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Mini-Assessment of Advanced Technology Options for NASAP

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Engineering Physics Division

MINI-ASSESSMENT OF ADVANCED TECHNOLOGY OPTIONS FOR NASAP

- HOMOGENEOUS MOLTEN SALT SUSTAINER REACTOR
- GRAPHITE-MODERATED HETEROGENEOUS GAS CORE REACTOR

Coordinated By:

R. T. Santoro
Oak Ridge National Laboratory

Principal Authors

R. J. Barrett
E. J. Gahan
G. Nugent
R. J. Reyer
R. L. Schwartzberg
Burns and Roe, Inc.

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OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
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1.0. INTRODUCTION

A mini-technical assessment was performed on the following two advanced reactor concepts:

- Homogeneous Molten Salt Sustainer Reactor (HMSSR)
- Graphite-Moderated Heterogeneous Gas Core Reactor (HGCR)

The information that was made available by the proposers for these concepts was very sketchy and allowed only a mini-technical assessment to be performed. The discussions in this mini-assessment follow closely the discussions given in Ref. 1.0-1. In fact this report should be considered a supplement to Ref. 1.0-1 and a complete understanding of the discussions will not be possible without Ref. 1.0-1.

The HMSSR concept is similar to the Denatured Molten Salt Reactor (DMSR) assessed in Reference 1.0-1 and in the technical assessment of the HMSSR reference and comparison was made with the DMSR.

The HGCR concept is similar to the Mixed Flow Gas Core Reactor (MFGCR) assessed in Reference 1.0-1 and in the technical assessment of the HGGR reference and comparison was made with the MFGCR.

The primary objective of this assessment was to compare the technical feasibility and proliferation resistance potential of these two concepts with their corresponding representatives in Ref. 1.0-1. A technical feasibility assessment of concepts in very early stages of development must consider their commercial potential and the issues that comprise it, such as reactor design, research and development needs, economics, environmental, and safety issues. Due to the nature of this assessment, the conclusions should be considered valid only for the evaluation within NASAP for which they were intended. Indeed, other concepts or major modification to the present concepts may be expected to improve the anticipated benefits and decrease the uncertainties associated with some of these advanced nuclear power concepts in the future. Additional research and development would be expected to either disprove or improve any given design, and design optimization of these two reactor concepts might easily cause changes in this assessment. This assessment faced the difficult task of evaluating two technologies that are in the very earliest stages of development. The evaluation of commercial potential is particularly difficult.

Most problems identified throughout this assessment require solutions through research and development. These problems and research needs are generally presented without providing details of the major programs specifically addressing them. Neither concept has advanced to the point where a detailed research and development program has been undertaken. The progress of any research program depends on the ability to resolve outstanding problems. Since these concepts are in a very preliminary stage of development, the problems identified in this assessment are likely to be supplemented by myriad other difficulties as the program proceeds. This aspect and the large uncertainty in the designs imply that any estimates of RD&D schedule and costs amount to no more than guesses. The reason for providing RD&D estimates is to provide a basis of comparison among the various concepts. The difficulties and methods used to estimate RD&D costs and schedules are further discussed in Appendix A of Ref. 1.0-1.

The uncertainty affecting economic estimates of the two concepts is also quite large. The methods and uncertainties for the economic estimates are further discussed in Appendix B of Ref. 1.0-1.

The evaluation of the proliferation resistance of each concept is an important aspect of this assessment. Unfortunately, no hard-and-fast rules exist for measuring the proliferation resistance of any reactor fuel cycle. Any nuclear reactor system will have some potential for proliferation because all reactors require fissile material for reactor operation. Most reactor systems have inherent features which resist misuse of the fissile material. Those features or services which offer the least resistance can potentially have their resistance increased by a number of proliferation resistance techniques which include technical adjustments, physical safeguards, institutional controls, and political arrangements and sanctions.

2.1 HOMOGENEOUS MOLTEN SALT SUSTAINER REACTOR

2.1.1 Conceptual Plant Design

2.1.1.1. Summary

The HMSSR is a thermal neutron spectrum reactor with a homogeneous core. Two different salts are circulated within the reactor core, a fissile fuel-salt within the reactor core and a fertile blanket-salt around the core. The reactor uses a $^{232}Th/^{233}U$ fuel breeding cycle and for purposes of this assessment is sized to provide a net electric output of 1000 MWe.

In comparison with the Denatured Molten Salt Reactor (DMSR) the HMSSR differs in design requirements as follows:

- Moderator is contained in the homogeneous fuel salt mixture rather than as fixed graphite core sections.
- Smaller size reactor vessel.
- Lower fissile (fuel-salt) inventory.
- More compact reactor design.
- Requires additional processing systems to clean up and separate constituents from both the fuel salt and the blanket salt.
- Requires an internal vessel to separate core from blanket salt.

2.1.1.2. Plant Arrangement

Plant arrangements were not described for the HMSSR, therefore an assessment of the arrangement cannot be provided directly and certain assumptions must be made. Based on information provided in Ref. 2.1-1, a HMSSR is expected to be capable of producing a net electric output of 1000 MW using a compact reactor vessel of spherical design. The HMSSR reactor vessel is anticipated to be much smaller in overall dimensions than the DMSR vessel. However, the overall size of the HMSSR reactor building will probably be larger than an equivalent DMSR building due to the additional systems which must be contained. In particular, the HMSSR must provide for the following systems which are unique to the homogeneous design:

- Thorium blanket salt circulating system.
- Fuel salt solid fission product removal and processing systems.

