indirect, multi-reheat, helium Brayton cycle. Most, but not all, of the components in this system are very similar to those required for the NGNP program. Consequently, the R&D will be coupled to that of the NGNP program.

A6.3.4 Materials

The major goal of the materials R&D is to identify and qualify materials with properties appropriate for MSR operating conditions, including corrosion resistance, mechanical performance, and radiation performance. The primary materials of interest are the moderator (graphite) and the reactor vessel/primary loop alloy (presently a Ni-based alloy). It is also necessary to develop corrosion control and coolant monitoring strategies for protecting the reactor vessel and primary piping alloys. The viability R&D will establish the primary candidate materials and control and monitoring strategies for further testing.

In addition to the historical experimental experience with molten salts at very high temperatures (~900°C) obtained for the Aircraft Nuclear Propulsion Program, an extensive materials development effort supported engineering code qualification for the MSBR to operate at 705°C. This temperature limit was largely due to the coupling required for steam cycle operations and did not represent a fundamental limit. Thus, there is a natural base to build on to extend candidate materials for the higher temperature objectives of the Generation IV program. The MSR and NGNP both use graphite as a moderator and various carbon-carbon composites for multiple structural applications; consequently, graphite and carbon-carbon research will be coupled to the NGNP. Because the NGNP is currently pursuing Ni-based super alloys for reactor components, development of Ni-based alloys for molten salts is coupled to the NGNP efforts. In parallel, there is ongoing European Community work in the Czech Republic and in Russia on testing of advanced MSR alloys. The ongoing work in the Czech Republic is coupled to these activities through the GIF program.

A6.3.4.1 Survey and Selection of Candidate Salt and Structural Materials

Candidate salts and materials will be selected based on literature survey, system design requirements, and investigation of materials use in industrial applications. In an MSR, the designer selects both the specific molten fluoride salt composition and the materials of construction. The demands of actinide burning may result in a choice of non-radioactive salt constituents that is different from previous applications. However, most of the work is not strongly dependent on the salt composition. Materials testing will take place over the range of temperatures, flows, and stresses expected in the MSR system.

A6.3.4.2 Irradiation Testing of Candidate Salts and Structural Materials

Candidate materials and salts will be irradiated under expected neutron spectrum conditions to extend the existing knowledge base to meet the Generation IV MSR requirements. Following irradiation, materials are screened for adequate mechanical performance, dimensional stability, and corrosion resistance.

A6.3.4.3 Materials Modeling

Advanced, mechanistically-based models for radiation performance will be developed. Developing materials modeling is expected to be a crosscutting activity.

A6.4 PROJECT COST AND SCHEDULE

A6.4.1 Fiscal Year 2006 Project Budget

The ten-year budget projection includes two classes of activities. The first is activities that support GIF R&D planning and coordination. As shown in table A6.1, this is the only class of activities included in the Fiscal Year (FY) 2006 budget. The second is those associated with the R&D program. Funding requirements are being defined. The program is based upon input from European GIF activities and the relevant U.S. NGNP activities aimed at providing a basis for an advanced MSR.

Table A6.1. FY 2006 budget profile for MSR activities ($K).

Task	FY-06
GIF R&D Planning and Coordination	40
Total	40

A6.4.2 Ten-Year Project Schedule

The DOE MSR program is a small activity, which is dependent upon the larger DOE NGNP, DOE NHI, and MSR GIF activities being supported by our European partners. Consequently, the schedule to determine the viability of an AMSR is determined by these larger programs. The European programs that are part of GIF (the largest and best funded MSR programs) plan to establish the viability of the MSR by 2015 and to optimize its design features and operating parameters by 2020. Per the NGNP program schedule, the NGNP plant will in 2017; that implies that the results of NGNP R&D programs will be available in a similar timeframe. The activities of the NGNP program, NHI program, and the GIF program planning are expected to provide a more detailed program schedule within the next one to two years.

A6.4.3 Ten-Year Project Milestones

The U.S. MSR activities are chosen to couple DOE and GIF activities to determine the viability of an AMSR and to address specific U.S. concerns. The milestones of the Ten-Year Program Plan are as follows:

FY 2006

- Completion of draft GIF R&D plan (lead by France)
- Assessment of MSR nonproliferation characteristics.

FY 2007

- Completion of final GIF R&D plan
- Assessment of actinide burning options (parallel to Global Nuclear Energy Partnership studies).

