# The potential of the molten salt reactor for warship propulsion

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**SYNOPSIS**

*This paper outlines the investigations into molten salt reactors undertaken at the Oak Ridge National Laboratory (ORNL) in the 1960s and 1970s. The advantages of the thorium cycle are described and the reason why the work was not taken further is given.*

*The paper outlines the thorium cycle and assesses its potential for warship propulsion by illustrating how a medium sized surface warship might be powered by a reactor plant based on the molten-salt demonstration reactor plant designed by ORNL.*

## INTRODUCTION

Nuclear power for submarine propulsion using moderately enriched uranium fuel in a Pressurised Water Reactor (PWR) has a long and highly successful history. But there are disadvantages: the high pressure demands high mechanical strength in large components which must be maintained through life; while the moderate operating temperature limits plant efficiency.

In comparison, thorium, used in a suitably designed reactor plant:

a. Operates at high temperature and low pressure simplifying the mechanical design and yielding increased thermodynamic efficiency with consequent reductions of component sizes.
b. Offers a much improved fuel cycle with fewer and less troublesome radioactive waste products.
c. Is much more abundant than all uranium isotopes.

The paper that now follows is split into two sections. The first summarises information that is available in the public domain and does not intend, or pretend, to offer new information or insights. Section 2 provides the authors' view on how a thorium molten salt reactor might be used to provide power for a medium sized surface combatant of around 8,000 te displacement.

## *Part 1: The Historical Context*

### INTRODUCTION

The thorium reactor has a history almost as long as the submarine PWR. Both had their origins in the Oak Ridge National Laboratory (ORNL).

#### US Air Force Reactor Developments

Under Dr Alvin Weinberg, ORNL successfully built and operated the Aircraft Reactor Experiment (ARE) reactor, to investigate the use of molten fluoride fuels for aircraft propulsion reactors. It used the molten fluoride salt $NaF\text{-}ZrF_4\text{-}UF_4$ as fuel, was moderated by beryllium oxide (BeO), used liquid sodium as a secondary coolant and had a peak temperature of 860 °C. It operated for a 1000-hour cycle in 1954. It was the first molten salt reactor.

In 1951, the US Air Force and the Atomic Energy Commission (AEC) had established the joint AEC/USAF Aircraft Nuclear Propulsion programme. It had two strands, the Direct Air Cycle concept, which was developed by General Electric, and the Indirect Air Cycle which was assigned to Pratt & Whitney. Both types were to use small reactors based on the ARE but both were cancelled by President Kennedy in June 1961 [1].

### The Molten Salt Reactor

When the USA military aircraft nuclear propulsion programme was cancelled, ORNL redirected its focus to a civilian version of the meltdown-proof molten salt reactor, aiming to use the unique characteristics of the Thorium cycle to design breeder reactors. The Molten Salt Reactor (MSR) was known as the "chemist's reactor" because it comprised a chemical solution of melted compounds containing the actinides (uranium, thorium, and/or plutonium) in a carrier salt. The carrier salt is typically composed of beryllium-fluoride ($BeF_2$) and lithium-fluoride (LiF). The lithium is isotopically enriched in Lithium-7 to prevent excessive neutron capture or tritium production. The Thorium cycle is described in Appendix 3.

Weinberg was removed from ORNL in 1973 after 18 years as the laboratory's director because he continued to advocate Molten Salt Reactors (citing greater nuclear safety) instead of the Liquid Metal Fast Breeder Reactor (LMFBR) chosen by the head of the AEC. This not only adversely affected development of the MSR but brought to an end all work exploring the use of thorium as a reactor fuel.

### THE MOLTEN-SALT REACTOR EXPERIMENT [2]

By the end of 1959, ORNL's engineering developments had proceeded to the point that justified a molten salt reactor experiment (MSRE). Having a power less than 10 $MW_t$, the AEC accounting rules allowed the use of operating funds. A higher power reactor would have required a capital appropriation, limiting the freedom to make changes. To keep the reactor simple, only the fuel stream of a 2 fluid breeder reactor was simulated, so no thorium fluoride was included. The MSRE is described in Appendix 1.

Design started in 1960 and construction started at the beginning of 1962. The reactor went critical in June 1965 and operation was terminated in 1969 so that funds could be applied to other developments. During the 4 years of operation, many features of molten salt behaviour and management were explored.

ORNL concluded [3]:

> *"The MSRE has shown that salt handling in an operating reactor is quite practical. The salt chemistry is well behaved, there is practically no corrosion, the nuclear characteristics are very close to predictions, and the system is dynamically stable. Containment of fission products has been excellent and maintenance of radioactive components has been accomplished without unreasonable delay and with very little radiation exposure.*
>
> *The MSRE is stable and self-regulating with regard to changes in heat load, with a response that becomes quicker and more strongly damped as the power level is increased. Responsible in large part for this behaviour are the strong negative temperature coefficients of reactivity associated with both the fuel salt and the graphite moderator. The system is quite simple to control."*

### THE DESIGN OF A MOLTEN SALT REACTOR DEMONSTRATION PLANT

The MRSE demonstrated the feasibility and investigated aspects of the chemistry, engineering and operation of molten salt reactors. Originally, the ORNL plan had been to follow the MRSE with a Molten Salt Breeder Experimental plant (MSBE) having all the technical features of a high performance breeder on an intermediate scale, generating 150 $MW_t$ from a supercritical steam plant and possessing the fuel reprocessing facilities required for a breeder. This in turn would be followed by a Molten Salt Breeder Reactor (MSBR) itself.

However, an alternative approach was to demonstrate the concept on a semi-commercial scale without developing the basic technology beyond the stage successfully demonstrated in the MSRE. Hence, when molten salt breeder reactor development ceased at ORNL, a Molten Salt Demonstration Reactor (MSDR) [4] had been designed but was never built. The MSDR is described in Appendix 2.

# *Part 2: The Warship Application and Concept Design*

## INTRODUCTION

As long as dieso remains widely available and affordable, it will continue to be the preferred fuel for warships. In due course, however, this situation will change and it will be necessary to adopt some alternative fuel. The possibilities are:

a. To use Liquid Natural Gas (LNG)
b. To convert coal to dieso, or equivalent liquid fuel.
c. To use liquid derivatives of shale gas or shale oil, which are predicted to become widely available and inexpensive.
d. To switch to nuclear power.

To examine the potential of the thorium reactor for warship propulsion it was decided to base the study on a 136$MW_t$ reactor delivering 50 $MW_e$. This was considered to be the most appropriate power, taking account of the weight and volume of shielding and collision protection required. Studies in the 1960s showed that these considerations determined that the minimum viable size for a nuclear powered surface warship is 8000te. For such a size, 40 $MW_e$ of propulsion power, leaving 10 $MW_e$ for ships services, would yield a top speed of approximately 28 knots.