- Thorium blanket gaseous and solid fission product clean-up, processing and removal systems.
- Thorium blanket fuel drain system.
- Thorium blanket heat-up system.

2.1.1.3. Reactor Design

The reactor design described in Ref. 2.1-1 is that of a spherical vessel in which a primary fuel-salt is circulated through the interior core of the vessel and a blanket salt is circulated around the core. The primary salt contains ^{233}U as a fissile material. The blanket is primarily a breeding region since its salt contains thorium. Thermalization of neutrons is provided by the neutron moderating effects of beryllium as a salt constituent in both regions. The reactor is designed for a breeding ratio of one, although this ratio can be varied to accommodate a wide variety of other design objectives, such as proliferation resistance, integral fuel processing, breeding gain, fissile or fertile material inventory, denatured or nondenatured fuel, etc. Enriched uranium would be required for initial operation after which the HMSSR would supply its own fissile material needs by breeding ^{233}U.

The HMSSR blanket is separated from the core salt by an internal reactor core shell to be fabricated of fiber-reinforced graphite. In addition to separating the fuel salt from the blanket salt, this internal vessel acts to enhance thermalization of neutrons leaked from the core to the blanket region. Figure 2.1-1 presents a schematic of the HMSSR. The development of the internal vessel represents a major technological problem for this concept. The major question regarding the development of a fiber-reinforced graphite shell is the ability of the material to withstand the radiation to which it will be exposed for a suitable reactor life. Hastelloy N alloy could be used to fabricate the internal shell and is suggested as an alternate. However, a penalty in the nuclear performance is imposed when this alloy is used. An outer neutron reflector region may be desirable to improve the breeding performance of the blanket region.

As is the case with the DMSR concept, a good deal of testing and investigative work remains to be performed to demonstrate the technical feasibility and includes:

- Reactor materials of construction under the irradiation, temperature, and molten salt environment.
- Reactor core hydrodynamics.
- Reactor vessel dimensional stability.
- Reactor control and safety features, including control and safety rod requirements, anticipated life, and instrumentation.
- Containment penetrations, seals, wiring and insulation materials when subjected to the operating conditions.

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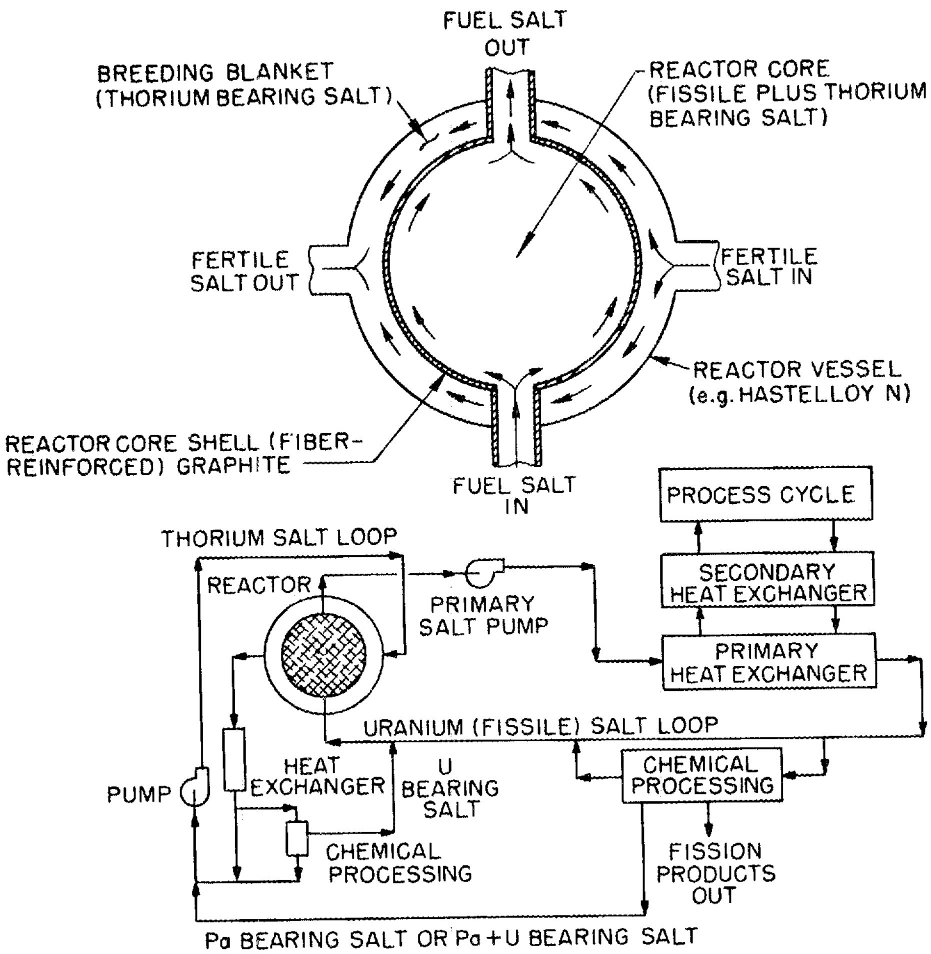

Fig. 2.1-1. HMSSR Schematic.

2.1.1.4. Primary Heat Transport System

Reference 2.1-1 does not describe the primary heat transport system for the HMSSR. However, it does state that the plant external to the reactor vessel could be similar to those of the ORNL design for the DMSR. Based on this statement fuel salt will be assumed to enter the reactor core at a temperature of approximately 1050°F, and leave at 1300°F. The principle difference in the primary heat transport systems of the two molten salt concepts then becomes the composition of the fuel salt and the requirement to process the fuel salt of the HMSSR in order to remove the fission product impurities. The processing system design and development appears to be a major obstacle to the construction of a HMSSR plant.