FY 2008

- GIF coordination
- Updating pre-conceptual design of an MSR based on NGNP, NHI , and GIF MSR technology developments

FY 2009

- GIF coordination
- Assessment of licensing issues

FY 2010

- GIF coordination
- Assessment of integrated development and commercialization plan

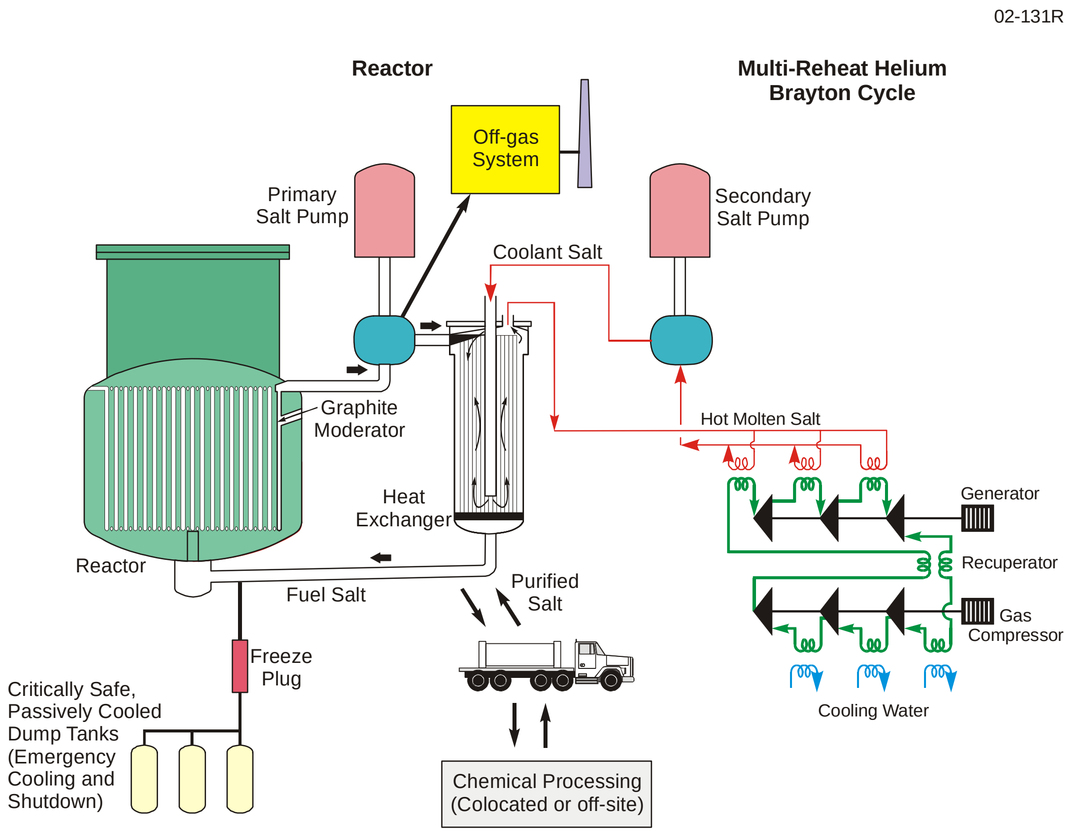
FY 2011-2015

- To be defined based on GIF R&D plan.

ADDENDUM A6-1: DESCRIPTION OF AN ADVANCED MOLTEN SALT REACTOR

General Design

In an MSR (Addm A6: Figure 1), the molten fluoride salt with dissolved fissile, fertile, and fission isotopes flows through a reactor core moderated by unclad graphite. In the core, fission occurs within the flowing fuel salt, which then flows into a primary heat exchanger where the heat is transferred to a secondary molten salt coolant. The fuel salt then flows back to the reactor core. The graphite-to-fuel ratio is adjusted to provide the optimal neutron balance—an epithermal neutron spectrum. In the preconceptual 1000-MW_e designs developed in the early 1970s, the liquid fuel salt typically enters the reactor vessel at 565ºC and exits at 705ºC and ~1 atmosphere (coolant boiling point: ~1400ºC). The reactor and primary system are constructed of modified Hastelloy-N or a similar alloy for corrosion resistance to the molten salt. Volatile fission products (e.g., krypton and xenon) are continuously removed from the fuel salt.