## A THORIUM MOLTEN SALT REACTOR FOR SURFACE WARSHIP PROPULSION

While several designs of thorium reactors have been proposed, it was decided to base this study on designs by ORNL since this is the only organisation whose work is available to have designed, made and operated thorium liquid salt reactors. This paper looks at two in particular: the MSRE of 7.5 $MW_t$ [Appendix 1] which operated successfully in 1966 to 1969 including, latterly, 2,500 equivalent full power hours using $U^{233}$ as the fissile component of the primary salt; and the 750 $MW_t$ MSDR [Appendix 2].

While this latter design was never built or operated, it is described in considerable detail in papers that are now available and it incorporates many features that result from the experience of operating the MSRE as well as those resulting from the 100-fold increase in power.

Hence to produce the sketch design of a 136 $MW_t$ reactor for a surface warship, the choice had to be made between scaling up the MSRE, or scaling down the MSDR design.

It was decided to scale down the MSDR design, because:

a. It represents the latest design by the ORNL team and incorporates the practical lessons learnt from operating the MSRE.

b. Designed as a power reactor, it incorporates more of the features relevant to the warship application.

c. It is a simplified design, lacking provision for removing fission product poisons other than xenon and krypton. The design envisages changing the primary salt for reprocessing ashore after 8 years' operation.

d. The power density is made sufficiently low for the graphite core to last 30 years. This accords with warship lifetimes.

The warship sketch design differs from MSDR in the following respects:

a. The MSDR has primary, secondary and tertiary salt systems. The purpose of the tertiary system is to act as a tritium trap, using a commercial salt called Hitec. The warship plant relies upon the helium/Nitrogen working fluid of its closed cycle gas turbine generators (CCGTs) to capture tritium and hence has only primary and secondary salt systems. Tritium removal is discussed in Appendix 4.

b. The MSDR has 3 salt chains, each comprising a primary salt pump and system, a secondary pump and system and a tertiary pump and system. In each chain there are 2 primary salt/secondary salt heat exchangers and 2 secondary salt/tertiary salt heat exchangers, making 12 heat exchangers in all. The rationale for this duplication of heat exchangers is given as:

> *"There are two heat exchangers in each leg in order to make the fabrication of these units more practical."* [5]

In the warship plant design, illustrated in Figure 1, at one seventh of the power, just one heat exchanger is provided in each primary loop.

c. Furthermore, rather than having 3 salt chains, the warship plant has two, so there are just 2 salt-to-salt heat exchangers.

d. The MSDR drain tank, which is continuously receiving primary salt and entrained gases from the centrifugal gas strippers in the primary pump by-passes, is cooled by natural circulation of Sodium-Potassium Alloy (NaK) to tubes in a water tank heat sink. In the warship design, the opportunity is taken to use this drain tank waste heat, in part, to power two organic Rankine cycle electrical generators to provide auxiliary power which continues to be available even after a complete reactor shut down, when molten salt systems drain down to their respective drain tanks.

e. As an ultimate protection against molten salt drain tanks cooling and solidifying, thereby preventing a restart, an emergency tank of dieso is provided, which can be used to heat the molten salt compartments.

Figure 1 below showing the salt systems does not include the off-gas system.

**Salt system heating**

All salt piping and vessels must be heated to prepare for salt filling and to keep the salt molten when there is no nuclear power. In the MSRE the pipes and components of the two salt systems were heated electrically. In the much larger and more compact MSDR with its three salt systems and numerous heat exchangers the reactor and heat exchanger compartments are heated by circulating the compartment atmosphere, which is nitrogen, over electrical heaters. Circulation is by three large blowers discharging gas at 566 $^{0}$C into the reactor compartment which has outlets to the heat exchanger compartments and the drain tank compartments.

The warship design follows MSDR heating philosophy and one of the main challenges facing the naval architect is to incorporate into the hull design the interconnected compartments containing the reactor plant and auxiliary systems which require a large temperature range: from an operating temperature of 538 $^{0}$C down to ambient conditions, when access by remote operated tooling is required for maintenance. The compartment boundaries incorporate thermal insulation, radiation shielding and external cooling.

**Generating Plant**

The MSDR has a single large conventional steam turbine, with HP, IP and 2 LP turbines on a common shaft. Reheat by a tertiary salt heat exchanger is provided between HP and IP stages. For the warship design, 4 schemes were considered: steam turbo-generators (as MSRD); open cycle gas turbine generators; combined cycle gas and steam turbine generators; closed cycle gas turbine generators (CCGTs).

CCGTs were chosen, two for each of the two salt chains, using a 80/20 by mass helium/nitrogen mix because:

a. The CCGT plants are compact units that can be fully built and tested before being fitted in the ship.
b. The helium/nitrogen working gas allows the units to be physically smaller than steam turbines or open cycle gas turbines of the same power.
c. The CCGT Brayton cycle can use higher cycle temperatures more readily than the other options and therefore yields the highest system efficiency. This is further explained in Appendix 4. A 36% overall efficiency is achievable with the CCGT / molten salt combination and this value has been used in the naval architecture studies and warship sketch design.