The blanket salt system is unique to the HMSSR with no equivalent system in the DMSR. The thorium salt undergoes some fissioning in the blanket region and is a source of both heat and fission products, both of which must be removed from the reactor vessel. These systems have not been described. Design and developmental work will be required to provide suitable means to remove both the heat and fission products.

Since the primary heat transport systems are assumed similar, the HMSSR offers no advantages over the DMSR in terms of reduced design, development, or engineering work. The requirements for programs to design, develop, and test heat exchangers and pumps are the same as for the DMSR as is the need for control, instrumentation, and flow measurement devices which will be capable of withstanding the temperatures, environment, and cyclic behavior of HMSSR.

2.1.1.5. Secondary Heat Transport System

The secondary heat transport system of the HMSSR is assumed to be the same as that of the DMSR. Therefore, the requirements will be the same. The major uncertainties of the DMSR system are summarized below:

- The effects of water intrusion into the coolant-salt and the degree of water intrusion which can be permitted must be studied to determine the precautions and limiting parameters which must be imposed on the steam generator design.

- The effects of a large scale-up for the coolant-salt circulation pumps must be considered.

- The effects of tritium diffusion from the primary heat transport system to the secondary heat transport system must be analyzed, and means of preventing the tritium from diffusion through the walls of the secondary heat transport system components and piping walls must be developed and demonstrated.

2.1.1.6. Containment System

Recognizing that the containment design is dependent upon the size and shape of the reactor vessel, HMSSR will require features not presently associated with light water reactors. In particular, the HMSSR containment must provide for penetrations of the secondary heat transport system as well as the blanket salt system piping, controls, wiring, and instrumentation.

As in the case of the DMSR the development of systems for inerting, cleanup, and disposal will be required. The design and development of such systems is considered technically feasible although much time and effort will be required to optimize these systems.

2.1.1.7. Radioactive Waste Control Systems

The need for and the function of the radioactive waste control systems for the HMSSR are outlined in Ref. 2.1-1. Unlike the DMSR, which is designed to preclude the removal of solid fission products from the fuel-salt, the HMSSR is dependent upon cleanup of the fuel for sustained operation. A small sidestream of the fuel salt is to be continuously processed in an in-plant processing system to remove solid fission products from the fuel-salt. In a similar manner the blanket-salt must also be processed in order to separate the bred ^{233}U from the salt and direct it to the fuel salt stream. These systems will require extensive development and design work.

In addition to the above salt processing systems, gaseous fission products must also be removed from the salts. An off-gas system similar to that of the DMSR will serve equally well in treating the HMSSR off-gas. As in the case of the DMSR, the gaseous isotopes of concern such as xenon, krypton and tritium, will be recovered and bottled with the objective of essentially zero release from the plant. As noted for the DMSR, the HMSSR will require additional studies for the off-gas system to determine:

- The feasibility of storing high level radioactive wastes on-site.
- The means of transporting wastes from their initial location to the storage area.
- Storage area requirements for size, shielding, and special systems (cooling, inerting, circulation, etc.).
- Removal of wastes from storage for processing and disposal at another location.
- Means of immobilizing wastes to accommodate shipment should a plan be adopted which calls for less than 30 years of storage on-site.

2.1.1.8. Engineered Safety Features

Information on the engineered safety features systems is not contained in Ref. 2.1-1; but a fuel-salt drain system is envisioned to be similar to that of the DMSR. This system will serve to stop the fissioning of uranium by removing the fuel from the core and putting it in a subcritical configuration. The drain system will require a means of removing the decay heat from the salt. Again, as in the DMSR, such a system appears technically feasible.

In addition to the fuel-salt drain system, blanket-salt drain and cooldown systems will be required. These systems should be similar to those for the fuel salt and will require the same degree of engineering design and reliability.

Owing to the design of the HMSSR, a "catch pan" similar to the one used in the DMSR for containing fuel-salt spills and leaks is difficult to provide. In the HMSSR the catch pan is needed; however, segregating fuel-salt spills from blanket-salt spills appears almost impossible as both systems are present in the reactor cell. A common catch pan to contain spills and a processing system capable of separating fuel and blanket salts appears to be required.

In designing the fuel-salt drain system, the blanket-salt drain system, and the catch-pan for the combined salts, the positive shutdown mechanism of the salts being separated from the graphite moderator will not exist as in the DMSR. Therefore, special design care will be required to assume an adequate shutdown margin of reactivity in the drained condition and specific safety features may be required for this purpose.

2.1.1.9. Auxiliary Systems

The auxiliary systems for the HMSSR are the same as for the DMSR, and the assessment is the same.

2.1.1.10. Plant Structures and Shielding

The plant structures of a HMSSR must be designed, developed, and tested. As the reactor vessel for this concept is of spherical design, a completely different means of support will be required. The support structure must permit the radial growth of the reactor vessel as well as provide for restraint of the vessel in case of a seismic event. Since the plant structures are unlike those employed in present LWR plants, they appear to be more challenging to design and thus should require significant time and effort to develop.

Shielding of the HMSSR appears to be a major undertaking. In addition to shielding the reactor vessel, the primary heat transport system components and the interconnecting piping, it will be necessary to shield the components of the solid fission product radioactive waste control system, the blanket salt system, and the off-gas processing systems as well as the spaces in which these components and their appurtenances can be transported, i.e., the hot cells and the waste storage areas.