Addm A6: Figure 1. MSR with Multi-reheat Brayton Cycle

The secondary coolant loop transfers the heat to the power cycle or hydrogen production facility. The secondary heat transport loop also uses a liquid salt that may be the same molten salt used in the primary system (except it is a clean salt with no fuel or fission products) or another fluoride salt. The secondary coolant (1) provides isolation between the low-pressure reactor and either the power cycle (if electricity is being produced) or a hydrogen production facility and (2) chemically reacts and traps tritium that escapes from the primary system. With a fluid-fuel reactor, the tritium is not trapped in the solid fuel and tends to migrate across hot heat exchangers. A small cleanup system removes the tritium from the secondary coolant.

The parameters developed for the 1000-MW_e MSBR conceptual design developed in the late 1960s are shown in Addm A6: Table 1. These parameters are for a large (2250 MW_t) ^{233}U-thorium, liquid-fuel breeder reactor designed for the production of electricity using a steam cycle.

Addm A6: Table 1. Design characteristics of the 1970s MSBR.

Parameter	Value
Net electric generation	1,000 MW
Thermal efficiency	44.4% (steam cycle)
Core height	3.96 m
Vessel design pressure	$5.2 \cdot 10^5$ N/m^2 (75 psi)
Average power density	22.2 kW/L
Graphite mass	304,000 kg
Maximum core flow velocity	2.6 m/s
Total fuel salt	48.7 m^3
^{233}U	1,500 kg
Thorium	68,100 kg
Salt components	^7LiF-BeF_2-ThF_4-UF_4
Salt composition (see entry above)	71.7-16-12-0.3 mol %

The reactor characteristics minimize the potential for accident initiation. Unlike solid-fuel reactors, MSRs operate at steady-state conditions with no change in the nuclear reactivity of the fuel as a function of time. Fuel is added as needed; consequently, the reactor has low excess nuclear reactivity. No excess fuel is needed at reactor startup to compensate for fuel depletion, and no excess reactivity is required to override xenon poisoning. No significant buildup of xenon occurs over time because the xenon gas continuously exits via the off-gas system. There is a strong negative temperature coefficient because increased temperatures lower the fuel-salt density and push fuel out of the reactor core. In normal operations, the control rods are fully removed from the reactor.

Early designs of the MSR proposed the use of a steam cycle for electricity production. Current proposals for an MSR use a multi-reheat helium or nitrogen Brayton cycle. The Brayton cycle has major advantages over the use of a steam Rankine cycle: simplified balance of plant with lower cost, improved efficiency, reduced potential for salt freezing in the heat exchangers, and simplified control of tritium within the reactor. The estimated helium Brayton power-cycle efficiency is 48%, compared to 44% for the MSR with steam cycle. This improved efficiency is a consequence of adopting a Brayton power cycle that is a better match to molten salt systems than steam power cycles. The helium or nitrogen Brayton cycle also minimizes difficulties in the control of tritium. In a liquid-fuel reactor, fission-product tritium is not trapped in solid fuel. It can migrate through hot heat exchangers to the power cycle. In a Brayton cycle, it is easy to remove any tritium that enters the power cycle from the dry gas. This is in contrast to a steam cycle where any tritium diffusing through hot heat exchangers with their very large surface area combines with the steam.

Decay Heat Cooling and Accident Management

Molten salt reactors use passive emergency core cooling systems that are radically different from those used in solid-fuel reactors. If the molten reactor fuel salt overheats, its thermal expansion causes it to overflow by gravity into an overflow weir. The fuel is then dumped to multiple critically safe storage tanks with passive decay-heat cooling systems. Freeze valves that open upon overheating of the salt can also be used to initiate core dump of fuel. Drains under the primary system also dump fuel salt to the storage tanks if a primary system leak occurs. This design approach allows very large reactors to be built with passive safety systems.

Many of the driving forces for an accident are reduced compared with those that exist for other reactors. Fission products (with the exception of xenon and krypton) and nuclear materials are highly soluble in the salt and will remain in the salt under both operating and expected accident conditions. The fission products that are not soluble (e.g., xenon and krypton) are continuously removed from the molten fuel salt, solidified, packaged, and stored in passively cooled storage vaults. There are no major stored energy sources within containment such as high-pressure fluids (helium and water) or reactive fluids (sodium). This reduces requirements for the containment.

Reactor Physics and Fuel Cycle

Reactor Physics

As noted, MSRs are fluid-fuel reactors. Such reactors have several reactor physics characteristics that are different from those of solid-fuel reactors.