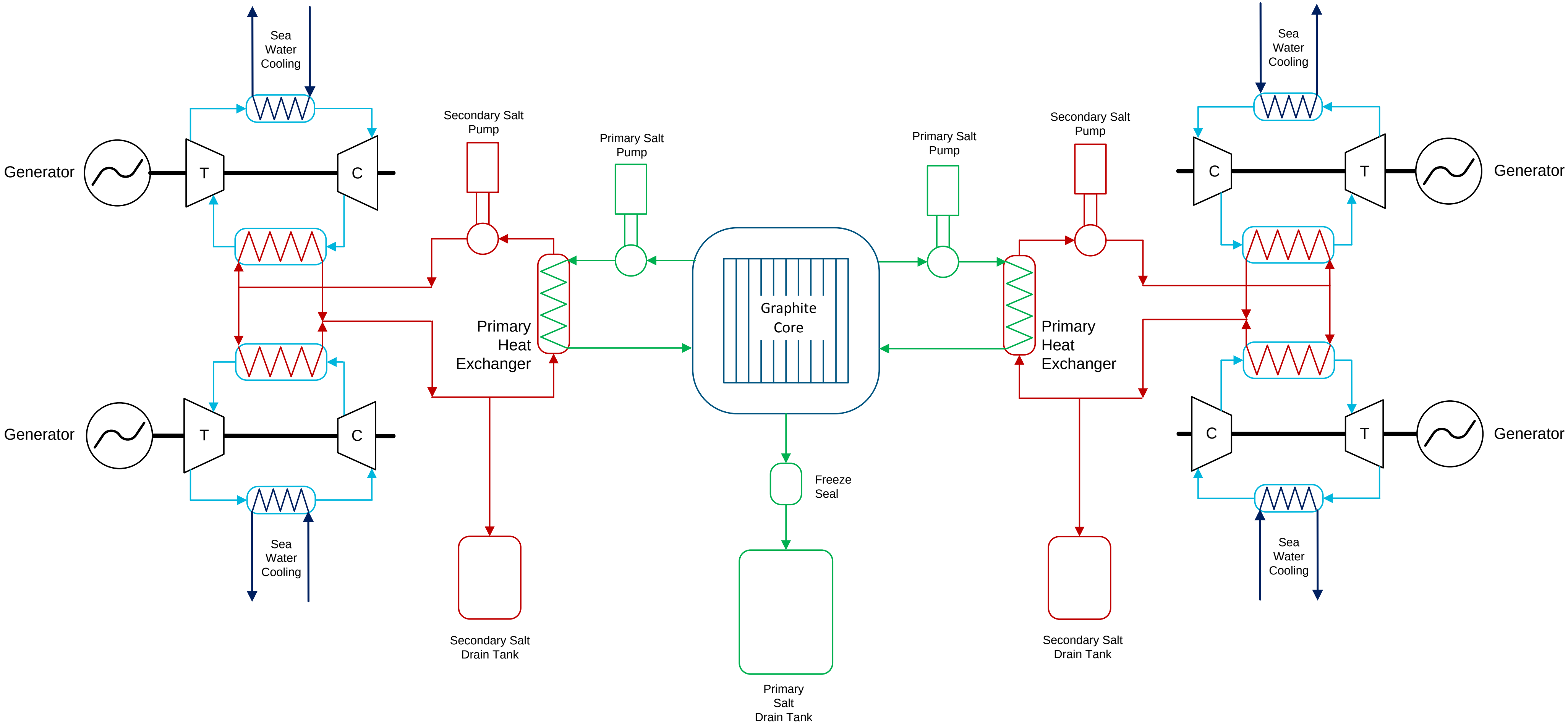


**Figure 1: Diagrammatic Plant Layout**

## NAVAL ARCHITECTURE

A destroyer of 8000 te displacement was the warship chosen to receive the propulsion plant powered by a molten salt reactor modelled on the MSDR.

### Arrangement

Priority was given to siting the reactor plant amidships and as low as feasible, for reasons of: stability; minimising ship motions and whipping effects; and survivability. Effort was made to maximise the standoff from the ship's hull. This was complicated by the requirement for the fuel drain tanks to sit vertically below the reactor vessel and salt chains to facilitate drainage of the salts by gravity, in the event of a reactor shutdown. This was a significant driver in the location of the plant.

The selection of a reactor as the source of primary power provision lends itself to Integrated Full Electric Propulsion (IFEP), allowing the prime movers to be located more conveniently with respect to the reactor. The selection of CCGTs also removes the significant driver, in conventional ship design, of uptakes and downtakes. This frees up additional space within the hull, and "real estate" on the top side needed for weaponry and sensors.

### Structure

Structurally significant are the inclusion of longitudinal bulkheads, added for the purposes of mounting the reactor plant and providing a boundary for thermal insulation, also improving survivability characteristics.

### Weight

Relative to a conventional ship power plant, having a reactor causes significant weight increase due to:

a. The reactor itself.
b. Supporting sub-systems.
c. The associated shielding.

In this design another significant weight is that of the longitudinal bulkheads. These are a necessary addition, but uncommon in a ship of this size. These weight increases have in part been offset by removing diesel generators and their fuel saving around 1000 te.

### Stability

The reactor, being below the typical vertical centre of gravity (KG) for such a ship, reduces the overall KG of the platform. However, the removal of diesel fuel from an even lower position within the ship, causes an increase of ship KG. The net result is a small increase in whole ship KG.

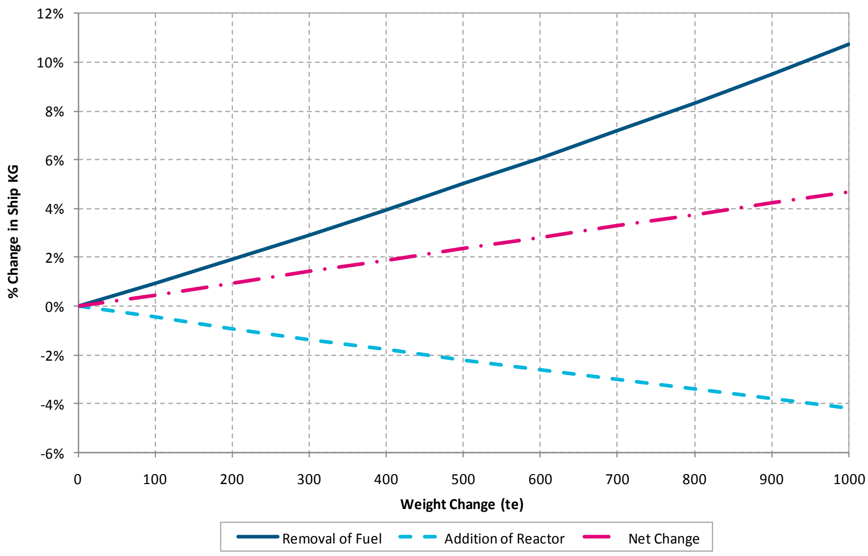


**Figure 2:Ship KG Vs Weight Change**

Obviously, this small rise in KG is a rise in the solid KG, the change in fluid KG, however, may not be as much as 5%, even at the extreme. This arises due to the lack of a free surface effect from the diesel tanks, which manifests itself as an effective increase in fluid KG.

The absence of uptakes and downtakes avoids these large down-flooding points, and improves the stability characteristics of the ship at large angles.

**Reactor**

The reactor vessel, primary and secondary salt circuits, and the fuel drain tanks are housed within a flexibly mounted fully enclosed rafted structure. This design feature was driven by two main concerns: containment of the radiologically active systems, and insulation of the heated cells.

The raft is split into 4 cells which are actively heated, as described above, to ensure the salt remains above freezing temperature. Rafting the reactor also allows a further layer of insulation. The raft sits within transverse and longitudinal bulkheads, creating a volume between the bulkheads and the shielding around the raft structure, which could either be evacuated or filled with water, for additional thermal and radiological protection.

**Salt Drain Tanks**

The salt drain tanks are situated vertically below the reactor vessel and secondary salt circuits, within the bottom cell of the reactor raft, and being actively heated require both radiological and thermal protection.

**Off Gas System**

The 47 hr Xeon hold up and 90 day delay charcoal beds, described in Appendix 2, are located adjacent to the reactor compartment, in order to minimise the length of the “off-gas” system runs.

**Prime Movers**

As described above 4 CCGTs were selected as the prime movers for the ship. They have been located in compartments adjacent to the reactor compartment in order to minimise the length of the salt runs, which need to be lagged outside the reactor compartment. Four CCGTs were chosen to provide redundancy and flexibility in operational loading and survivability.

4 closed cycle organic Rankine engines have also been included to provide auxiliary power in the event of reactor shut down, utilising heat within the cell of the salt drain tanks.

**Fire**

Compared to a conventional ship, one fire risk has been exchanged for another. The fire risk presented by diesels has been replaced by the risk posed by the hot salt. The risk of fire within the cells of the reactor compartment is negligible, as the atmosphere is Nitrogen.

**Survivability**

The differences in survivability, between a conventional ship and a thorium reactor powered ship, can be broken down into changes in susceptibility, vulnerability and recoverability.