2.1.1.11. Steam and Power Conversion Systems

The steam and power conversion systems for the HMSSR are anticipated to be identical to those planned for the DMSR, and the assessment is the same.

2.1.2 Technology Status and Research, Development, and Demonstration

2.1.2.1. Summary

The capability of a HMSSR to sustain fission has not been demonstrated. The technological feasibility of the concept thus remains to be proven. In particular, the Research, Development, and Demonstration (RD&D) work associated with the reactor vessel, at this point in time represents a paper design. None of the steps required to prove the principle of operation of the HMSSR have been taken. Experimental programs to demonstrate the reactor concept and the soundness of the core physics remain to be accomplished. Additional barriers, namely those related to the processing of both the fuel and blanket slats, must be overcome in the process of demonstrating the concept's ability to provide an energy source for the future.

With the exception of the RD&D requirements of the DMSR moderator graphite, all of the other RD&D programs, tests and developmental activities of the DMSR are applicable to the Homogeneous Molten Salt Reactor concept.

The RD&D needs of the HMSSR are anticipated to exceed those of the DMSR. The time required to place a lead commercial HMSSR plant in operation is roughly estimated to be approximately 40-45 years, and the cost of the program leading to placing such a plant in operation should approximate $7 to 12 billion (1979 $) (Appendix A of Ref. 1.0-1 contains a discussion of estimating methods and uncertainties.)

2.1.3 Proliferation Resistance Features

While the proliferation resistant features of the HMSSR have not been investigated in detail, for the most part they are expected to be quite similar to the DMSR's proliferation resistant features. However, the addition of fuel and blanket reprocessing systems will substantially increase the proliferation vulnerability of the HMSSR and may require deployment in secure fuel service centers.

2.1.4 Reactor Physics Considerations

A reactor physics model has not been developed to verify the feasibility of a HMSSR. Much work remains to be done before the HMSSR can be considered an attractive concept. The uncertainties associated with the HMSSR outnumber those of the DMSR. Some of the specific items which must be investigated are:

- Whether fission can be sustained using moderator dispersal in the fuel and blanket salts. This is basic to the feasibility of the concept.
- The neutronics effects of a reactor core shell made of Hastelloy N versus graphite.
- The need for a neutron reflector region for the blanket.
- Criticality investigations of the fuel and blanket salt draining operations.
- Optimization of core and blanket parameters.
- Determination of coefficients of reactivity.
- Determination of radiation source terms for shielding design and for determining effects on reactor components (e.g., core shell).
- Determination of breeding ratios.

2.1.5 Fueling Alternatives and Resource Utilization

The fuel cycle for the HMSSR requires on-site processing for both the molten fuel-salt and the thorium bearing blanket-salt. Through this reprocessing scheme, the HMSSR can convert thorium to ^{233}U in the blanket region and then, after processing the blanket-salt extract the ^{233}U, and direct the fissile material into the fuel-salt stream for use in sustaining reactor operations.

While other fuels could be used for startup, startup with 20% enriched uranium is estimated to require about 600 to 800 kg of ^{235}U. Based on 0.2% tails the uranium resource requirement for this startup load is estimated to be between 150 and 200 short tons of U_3O_8. This is also the U_3O_8 resource requirement for the plant lifetime since this concept is predicted on a non-refueling cycle. Thus, the plant will require a factor of 30 to 40 less uranium over a 30-year plant life compared to a LWR once-through fuel cycle.

2.1.6 Mechanical and Thermal-Hydraulic Considerations

Mechanical and thermal-hydraulic considerations for the HMSSR, as compared with the DMSR would differ significantly in the following respects:

- Thermal hydraulics of the fuel and blanket salts inside of the HMSSR reactor vessel would be different because of the lack of graphite core members.
- There would be two salt circulating loops rather than one as in the DMSR.
- The requirement to mechanically separate the fuel and blanket salts.
- Avoidance of criticality in fuel and blanket salt draining arrangements.

2.1.7 Materials Selection and Resources

The materials of construction of an HMSSR appear to be identical with those of the DMSR with the exception that the HMSSR uses beryllium in the fuel and blanket salts as a moderator whereas the DMSR uses graphite core members. Thus, the uncertainties associated with the DMSR moderator are not applicable to the HMSSR. However, the HMSSR design uses an inner core shell, not found in the DMSR. This inner core shell is to be fabricated of fiber-reinforced graphite. As stated in Ref. 2.1-1, "Since the fiber-reinforced graphite has a much different structure than nuclear grade graphite, the ability to develop a fiber-reinforced graphite vessel of suitable radiation resistance is open to question." The suggested alternate material for this core shell is Hastelloy N, which would have satisfactory radiation resistance but would degrade the reactor neutronics to some extent.

Sources of supply must be developed for the material of the fuel and blanket salts. Although little beryllium is used in present-day reactor technology, supplies have been developed in the past and should be available for this application.

2.1.8 Engineering and Operability

Plant operations are assumed to be conducted in much the same manner as those of the DMSR. Again, as in the case of the DMSR, work is required to develop control modes to define the interactions of the reactor plant with the generator plant and to analyze the transient conditions and various other interrelations with the reactor plant.

Startup of the plant will require the addition of heat to obtain the molten salt state required to circulate the fuel and blanket salts through the reactor. Technology for this is relatively straightforward.

All maintenance and inspection operations involving the reactor vessel will have to be carried out remotely as the reactor cell of an HMSSR plant will be impossible to enter once operations of the reactor are begun.