- ***Nuclear Reactivity:*** Negligible xenon effect occurs because xenon continuously escapes from the fuel salt into the off-gas system. There is no change in core reactivity with time because fuel is continuously added as required. The fuel inventory in the reactor core is coupled to the reactor temperature. An increase in reactor temperature reduces the fuel inventory by expansion of the fuel salt with less mass of fuel salt in the reactor core.
- ***Fissile Inventory:*** As a class, MSRs have very low fissile inventories compared with other reactors for several reasons: (1) thermal neutron reactors require less fissile inventory than fast reactors, (2) a low fuel-cycle fissile inventory exists outside the reactor system (no conventional spent nuclear fuel [SNF]), (3) little excess reactivity is required to compensate for burnup because fuel is added on-line, (4) direct heat deposition in the fuel/coolant allows high power densities, and (5) the high-absorption fission products, such as xenon, are continuously removed. As a consequence, the MSR requires <2 kg of fissile material per MW_e to reach criticality compared with 3 to 5 kg/MW_e for LWRs and over 25 kg/MW_e for fast-spectrum reactors. This implies that the MSR has the potential to provide long-term, sustainable energy production while limiting the global inventory of plutonium and minor actinides to a total quantity over an order of magnitude lower than solid-fuel reactors.
- ***Burnup and Plutonium Isotopics:*** Relative to solid-fuel reactors, MSR fuel cycles have very high equivalent fuel burnups and unusual plutonium isotopics with high concentrations of ^{242}Pu.
  - In solid-fuel reactors, SNF burnup is limited by fuel-clad lifetime that, in turn, limits fuel burnup and the burnout of plutonium. In non-breeder reactors, SNF burnup is also economically limited, independent of the technology. Excess fissile material is in fresh fuel when it is initially placed in the reactor core. This is required to compensate for fuel burnup over time. To assure reactor control, burnable neutron absorbers are then added to the fresh fuel to avoid excess nuclear reactivity in new fuel assemblies. There is an economic cost (extra enrichment) in "storing" excess fissile fuel in the new fuel assembly until it is needed toward the end of the fuel assembly lifetime. These factors fundamentally limit solid-fuel burnup.

- In an MSR, fuel is added incrementally to the liquid as required. No excess fuel and associated burnable absorbers are required. Selected fission products are removed from the molten salt and solidified as a waste form. Consequently, the normal burnup limits that define solid fuels do not apply to a liquid-fuel reactor. The plutonium remains in the salt, with the lighter plutonium fissile isotopes burned out faster than ^{242}Pu. This has major implications in terms of proliferation resistance. The high ^{242}Pu content makes the plutonium from an MSR much less desirable than plutonium from any other reactor type for use in weapons because of its very high critical mass.

- ***Delayed Neutron Fraction:*** In all reactors, some fraction of the fission neutrons are delayed neutrons emitted from the decay of very-short-lived fission products. This fraction is used for reactor control. Unlike solid fuel reactors, the flowing fuel implies that some fraction of the delayed neutrons will occur in flowing fuel that has left the reactor core. This must be accounted for in all reactor physics calculations and evaluations.

Fuel Cycle Options

Four major fuel cycle options exist to address different goals of reactor operation. The basic reactor remains unchanged except for the salt composition, salt-cleanup systems, ratio of salt-to-moderator ratio, and fuel cycle operations. Any of the MSR/fuel-cycle options can be started up using low-enriched uranium or other fissile materials. With the exception of the breeder reactor fuel cycle, the fuel salt processing for all the other fuel cycles can be performed off-site with removal of the fuel salt every few years.

Actinide Burning

This fuel cycle burns multi-recycle plutonium, americium, and curium from LWR SNF or other sources to reduce the long-term hazard of wastes to an SNF repository. It can also produce denatured ^{233}U as a by-product. The penalty for burning actinides in an epithermal neutron flux is partly offset by the greater fission neutron yield of the higher actinides. As an actinide-burner, the MSRs will produce 10% more electricity than the other reactors that originally generated the actinides. This mode of operation requires a molten salt, such as a sodium-zirconium fluoride salt, that has a high solubility for actinides. In the process of burning actinides, the actinides with high fission cross sections are burnt out first. It requires substantially longer times to burnout low nuclear-cross-section actinides. Consequently, there is a buildup of low cross-section actinides in the reactor. This implies that any reactor burning actinides from LWRs will have a larger inventory of actinides in the reactor core than with other MSR fuel cycles.