- The ship’s susceptibility benefits from the absence of exhaust and associated infra-red signature
- The addition of longitudinal bulkheads, improves the vulnerability performance of the ship. However, the reactor stands out as a single point of failure affecting all four CCGTs simultaneously.
- The recoverability of the ship hinges on the start-up time required by the reactor after a shut down event.

**Maintainability**

Evidence from the MSRE shows it was operated successfully without unreasonable delay, especially notable as the MSRE was the first of its kind. By incorporating experience gained from the MSRE, the MSDR includes improvements to the design. The ship design seeks to increase the maintainability of the ship by siting pump motors outside the containment volume. However, salt system maintenance will require remote methods. The

ORNL investigated methods and developed tools for the remote inspection, maintenance and equipment replacements inside the radiated zones of molten salt reactors. Similar action will be required for warships.

## SHORE FACILITIES

It is envisaged that the following shore facilities would be needed: shore prototype, operator training simulator, $U^{233}$ production plant and thorium salt reprocessing plant.

## DISCUSSION

It is recognised that basing the warship design on a reactor plant developed 50 years ago, lays it open to the challenge that it is out of date and far from optimal. This is a valid criticism. The MSDR represents the very latest design development by ORNL, but the tritium removal system was untried and it constrained the plant to operate at a temperature having an adverse effect on efficiency. The warship plant assumes that an alternative tritium removal system is feasible therefore allowing the efficiency benefits of the molten salt reactor / CCGT combination to be realised but, in common with many other aspects of the design, further development is necessary. It is also to be expected that the heat exchanger developments that have occurred over the last half a century will offer further advantages in size and effectiveness.

To avoid introducing too many features having little or no provenance, the baseline warship plant is deliberately kept close to the MSDR, the changes being:

a. Those resulting from scaling down from 950 $MW_t$ to 136 $MW_t$
b. An entirely different method of tritium capture and removal.
c. Providing one primary/secondary heat exchangers in each of 2, rather than 3, chains of salt systems.
d. Substitution of a single steam turbo generator by 4 closed circuit gas turbine generators with helium or a helium-nitrogen mixture as the working fluid.
e. The use of drain tank heat to generate electricity by Rankine engine generators

Shielding remains a major uncertainty at this stage. The design given in this paper leaves 0.65 metres of space around the compartment boundary to accommodate both thermal and radiation shielding. The radiological shielding was based on the shielding space requirements of pressurised water reactors and being speculative must be further analysed.

Before it can be proved that the molten salt reactor has potential for warship propulsion the following aspects require further work:

a. Thermal and radiological shielding and insulation.
b. Overall plant heat management including drain tank cooling.
c. Tritium removal from the salt chain.

Nevertheless no insurmountable problems are foreseen.

## CONCLUSION

Basing the sketch design of a warship propulsion plant on the MSDR, it has been possible to propose a design which illustrates the potential of the thorium molten salt reactor for warship propulsion, albeit work remains to be done in virtually every aspect of the design.

## REFERENCES


1 Wikipedia “Aircraft Nuclear Propulsion” quoting from "Nuclear Powered Aircraft" (html at radiationworks.com). Brookings Institute.

2 H G MacPherson “The Molten Salt Reactor Adventure”. NUCLEAR SCIENCE & ENGINEERING: **90**, 374-380 (1985)

3 Paul N Haubenreich and J R Engel “Experience with the molten-Salt Reactor Experiment”. Article in NUCLEAR APPLICATIONS & TECHNOLOGY Vol. 8. February 1970

4 E S Bettis, L G Alexander, H L Watts "Design Studies of a Molten Salt Reactor Demonstration Plant" ORNL-TM-3832. Oak Ridge National Laboratory. 1972

5 Ibid, p26

# APPENDIX 1

# The Molten Salt Reactor Experiment

The information in this Appendix is drawn from Oak Ridge National Laboratory (ORNL) information, available in Wikipedia[1]

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The Molten Salt Reactor Experiment (MSRE) was a 7.4 MW test reactor built and operated by ORNL to verify the fission behaviour of the single-fluid molten salt reactor. Since this was a limited engineering test, the experiment did not set out to demonstrate breeding but neutron measurements around the graphite core were taken. Heat from the reactor vessel was transferred to a molten salt cooling system and shed by a radiator cooled by air blowers.

The reactor vessel and piping systems were made from Hastelloy-N[1]. The fuel was $LiF-BeF_2-ZrF_4-UF_4$ (65-30-5-0.1) and the secondary coolant was $2LiF-BeF_2$.

The operating temperature was 650 $^oC$. The plant operated successfully over a 4½ year period from June 1965 to December 1969, achieving the equivalent of about one and a half years of full power operation.

Initial criticality was achieved by adding enriched $U^{235}$ as $UF_4$-LiF eutectic to the carrier salt to make the reactor critical. In 1968, the fissile component was changed to $U^{233}$ and the MSRE became the world's first reactor to operate on this fuel.

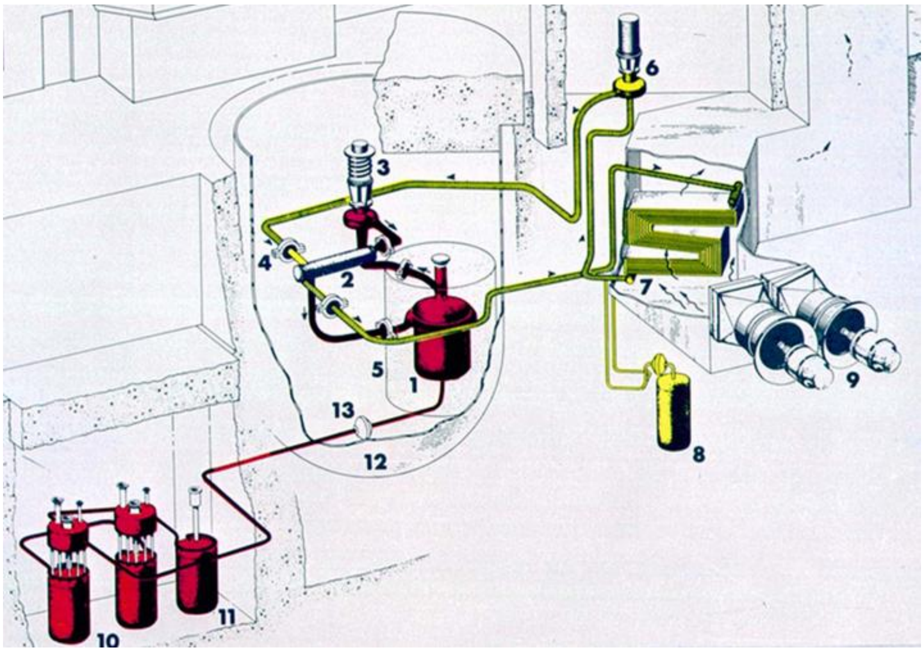


Figure 1: The Molten Salt Reactor Experiment

1. Hastelloy, the registered trademark name of Haynes International, Inc, comprises a range of 22 different highly corrosion-resistant metal alloys. The predominant base constituent is nickel to which molybdenum, chromium, cobalt, iron, copper, manganese, titanium, zirconium, aluminium, carbon, and tungsten are added in varying percentages. In addition to nickel Hastelloy-N comprises: 0.2% cobalt, 7% chromium, 16% molybdenum, 4% tungsten, 5% iron, 1% silicon, 8% manganese and 0.12% carbon.