The graphite inner core shell has a suggested design lifetime of just 3-1/2 years, and, therefore, a means of replacing the inner core shell must be developed as a standard maintenance procedure. A possible problem, which is unique to the HMSSR, is a faulty inner core shell whereby it would be possible for the fuel salt and blanket salt to become mixed. Upon remedying a faulty inner core shell, the various salts must be separated from one another in order to regain the proper fuel and blanket salt compositions to resume reactor operations.

2.1.9 Licensing and Safety

The issues and potential areas of concern for the DMSR concept will probably form the core of the evaluation of an HMSSR. One significant difference between the HMSSR and DMSR for safety and licensing is the assurance of a positive shutdown mechanism upon draining the salt solutions. In the DMSR design this shutdown mechanism is the separation of the fuel salt from its fixed graphite moderator in the core. In the HMSSR design the beryllium moderator is a constituent of the fuel and blanket salts. Therefore another shutdown mechanism must be devised. Acceptable ways of providing shutdown margin such as geometry control and/or use of fixed poisons in the drain tanks appear reasonably possible.

2.1.10 Environmental Considerations

The environmental impacts of an HMSSR should be similar to those of a DMSR. Additionally, operational and accidental releases and explosive hazards resulting from the use of fluorine in the HMSSR blanket salt system must be addressed.

2.1.11 Economics

The capital cost estimate for an HMSSR is estimated to be slightly higher than that of a DMSR. In 1979 dollars, an HMSSR is roughly estimated to be between $1100 and $1500 million.

The nonfuel operation and maintenance cost of a 1000 MWe HMSSR plant will also be slightly higher than those of a comparably sized DMSR, which is approximately double that of a conventional PWR plant.

The only area in which an HMSSR appears to offer an economic savings is that of fuel cost. If, as reported in Ref. 2.1-1, the HMSSR can generate 1000 MWe at a lower reactor fuel inventory, then an economic advantage may be generated. However, the total generating costs of an HMSSR may still not be lower than for the DMSR.

2.1.12 Commercial Feasibility

Assessment of the commercial feasibility of the HMSSR indicates that it should not differ much from that for the DMSR. Small differences in the economics, as reported in Section 2.1.11, could have minor effects on the commercial feasibility.

2.2 GRAPHITE-MODERATED HETEROGENEOUS GAS CORE REACTOR

2.2.1 Conceptual Plant Design

2.2.1.1. Summary

The Graphite-Moderated Heterogeneous Gas Core Reactor (HGCR) is similar to the Mixed Flow Gas Core Reactor (MFGCR). The basic features of this concept as described in Ref. 2.1-1 which differ from those of the MFGCR include:

- UF_6-He fuel gas mixture in lieu of UF_6-CF_4 gas mixture.
- Medium enrichment versus low enrichment fuel cycle for the MFGCR.
- Less uranium mass in core than MFGCR.
- Small separate regions of moderator and coolant channels versus large separate regions of fuel and moderator for the MFGCR.
- Graphite moderator, with alternates of light water or heavy water versus beryllium for the MFGCR.
- Improved neutron economy, fuel economy, heat transfer characteristics, power density, and power distribution over the MFGCR.

Some of the advantages of gas core reactors versus solid fuel reactors include:

- The gaseous fuel can be used as the core coolant and operated at comparatively low pressures.
- Due to the high operating temperature of the fuel, the gaseous core reactor overall plant thermal efficiency can be higher than for LWRs as the result of being able to use high-temperature, high-pressure steam.
- Unlike solid fueled reactors, there are no significant problems associated with fuel rod swelling, hydriding and fuel melting. The gas core reactor can achieve much higher levels of fuel utilization than present day designs. Almost all the fissile material can be utilized by blending depleted fuel with fresh fuel.
- Since the fuel is in the gaseous form, its fabrication is eliminated and reprocessing and waste disposal simplified.

- A gas core reactor with circulating gaseous fuel makes on-line fueling possible.

- The allowable response time required for actuation of the emergency core cooling system would be increased by the inherent safety of an expanding gas which cools itself as it expands.

Reference 2.2-1 provides a brief description of the HGCR. Specifications and operating conditions are also presented for comparison with other reactors. Reference 2.2-1 was prepared to suggest that the HGCR merits an in-depth investigation as an alternative nuclear power system.

Initially, the HGCR research program focused on H_2O and D_2O moderated systems. Most of the recent efforts, however, have been on graphite moderated configurations. The latter is claimed to be superior to the H_2O-moderated configurations from the standpoint of fuel utilization, overall thermal efficiency, safety and usage of developed technology. The research program presented in the proposal has as its primary objective the investigation of the graphite-moderated HGCR.

2.2.1.2. Plant Arrangements

The plant arrangement of the HGCR will be very similar to the MFGCR. Arrangement of the reactor vessel, intermediate fuel piping, heat exchangers, and circulators should be similar to the DMSR.

2.2.1.3. Reactor Design

Construction of the reactor vessel will be simpler for this concept than the MFGCR. This concept consists of an array of graphite moderator cells. The graphite will be contained by niobium tubes and the core barrel will be constructed of niobium alloy. The niobium metal would have to be clad with nickel sheet or Hastelloy G to withstand the fuel gas. This change would effect the neutronics of the core. Suitable materials will be needed to contain the molten salt blanket. Fig. 2.2-1 shows a simplified sketch of a cross section through the reactor vessel for the H_2O moderator/coolant HTGR concept. Drawings of the graphite moderated concept were not prepared for Ref. 2.2-1.