Much of the current interest in MSRs is a result of the reactors' capabilities to burn actinides to reduce waste management burdens. Because they are liquid-fuel reactors, MSRs offer three advantages over solid-fuel reactors in this application:

- ***No Isotopic Blending:*** Different lots of SNF have different Pu, Am, and Cm isotopics. The MSR has a homogeneous liquid fuel. Any fissile material can be fed to the reactor where it mixes with the whole volume of the fuel salt. The different nuclear characteristics of different batches of higher actinides are addressed by the rate of addition to the homogeneous molten fuel salt. In contrast, in solid-fuel reactors, the quantity and isotopics of the fissile materials in every location of every fuel assembly must be controlled to avoid local overpower conditions that burn out the fuel. With complex mixtures of isotopics, the process of mixing fissile materials to obtain uniform solid-fuels is expensive and difficult to accomplish.

- ***No Fuel Fabrication:*** The higher actinides have small critical masses and high rates of decay heat representing a serious technical and economic challenge for solid-fuel fabrication. This is a non-issue for an MSR because no fuel fabrication is required.
- ***Minimal Reprocessing:*** In an MSR, fission products are removed from the molten salt, while actinides remain in the salt. This is the reverse of traditional processing, in which clean fissile materials are extracted from SNF. In an MSR, the cleaned fuel salt is to be mixed back with the salt in the reactor. Some fraction of the fission products must be removed, but there is no reason to fully clean the salt. Processing would be done as a batch process at a collocated or off-site location.

Once-Through Fuel Cycle

The once-through fuel cycle converts thorium to ^{233}U internally in the reactor and uses 20% enriched uranium as fresh fuel to the reactor. The annual fuel consumption is ~45 t/GW_e, or about one-fifth that of an LWR. No recovery of fissile material from the discharged salt (SNF) would be required.

^{233}U-Thorium Breeder Cycle

MSRs can operate as breeder reactors. After startup, only thorium is added as a fuel. A breeder reactor with efficient fuel production also requires on-line processing of the fuel salt because of the nuclear characteristics of breeding fuel with thermal neutrons using the ^{233}U-thorium fuel cycle. In a thermal neutron breeder reactor, the breeding reaction is $^{232}Th + n \rightarrow {}^{233}Pa \rightarrow {}^{233}U$. Unfortunately, ^{233}Pa has a moderately large absorption cross section and a half-life of 27 days. If it is left in the reactor, parasitic capture of neutrons by ^{233}Pa will occur resulting in a significant reduction in the breeding ratio. To avoid this scenario and to obtain high breeding ratios, on-line processing is required for removal of the ^{233}Pa and storage outside of the reactor until it decays to ^{233}U. With an efficient processing system, the breeding ratio is ~1.06, with an equilibrium ^{233}U inventory of about 1,500 kg. If the reactor is to be a breeder reactor, the fuel salt characteristics must be optimized and will almost certainly be a mixture of ^{7}LiF, BeF_2, ThF_4, and UF_4. This salt mixture provides better neutron economy. The use of a ^{233}U-thorium breeder reactor cycle results in a high-level waste with a very low actinide content because as neutrons are added to the thorium, the various fissile isotopes that are produced (^{235}U, ^{239}Pu, etc.) tend to fission.

There has been one important change in the breeder reactor fuel cycles. In the 1960s, it was thought that uranium resources were limited; thus, the goal was to maximize the breeding ratio to provide the fuel for a rapid buildup of additional nuclear power plants. It is now recognized that the uranium resources are much larger than originally estimated. Consequently, existing fissile fuel resources may be sufficient to initially fuel any required number of reactors. In this case, the only long-term requirement is to make fuel as fast as it is consumed. This requires a net breeding ratio of 1. Lowering the required breeding ratio reduces the requirements for the on-line processing of the fuel salt and may allow major simplifications in salt processing. These options are being investigated in Europe.

Denatured ^{233}U-Thorium Breeder Cycle

This is a breeder reactor fuel cycle designed to maximize proliferation resistance by minimal processing of the fuel salt and by addition of ^{238}U to isotopically dilute fissile uranium isotopes. This lowers the breeding ratio to slightly above 1 and results in a very low fissile plutonium (^{239}Pu and ^{241}Pu) inventory of ~0.16 kg/MW_e. The use of a thorium-^{233}U breeder reactor cycles results in a high-level waste with a low actinide content because as neutrons are added to the thorium, the various fissile isotopes that are produced (^{235}U, ^{239}Pu, etc.) tend to fission.

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