| Item | Component | Item | Component | Item | Component |
|---|---|---|---|---|---|
| **1** | Reactor Vessel | **6** | Coolant Pump | **10** | Fuel Drain Tanks |
| **2** | Heat Exchanger | **7** | Radiator | **11** | Flush Tank |
| **3** | Fuel Pump | **8** | Coolant Drain Tank | **12** | Containment Vessel |
| **4** | Freeze Flange | **9** | Fans | **13** | Freeze Valve |
| **5** | Thermal Shield | | | | |

Table 1: Key for Figure 1

**MSRE Safety**

The MSRE employed a safety feature common to all molten salt reactors. Being molten, the fuel can be drained down to tanks where it is stored in a safe, non-critical configuration. By employing a freeze-seal in the drain pipe cooled, for instance, by a fan the reactor vessel is automatically emptied in the event of a power failure. Freeze seals were also used, if required for maintenance, to isolate the fuel pump and heat exchanger; also to permit the coolant to be drained down to its drain tank.

**Conclusion**

The MSRE demonstrated that a molten salt fuelled reactor was viable. It ran for long periods, yielding valuable information and maintenance was accomplished safely and without excessive delay. It demonstrated that fuel salt was immune to radiation damage; that the graphite moderator was not attacked by the fuel salt; and that corrosion of Hastelloy-N was negligible. (Subsequently it was found that shallow inter-granular cracking could be reduced by adding small amounts of niobium to the Hastelloy-N).

Noble gases were stripped out from the fuel salt by a spray system, thereby reducing $Xe^{135}$ poisoning by a factor of about 6. Most of the fission product elements remained stable in the salt, allowing their easy removal during processing.

Key components performed well. Handling the high melting temperature salt proved to be easy and it was found that the coolant salt density changes very little on freezing and thawing.

**REFERENCES**

1 http://en.wikipedia.org/wiki/Molten-Salt_Reactor_Experiment

# APPENDIX 2

# The design of a molten salt reactor demonstration plant

## INTRODUCTION

The information in this Appendix is drawn from Oak Ridge National Laboratory (ORNL) information [1].

The ORNL Molten Salt Reactor Experiment (MRSE) demonstrated the feasibility and investigated aspects of the chemistry, engineering and operation of molten salt reactors. Originally, the ORNL plan had been to follow the MRSE with a Molten Salt Breeder Experimental plant (MSBE) having all the technical features of a high performance breeder on an intermediate scale, possessing the fuel reprocessing facilities required for a breeder

However, an alternative approach was to demonstrate the concept on a semi-commercial scale without developing the basic technology beyond the stage successfully demonstrated in the MSRE. This was the Molten Salt Demonstration Reactor (MSDR) which was designed but never built.

## MSDR REACTOR VESSEL

The design incorporates a reactor vessel, Figure 2, 26 ft (7.9m) in diameter and height, filled with a matrix of graphite slabs forming flow passages for the molten salt ($LiF$-$BeF_2$-$ThF_4$-$UF_4$ ). The inner surface of the Hastelloy N reactor vessel has a lining of 2.4 ft of graphite as a reflector. The power density is limited so that the graphite core achieves a 30 year life and there is no provision for removing fission products on a short time scale, other than gaseous xenon and noble elements. Hence the reactor has a breeding ratio less than unity and is described as a converter (of $Th^{232}$ to $U^{233}$) as opposed to a breeder. It was intended to remove the fuel salt for reprocessing after 8 years' operation.

## MSDR SALT SYSTEMS

The design incorporates 3 salt systems. The fuel salt is circulated through heat exchangers transferring heat to the secondary system of salt of the same composition as the fuel salt carrier ($LiF$-$BeF_2$). Secondary salt is circulated through another set of heat exchangers transferring heat to the tertiary system containing a eutectic mixture of $KNO_3$-$NaNO_2$-$NaNO_3$, which has the commercial name "Hitec".

The purpose of the tertiary salt system is to provide a tritium trap to prevent diffusion of tritium from the primary circuit into the prime mover system via the circulating salt systems. The oxygen in Hitec combines with the tritium to form tritiated water which can be recovered from the system. In the MSRE, the production of tritium was greatly reduced by using $Li^7$ rather than $Li^6$ in the primary salt mix. Even so, the MSRD designers saw fit to incorporate this tertiary system. It has the significant advantage that the nitrate-nitrite eutectic has a low melting point of $288^0$ F, thus reducing the probability of salt freezing in the prime mover heat exchangers which, by definition, represent the coolest part of the chain of salt systems.

Hitec could not be used as the secondary system salt because in the event of a primary heat exchanger leak, it would react with the fuel salt to precipitate thorium and uranium oxides.

The tertiary system salt remains nonradioactive, so the steam plant can be sited outside the containment and accessible for maintenance.

## MSDR PRIMARY PUMPS

Three primary fuel salt pumps are sited immediately over the reactor vessel. Previous experience showed that sump type centrifugal pumps are suitable. Each pump has a capacity of 8,100 gpm and develops a head of 150 ft.

Each primary pump is submerged in a tank with an excess gas volume to allow for primary salt expansion. Salt must be kept free of oxygen, so helium is used as the cover gas over primary and secondary fuel salt pumps; also being supplied as a purge around pump seals.

## REMOVAL OF XENON AND NOBLE GASES

In the MSRE, the primary pump bowl is the surge space for the circulating loop. Helium at 5 psig blankets the salt and 40% of the pump discharge is sprayed into the gas space to allow $Xe^{135}$ and krypton to escape from the salt. A flow of helium carries the xenon and krypton away to charcoal beds. Apart from some minor problems with frozen salt in the off gas line and small variations of gas entrained in the primary salt system, this simple method of removing xenon and krypton worked well.

However, for the larger MSRD plant, the system was developed and changed significantly. A bypass from each primary pump discharge, back to its inlet, is sized to take about 10% of the flow. In this bypass line there is a bubble generator to admit helium gas to the salt and a gas stripper to remove it. The gas stripper removes about two volumes of salt with each volume of gas removed. This mixed stream, containing helium, xenon and krypton, is led to the drain tank, which thus becomes the gas separating volume. In the drain tank two jet pumps return salt to the primary system, while half of the gas is used in the bubble generators and the remainder goes to carbon beds. After a 90 day dwell period, this gas is returned to the helium blanket system.

## THE DRAIN TANK

The drain tank is far from being the simple component it might seem from over-simplified descriptions of liquid salt reactors. In the MSRD design it has 3 key functions, namely:

a. To receive the full charge of fuel salt should heat removal by the power generating systems fail. A single 6 inch diameter pipe, fitted with a freeze valve, connects the bottom of the reactor vessel to the drain tank. Decay heat from the fuel salt must then be removed
b. To receive the flow of primary salt and highly radioactive entrained gas from the 3 centrifugal gas strippers in the primary pump by-passes. The drain tank allows the liquid salt and the gas to separate. During normal operation this function requires removal of 8MW of heat from the drain tank.
c. To receive liquid salt spills resulting from the unlikely event of leakage from any part of the plant. Catch pans in each cell drain to the drain tank through a rupture disc.