2.2.1.4. Primary Heat Transport System

Similar to the MFGCR concept, a research and development program is required for an intermediate heat exchanger. A suitable design is also needed for gas tight seals for the primary circulators. Development of remote maintenance techniques for major and minor equipment in the facility would be a major undertaking.

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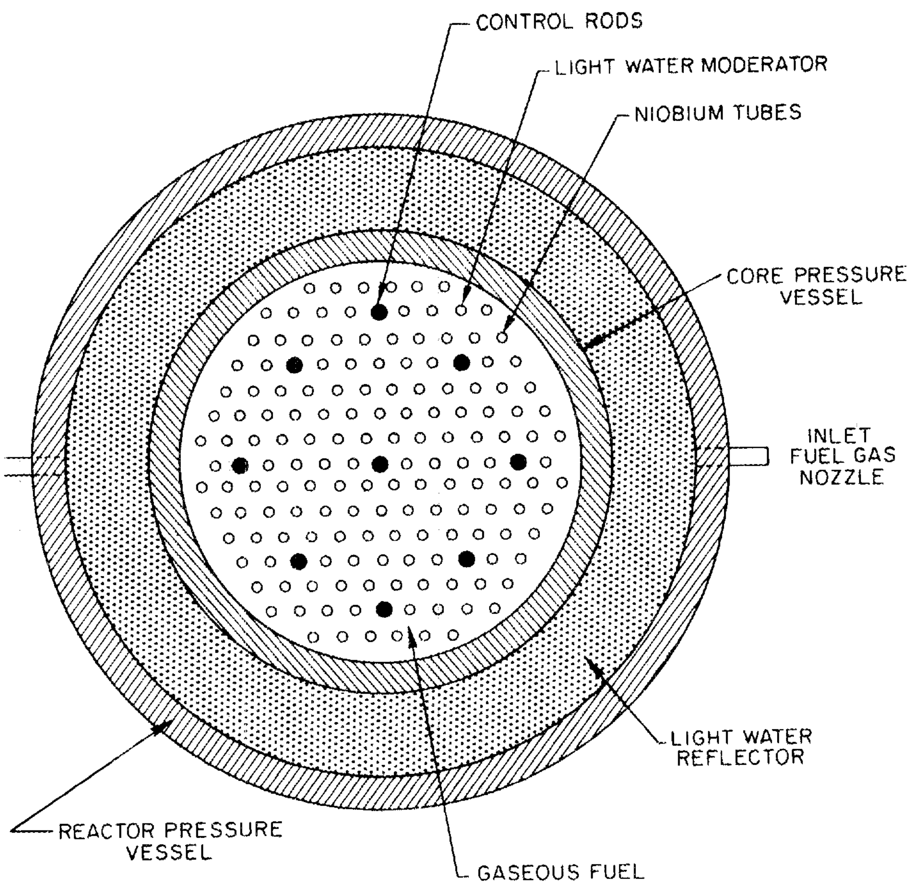

Fig. 2.2-1. HGCR Plan View.

2.2.1.5. Secondary Heat Transport System

As for the MFGCR concept, the equipment required for the secondary heat transport system has been developed and is operating in a 330 MWe HTGR station. This equipment would have to be scaled up for a 1000 MWe unit.

2.2.1.6. Containment Systems

None of the basic differences between the HGCR and the MFGCR are expected to significantly affect the containment requirements.

2.2.1.7. Radioactive Waste Control Systems

Other than consideration of the tritium-forming reaction associated with the presence of helium in the fuel gas mixture, radwaste requirements should not differ greatly from those of the MFGCR.

2.2.1.8. Engineered Safety Features

Engineered safety features systems will have to take into account the presence of tritium as a radiological hazard. Other than this, requirements for engineered safety features systems for the HGCR and the MFGCR are basically the same.

2.2.1.9. Auxiliary Systems

Auxiliary system requirements and design for the HGCR remain to be established. Such systems would not be expected to differ greatly from those for the MFGCR except that provision for tritium extraction would be required.

2.2.1.10. Plant Structure and Shielding

Requirements for plant structures and shielding, which are currently not defined, should be essentially the same as those for the MFGCR.

2.2.1.11. Steam and Power Conversion System

A modern, conventional turbine-generator with high-pressure, high-temperature steam and regenerative feedwater heating is applicable to the concept.

2.2.2 Technology Status and Research, Development, and Demonstration

Similar to the MFGCR, the HGCR is in a primitive stage of development. The concept is essentially a proposal for future study. Before it can be developed into a practical power plant, basic research must be performed to demonstrate the feasibility of the concept. The principal technology difficulty is to demonstrate that the materials of construction can withstand the fuel gas at elevated temperature and pressure. In addition, the reactor physics with the core materials and fuel gas mixtures proposed must be proven to be sound, by experimental programs. Additional difficulties are related to fuel gas cleanup, blanket processing, system dynamics and control, safeguards and analysis and specific component and system design.

Using the estimated MFGCR program as a guide, the development of this concept should take at least 52 years and cost approximating 10 to 15 billion dollars. (See Appendix A of Ref. 1.0-1 for estimating methods and uncertainties.)

2.2.3 Proliferation Resistance Features

While the proliferation resistant features of the HGCR have not been investigated in detail, they are expected to be similar to the MFGCR proliferation resistant features.

2.2.4 Reactor Physics Considerations

As part of the research program the following investigations have been proposed:

Basic Reactor Physics Calculations

Core sizing and power density studies need to be extended. Monte Carlo calculations need to be carried out for the HGCR "inverted cells." Energy deposition studies in the moderator and structure due to neutron and gamma rays must be performed.