The drain tank comprises a tank in a tank. The outer tank is 24 ft tall and 12.5 ft diameter. Heat removal from the inner tank is effected by vertical thimbles suspended from the lid through which NaK flows by natural circulation to tubes in a water tank heat sink.

Gas from the drain tank passes through a particle trap to remove solid fission product daughters of the stripped xenon and other gaseous fission products. The gas stream then divides:

a. 50% returns to the primary pump by-pass bubble generators, after a 6.5 hour transit time which reduces the radioactivity somewhat.
b. 50% passes to one of two charcoal absorber beds where each charge must be kept for 90 days before it can be used to supplement the helium in the off gas system.

Separated liquid fuel salt from the centrifugal gas strippers is returned to the fuel system by two jet pumps suspended from the centre of the drain tank lid, one of which extends to the bottom of the drain tank.

## FLUSH TANK & FUEL TANK

Two further tanks are provided for use during commissioning and operating the plant. The Flush Tank contains carrier salt, without fissile fuel, that is used to flush primary and secondary systems before salt containing fissile fuel salt is added to the primary circuit. The fuel tank is provided to hold salt containing fissile fuel salt which is added to the primary system during the approach to initial criticality and which is added during operation, as required to maintain the reactivity of the primary system as the initial charge is depleted.

## REACTOR CONTROL

The main control is by adjusting the concentration of the fissile component of the fuel salt. For auxiliary control, the reactor is provided with 6 cruciform control rods comprising boron carbide clad in Hastelloy N. Full insertion of one control rod shuts down the reactor.

Such a reactor has a marked negative temperature coefficient of reactivity with a quick response. As power demand increases, reducing the mean temperature of fuel salt in the reactor vessel, the salt density increases and fuel is made up from the primary fuel pump tanks. The higher fuel density results in increased rate of fission. With a power reduction, fuel salt temperature and density increase and the rate of fission falls.

## STEAM PLANT

The steam plant has conventional, high, intermediate and low pressure steam turbines on a common shaft generating 350 MW(e) from steam at 2400 psi and 900$^{o}$F. The overall efficiency was estimated to be 36.6%

## SUMMARY

The MSRD design represents a conservative design building on the success of the MSRE and incorporating solutions to deal with the 100-fold power increase and the incorporation of a steam generating plant.

## ILLUSTRATIONS

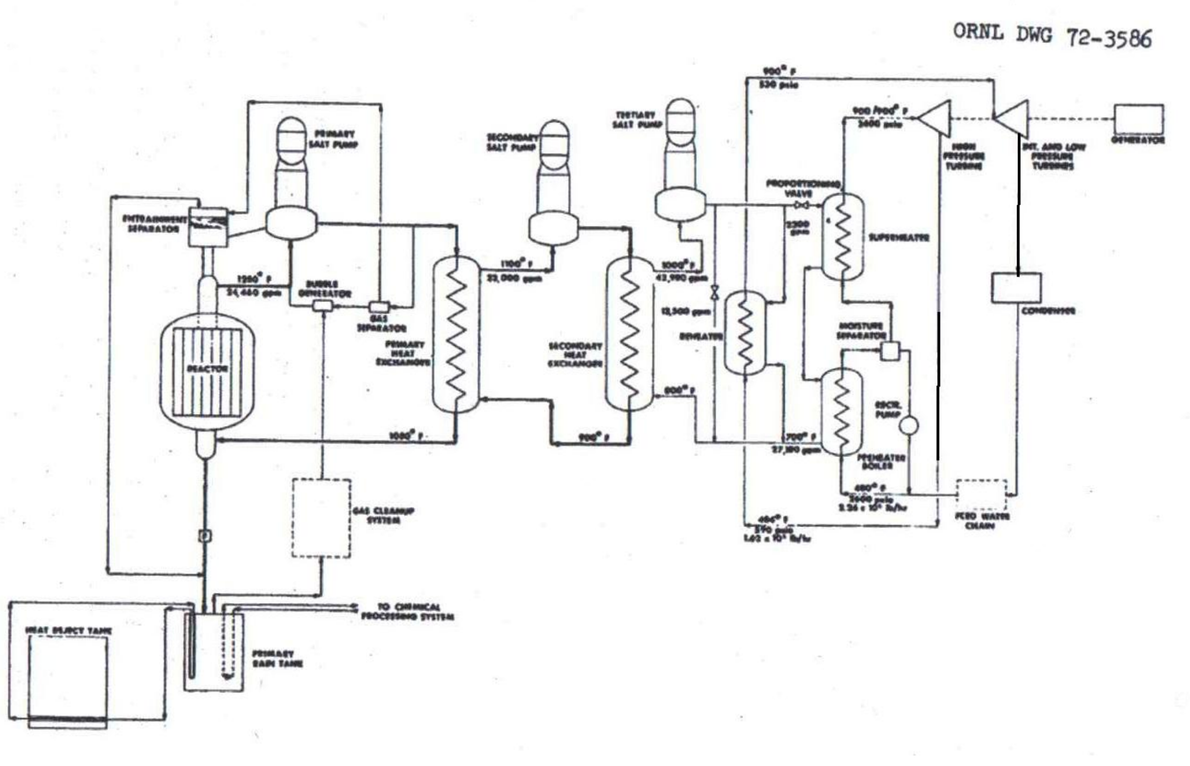


Figure 1: Simplified Flowsheet for 350 MWe Molten Salt Demonstration Reactor [2]

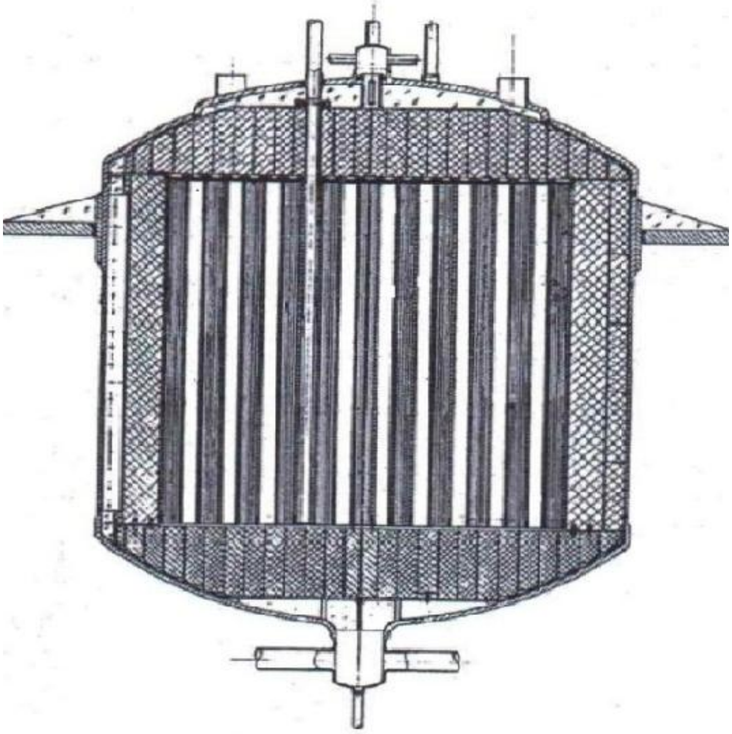

Figure 2: Reactor Vessel Elevation

**REFERENCES**

1 E S Bettis, L G Alexander, H L Watts “Design Studies of a Molten Salt Reactor Demonstration Plant” ORNL-TM-3832. Oak Ridge National Laboratory. 1972

2 ORNL Drawing, ORNL DWG 72-3586

# APPENDIX 3

# Thorium

Thorium is a natural radioactive chemical element with the symbol Th and atomic number 90. It was discovered in 1828 and named after Thor, the Norse god of thunder. In nature, thorium is found solely as thorium-232 ($Th^{232}$). It decays by emitting an alpha particle and has a half-life of about 14.05 billion years. It is estimated to be about three times more abundant than uranium in the Earth's crust.

Pure thorium is a silvery-white metal which is air-stable and retains its lustre for several months. Thorium has one of the largest liquid temperature ranges of any element, with 2,946 °C between the melting point and boiling point.

**The Thorium Cycle**

Thorium is not a fissile element, however it is fertile because $U^{233}$, which is fissile, can be obtained from thorium through neutron capture. When a nucleus of $Th^{232}$ captures a neutron it becomes $Th^{233}$. This then decays by beta emission to proactinum-233 ($Pa^{233}$) with a half-life of 22 minutes. In turn the $Pa^{233}$ decays by further beta emission to $U^{233}$ with a half-life of 27 days:

$$n^1 + {}_{90}Th^{232} \rightarrow {}_{90}Th^{233}$$

$${}_{90}Th^{233} \rightarrow {}_{91}Pa^{233} + {}_{-1}\beta$$

$${}_{91}Pa^{233} \rightarrow {}_{92}U^{233} + {}_{-1}\beta$$

Hence thorium is well suited to be the blanket in a breeder reactor in which the seed is a fissile fuel such as $U^{235}$, $Pu^{239}$ or $U^{233}$ itself. This was the application being developed by Weinburg in the ORNL, using molten salt reactors. Designs were prepared of two types of breeder reactor, namely 2 fluid and single fluid.

# APPENDIX 4

# Tritium Removal

*"At the time the MSR program was cancelled, tritium control was considered the largest remaining engineering development challenge …"*

In a molten salt reactor, tritium is formed when lithium is irradiated with neutrons, according to the reaction $^{6}Li + n \rightarrow {}^{4}He + {}^{3}T$. The production of tritium is greatly reduced if the lithium salts use lithium-7, purified to remove lithium-6, but lithium-6 is also a fission product.

Hence in a molten salt reactor means to remove tritium must be provided because at high temperatures the radioactive tritium, which is chemically like hydrogen, penetrates metals quite readily and, in a steam power plant application, unless captured in some way would appear in the steam generators and reach the atmosphere.

Following the successful operation of the MSRE, which had no arrangements for tritium removal, after considerable development it was found that a mixture of sodium fluoride and sodium fluoroborate would capture the tritium and this salt was proposed as the intermediate (secondary) salt in the early design of a molten salt breeder reactor (MSBR). Writing in retrospect in 1985, H G MacPherson, deputy director of ORNL until 1970, believed this had solved the tritium problem [1]. However, corrosion studies [2] [3] indicated that impurities in the salt, including water (such as would be produced from tritium capture) increased the rates of corrosion of Hastelloy N alloys. Whether for this or other reasons, the search continued for means of capturing tritium and attention turned to the nitrate - nitrite heat transfer salt, $KN0_3$-$NaN0_2$-NaNO, (known commercially as "Hitec"). A 1971 report [4] investigating the cost of using Hitec in a tertiary salt system stated:

> *"Studies are being made at ORNL of several different methods of tritium control; of these, the introduction of a third salt-circulating system to chemically trap the tritium between the secondary salt and the steam system is the only one well within present technology and, on the basis of present knowledge, offers assured confinement of the tritium."*

The problem with Hitec, however, is that at temperatures above 850°F (454°C) the salt, when used in a closed system, undergoes a slow thermal breakdown of the nitrite to nitrate, alkali metal oxide, and nitrogen: $5NaNO_2 \longrightarrow 3NaNO_2 + Na_2O + N_2$. Limiting the heat transfer from the salt to the generating plant to this temperature largely negates one of the significant advantages of molten salt reactors, namely the efficiency benefit of high temperature operation. Nevertheless, this was the solution adopted by the MSDR with its steam generating plant.

In 2006, Charles Forsberg of ORNL presented a paper [5] in which he explored how technological advances that had occurred since the cessation of development might solve or alleviate the outstanding problems of molten salt reactors. He stated:

> *"At the time the MSR program was cancelled, tritium control was considered the largest remaining engineering development challenge, because any tritium that entered the steam cycle resulted in tritiated water. Isotopically separating tritiated water from non-tritiated water in the steam cycle is difficult and expensive. …*
>
> *"… The MSR program partly developed the use of a fluoroborate coolant salts in the secondary heat transfer system to trap the tritium. While technically workable, such systems are potentially complex and expensive for high levels of tritium trapping."*

He pointed out that Helium-cooled high-temperature reactors produce tritium from nuclear reactions with $He^3$ and from leaking fuel; consequently, these reactors are equipped with systems to remove the tritium from the helium. Hence he went on to say:

"… *Adoption of a Brayton cycle provides an alternative tritium trapping option where the tritium is removed from the helium in the Brayton power cycle. This is potentially a high performance low-cost option based on demonstrated inexpensive methods to remove tritium gas or tritiated water from helium. ….*"

The warship molten salt power plant described in this paper, does in fact propose the use of Brayton Cycle CCGT generators, partly because they are suited to a high temperature heat source and also because they do not give rise to an infrared signature.

If, therefore, the helium working fluid of the CCGTs can be relied upon to trap tritium and a tritium removal system can be provided on board, it is possible to discard the tertiary Hitec system and provide the CCGTs with a higher temperature secondary system heat source, thereby achieving the efficiency benefits of the molten salt reactor / CCGT combination.

## REFERENCES


1 H G MacPherson "The Molten Salt Reactor Adventure". NUCLEAR SCIENCE & ENGINEERING: **90**, 374-380 (1985)

2 J. W. Koger and A. P. Litman "Compatibility Of Fused Sodium Fluoroborates And $BF_3$ Gas With Hastelloy N Alloys" . 0RNL-TM-2978. June 1970

3 A. N. Smith. "Experience With Sodium Fluoroborate Circulation In An MSRE-Scale Facility". 0RNL-TM-2344. September 1972.