Fuel Cycle Analysis

Fuel cycle studies in which the UF_6 gas pressure in the core is slowly increased so as to help compensate for burnup effects are proposed. Studies are proposed to optimize conversion ratio and fuel utilization and minimize plutonium discharge. Studies on the blanket regions are proposed. Also, it is proposed to extend the computations from one-dimensional diffusion theory to include two dimensional calculations.

Reactor Dynamics

Calculations for reactivity coefficients need to be broadened to include a wide range of temperatures, pressures, and fuel compositions. Dynamics calculations must be expanded to include not only temperature and density feedback, but also mass flow feedback and delayed neutron feedback. Dynamic studies need to be extended to include not only the point reactor model, but also space-dependent kinetics models.

The niobium metal selected for the core tube and container materials will not withstand the fuel gas mixture. Other materials of construction will have to be used, and these may affect neutronics in an adverse way (e.g., lower neutron efficiency).

Since the reactor physics model is still in development, this introduces uncertainty in the predicted reactor mass flows, fissile inventories, and fuel cycle economics.

2.2.5 Fueling Alternatives and Resource Utilization

As for the MFGCR, the HGCR has many basic options. Some of these options are as follows:

- Operation with highly enriched uranium.

- Operation as a denatured uranium burner (provided PuF_6 can be stabilized in the gas phase) with improved fuel utilization.

- Operation as a high-performance converter or breeder with a physically separated fertile blanket system.

- Operation with both liquid or solid moderators.

Gaseous fuel can be added to the primary loop to keep the system functioning without necessitating reactor shutdown. Similar to the MFGCR, the HGCR has better fuel utilization and requires substantially less uranium to sustain 30 years of operation than a standard LWR using a throwaway fuel cycle. However, the fuel utilization of the HGCR relative to the MFGCR suffers somewhat from the fact that a higher enrichment of uranium is required.

2.2.6 Mechanical and Thermal-Hydraulic Considerations

As in the case of the MFGCR considerable research still needs to be performed before a final design is selected. As part of the research program the following has been proposed:

- Extend preliminary studies of velocity, temperature and pressure distribution in a typical unit cell of the core.
- Investigate the thermal problems between the peripheral unit cell elements and the core containment walls.
- Conditions at the entrance and exit region of the core must be analyzed.
- Analysis of both primary and secondary heat exchangers.

In addition, some other areas which should be addressed are moderator cooling, blanket cooling, and the effects of density gradients on the power generation within the core.

2.2.7 Materials Selection and Resources

The use of the intended niobium zirconium material for the reactor core coolant tubes and containment material in contact with a uranium hexafluoride (fuel gas) environment is probably unsatisfactory. This material in the presence of the gas, would rapidly change to niobium pentafluoride (NbF_5). Niobium pentafluoride has a melting point of $158^{\circ}F$, a boiling point of approximately $428^{\circ}F$, and is very soluble in water.

The potential for decomposition of plutonium hexafluoride presents another materials problem. In the isotropic composition of the UF_6 gas of the core, some ^{238}U exists which will lead to the production of ^{239}Pu. This plutonium could be removed from the process or consumed by fission. However, PuF_6 decomposes into $PuF_4 + F_2$ at high temperatures. Such decomposition of PuF_6 within the gaseous core reactor would result in removal of the plutonium by precipitation of PuF_4 on container surfaces. The volatility of PuF_4 is too low for any significant amount of the compound to remain in the gas phase in the temperature range of the gaseous core reactor. As elemental fluorine gas is a product of the PuF_6 a sufficiently large partial pressure of fluorine in the gas of the reactor core may prevent the decomposition. Data are not available to determine the partial pressures required at the operating temperatures of the gaseous core reactor. This will have to be developed by tests. The addition of free fluorine to the fuel gas will cause serious material problems. The niobium metal used for the vessel and tube materials of the core cannot withstand the free fluorine at high temperatures.

Because of the above two problems other reactor materials will have to be used. To protect the niobium from the fuel gas, it may be possible to clad the material with thin nickel sheet or Hastelloy G. Hastelloy G is a Ni, Cr, Mo, Fe alloy that is highly resistant to fluorides and fluoride gas. The cladding could be done by explosive bonding techniques. Cladding would be much better than electroplating or vacuum deposition techniques because it would be free from pin holes.

Three manufacturers of niobium alloys were contacted and stated that they were capable of fabrication of the core coolant tubes in lengths of 8 to 10 feet. The tubes would be made either seamless or welded. The sheet material for the core barrel would have to be made from an ingot and then rolled or forged to size. Sheets approximately 6" x 40" x 24" are feasible. Joining would be possible by tungsten inert gas (TIG) or electron beam welding with TIG being the more practical, but requiring a great deal of weld end preparation.

Material problems for this concept are similar to those discussed for the MFGCR.

2.2.8 Engineering and Operability

Engineering and operability requirements for the HGCR should be basically the same as those for the MFGCR except for the consideration of tritium production in the helium component of the fuel gas mixture and its removal.

Similar to the MFGCR, the toxicity and explosive nature of the free fluorine in the fuel gas will make this plant unattractive to operate from both the plant operator's and public's viewpoint. Protective outergear as well as breathing apparatus are required for any maintenance operation. Development and testing of maintenance methods and design of primary system equipment to facilitate maintenance remotely will be a major undertaking in the facility design.

2.2.9 Licensing and Safety

As for the MFGCR concept, the degree of difficulty encountered in licensing the HGCR will be dependent upon the identification and satisfactory completion of all necessary research and development programs. These programs must address all uncertainties in plant design, the consequences of postulated accidents, and verification of plant performance under normal and accident conditions.