4 Roy C. Robertson. "Estimated cost of adding a third salt-circulating system for controlling tritium migration in the l000-MW(e) MSBR. July 1971

5 Charles W. Forsberg, Oak Ridge National Laboratory. "Molten-Salt-Reactor Technology Gaps". Paper 6295. Proceedings of ICAPP '06. Reno, NV USA, June 4–8, 2006.

# APPENDIX 5
# Plant & Ship Design

Below follows a series of images which depict the notable features of the plant and ship design.

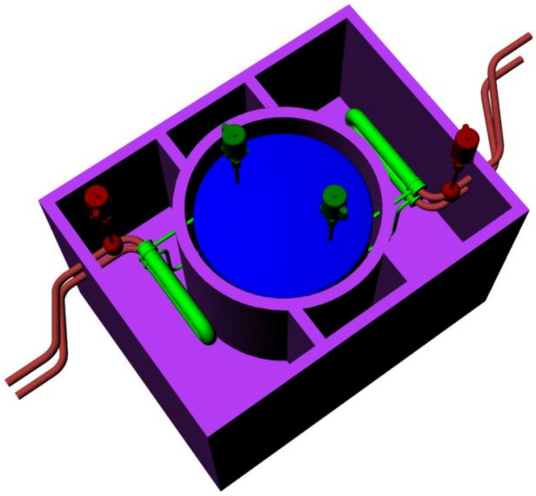

**Figure 2: Reactor Cell Layout**

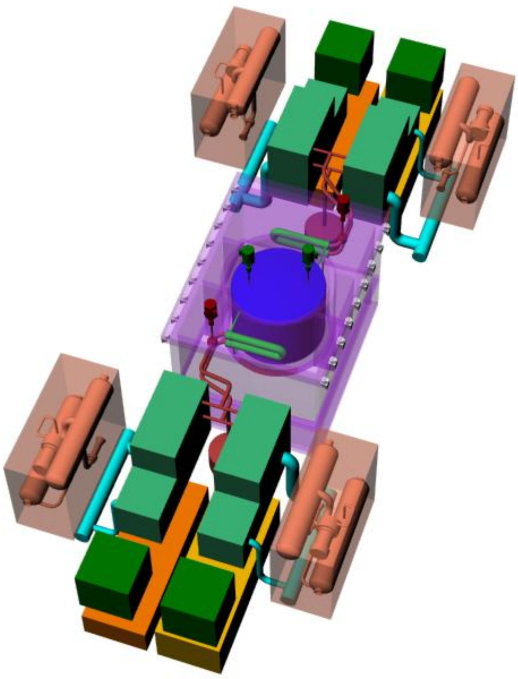

**Figure 6: Machinery & Reactor Arrangement**

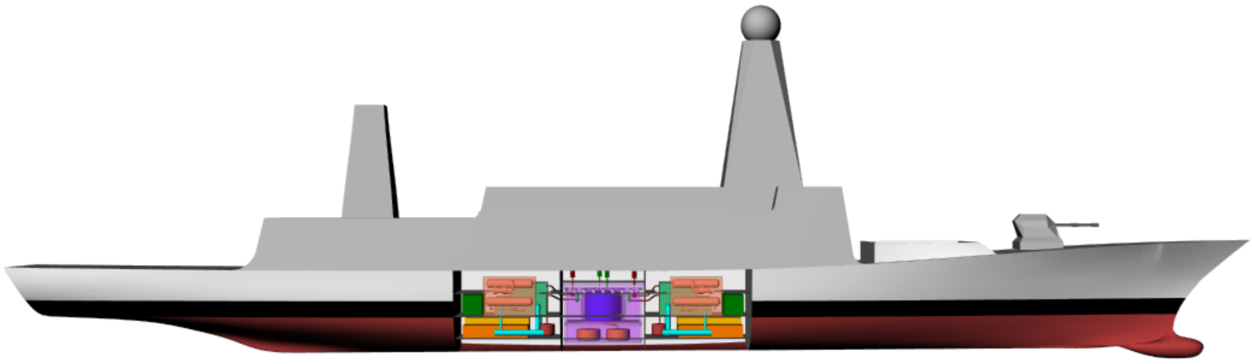

**Figure 9: Whole Ship View**

# APPENDIX 6

# Summary of thorium reactor characteristics

### INTRODUCTION

The thorium breeder reactor was seen as an alternative to the plutonium fast breeder reactor. However, the latter was preferred for political reasons and development of the thorium reactor ceased. Nevertheless, sufficient theoretical and practical work had been done to demonstrate the practicability and the advantages of the thorium reactor. In the context of a naval plant, which is a converter rather than a breeder, the thorium reactor has several advantages. Inevitably, there are some disadvantages. While some of the disadvantages are inherent, others could be alleviated by further work.

### ADVANTAGES

a. It operates at atmospheric pressure, thus avoiding accident sequences that with other types of reactor originate with low pressure.
b. It does not require an expensive, high quality pressure vessel.
c. It is inherently stable and load following, with a quick response.
d. Xenon135 and gaseous fission products are removed from the fuel continuously, yielding consistent reactivity and easing reactor control throughout power transients.
e. It has low pumping requirements.
f. It does not have nearly such demanding manufacturing requirements as are imposed by a solid core incorporating fissile fuel.
g. It operates at high temperature which, with appropriate generating plant, gives high thermal efficiency and high power to weight and size ratios.
h. There is a large (500 $^{0}$ C) margin between the operating temperature and the boiling point of the fuel salt, allowing time to react in the event of loss of decay heat cooling.
i. The fuel salt can be “spiked” to confer virtually insurmountable resistance to proliferation and its use in nuclear weapons.
j. There is an abundant world supply of thorium, which is used in the reactor in its natural state, requiring no separation or pre-use processing. The earth’s crust contains three times as much thorium as U238 and 400 times as much as U-235 .
k. After reprocessing, the wastes are predominantly short-lived fission products with relatively short half-lives. In a waste repository, safe radiation levels would be reached in 300 years, as opposed to the tens of thousands of years of actinides with much longer half-lives.

### DISADVANTAGES

a. Pipes and components comprising the salt systems must be maintained above the high melting temperature of the salt until emptied by draining to the drain tank. Isolating the ship’s structure and other compartments from the high temperature systems and compartments is arguably the main problem facing the ship designer.
b. Fuel salt drained from the reactor into the drain tank requires to be cooled to remove decay heat and the cooling system needs high reliability.
c. Due to fission product activity, systems and components containing fuel salt are highly radioactive and remote maintenance equipment is needed. This also applies to the off-gas and drain tank systems.
d. Bare graphite used in the core is susceptible to distortion and damage in a high neutron flux. Either the power density must be restricted in order to give the graphite longer life as in the case of the MSRD,

(see Appendix 1) or the plant configuration and the design of the containment must allow for periodic renewal of the graphite core throughout the life of the reactor.

e. Tritium removal was not demonstrated by ORNL and although it is technically feasible the warship application needs development.

f. The fuel in a molten salt reactor is dispersed and this complicates the shielding requirements which are notably different to those of a PWR. The requirement to also thermally insulate the hot compartments exacerbates the naval architect's problems.