Continuous fission product removal during gas cleanup operation largely reduces the inventory of radioactive fission products. This has a very beneficial impact on safety. However, the consequence of small or large breaks in the primary system involves dispersing the entire fissile inventory throughout the containment. A double containment system, and possibly an additional level of containment, may be necessary to achieve release rates equivalent to light water reactors during normal operation and loss of coolant accidents.

2.2.10 Environmental Considerations

Similar to the MFGCR concept, the HGCR will reject less heat to the environment than does a similar sized LWR. Plant efficiency is approximately 45% versus 33% for LWRs. Other environmental considerations are the same as in the MFGCR concept.

2.2.11 Economics

Similar to the MFGCR, the total facility costs for the HGCR without interest during construction are estimated to be between 50-100% higher than for a commercialized LWR. Due to remote maintenance and operation of the HGCR, the operating and maintenance costs are estimated to be at least double those of the LWR. The levelized 30-year fuel cycle costs for the HGCR are suspected to be substantially lower than for the LWR and slightly lower than the MFGCR. The overall affect on total generating cost should make the HGCR cost similiar to those of the MFGCR.

2.2.12 Commercial Feasibility

Commercial feasibility for the HGCR would be basically the same as that for the MFGCR.

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Senior Program Assistant
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Nuclear Energy Division
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Germantown, MD 20767

95. Dr. Raymond J. Braitsch
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Energy Technology, Office of Planning and Technology Transfer
U.S. Department of Energy
Rm. 7117
Cassimir Pulaski Bldg.
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U.S. Department of Energy
Room A-220
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Room A-23204
Germantown, MD 20545

99. Dr. Charles E. Till
Director, Applied Physics
Argonne National Laboratory
9700 South Cass Avenue
Argonne, IL 60439

100. Mr. Walter Lipinski
Project Manager
Argonne National Laboratory
9700 South Cass Avenue
Argonne, IL 60439

101. Mr. Charles E. Klotz
Director, Program Management Support
Argonne National Laboratory
9700 South Cass Avenue
Argonne, IL 60439

102. Dr. Herbert J. C. Kouts
Chairman, Dept. of Nuclear Energy
Brookhaven National Laboratory
Associated Universities, Inc.
Upton, NY 11973

103. Dr. Walter Y. Kato
Principal Investigator
Brookhaven National Laboratory
Associated Universities, Inc.
Upton, NY 11973

104. Dr. Ronald P. Omberg
Manager, Advanced Concepts
Westinghouse Hanford Company
Hanford Engineering Development Lab.
P.O. Box 1970
Richland, Washington 99352

105. Mr. James M. Williams
Los Alamos Scientific Laboratory
P.O. Box 1663
Los Alamos, New Mexico 87545

106. Mr. Robert J. Sorenson
Project Manager
Battelle-Pacific Northwest Lab.
P.O. Box 999
Richland, Washington 99352

107. Mr. Veikko Uotinen
Project Manager
Babcock & Wilcox Company
P.O. Box 1260
Lynchburg, VA 24505

108. Mr. Bruch A. Pasternack
Research Director and Director
of Energy Management Strategies
Booz-Allen and Hamilton
4330 East-West Highway
Bethesda, MD 20014

109. Mr. Samuel Zwickler
Project Engineer Security
Officer
Burns & Roe Industrial Services Corp.
P.O. Box 663
Route 17, South
Paramus, NJ 07652

110-124. Mr. George Nugent
Principal Investigator
Burns & Roe, Inc.
185 Crossways Park Drive
Woodbury, NY 11797

125. Dr. Norton Shapiro
Manager, Advanced Design Projects
Combustion Engineering, Inc.
1000 Prospect Hill Road
Windsor, Connecticut 06095

126. Mr. Russell Crowther
Project Manager
General Electric Company
175 Curtner Avenue
San Jose, CA 95125

127. Dr. Jack N. Barkenbus
Political Scientist
Institute for Energy Analysis
Oak Ridge Associated Universities
P.O. Box 117
Oak Ridge, TN 37830

128. Mr. Steven Shantzis
International Energy Associates, Ltd.
2600 Virginia Ave., NW, Suite 200
Washington, D.C. 20037

129. Dr. Michael J. Driscoll
Massachusetts Institute of Technology
Dept. of Nuclear Engineering
138 Albany Street
Cambridge, Massachusetts 02139

130. Dr. Marvin M. Miller
Massachusetts Institute of Technology
Room 24-207
Dept. of Nuclear Engineering
Cambridge, Massachusetts 02139

131. Dr. Norman C. Rasmussen
Massachusetts Institute of Technology
Room 24-102
Department of Nuclear Engineering
Cambridge, Massachusetts 02139

132. Mr. Jack N. Sorenson
NUS Corporation
4 Research Place
Rockville, MD 20850

133. Dr. Edward Straker
Science Applications, Inc.
1205 Prospect Street
La Jolla, CA 92038

134. Dr. Gerald Atkinson
Science Applications, Inc.
8400 Westpark Dr.
McLean, Virginia 22102

135. Mr. Michael J. Higgins
Vice-President for Energy and
Environment
JRB Associates
8400 Westpark Drive
McLean, Virginia 22102

136. Mr. Sidney M. Stoller
President-Project Manager
S.M. Stoller Corporation
1250 Broadway 29th Floor
New York, NY 10001

137. Mr. John McAdoo
Project Manager, Assistant Mgr.
Nuclear Safety Department
Westinghouse Electric Corp.
P.O. Box 355
Pittsburgh, PA 15230

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