Politeknik Dergisi
Cilt: 5 Sayı: 3 s. 227-231, 2002

Journal of Polytechnic
Vol: 5 No: 3 pp. 227-231, 2002

# NEUTRONIC INVESTIGATION OF THE ARIES-ST FUSION REACTOR BY USING MOLTEN SALT WITH $ThF_4$

Mustafa ÜBEYLİ[*], Taner ALTINOK[**]
[*]Gazi University, Technical Education Faculty, Department of Mechanical Education,
06500 Teknikokullar, ANKARA
[**]Kara Kuvvetleri Komutanlığı, ANKARA

## ABSTRACT

A hybrid modeling of ARIES-ST of the 1000 $MW_{el}$ power plant is investigated by using molten salt containing $ThF_4$. Calculations are done by using one-dimensional code of SCALE4.3. Tritium breeding ratio is greater than 1.05 so that tritium self-sufficiency is maintained for DT fusion driver. In this hybrid model, a significant amount of fissile fuel can be produced with a fissile fuel breeding ratio of $^{233}U$ = 0.123 per incident neutron at start-up conditions, which corresponds to 3742 kg $^{233}U$/year by a full fusion power of 2740 MW.

**Key words:** ARIES-ST fusion reactor, molten salt, fissile fuel breeding.

# $ThF_4$ İÇEREN ERİYİK TUZ KULLANILARAK ARIES-ST FÜZYON REAKTÖRÜNÜN NÖTRONİK OLARAK İNCELENMESİ

## ÖZET

$ThF_4$ içeren eriyik tuz kullanarak 1000 $MW_{el}$ gücündeki ARIES-ST reaktörünün hibrid modellenmesi incelenmiştir. Hesaplamalar tek boyutlu SCALE4.3 kodu kullanılarak yapılmıştır. Trityum üretim oranı 1.05'den büyüktür ve bundan dolayı DT füzyon kaynağı için trityumun kendi kendine yeterli olması sağlanmaktadır. Bu hibrid modelde, önemli miktarda fisil yakıt üretilebilecektir. Başlangıç şartlarındaki nötron başına fisil yakıt üretim oranı $^{233}U$ = 0.123'tür ve bu da 2740 MW'lık tam füzyon gücü altında 3742 kg $^{233}U$/yıl demektir.

**Anahtar kelimeler:** ARIES-ST füzyon reaktörü, eriyik tuz, fisil yakıt üretimi.

## 1. INTRODUCTION

ARIES-ST is a 1000 MW fusion power plant design based on the spherical tokamak concept to serve as a commercial power plant. The spherical tokamak concept has many attractive features, including high beta and power density, low magnetic field, high self-driven current fraction and a compact power core (1). There have been many studies to improve the performance of the ARIES-ST fusion reactor (1-5). However the penetration of a competitive pure fusion reactor into the energy market is not expected before the year ~ 2050.

The fusion-fission hybrid is a combination of the fusion and fission processes, having features, which are complementary. The idea is to surround the fusion plasma with a blanket made of the fertile materials ($U^{238}$ or $Th^{232}$) to convert them into fissile materials ($Pu^{239}$ or $U^{233}$) by transmutation through the capture of the high yield fusion neutrons. The fertile materials may also undergo a substantial amount of fission, especially, under the irradiation of the high energetic 14 MeV- (D,T) neutrons. Also, the bred fissile material can be partly burnt in the hybrid blanket "*in situ*" and partly it can be extracted as an additional fuel supply for the existing great number of light water reactors (LWRs). The multiplication of the total plant energy together with fissile fuel production can lead to a commercial hybrid plant driven by a non-commercial fusion reactor with a low or modest performance.

The hybrid mode of a different fusion reactor design concepts, the hybrid-PROMETHEUS reactor (6) and hybrid-ARIES-RS (7) have been presented recently. In this study, neutronic performance of hybrid modeling of the ARIES-ST design is investigated by using heavy molten salt containing $ThF_4$.

## 2. THE ARIES-ST HYBRID REACTOR CONCEPT

The engineering details of the ARIES-ST reactor are given in references (1-5). The ARIES-

ST consists of high temperature shield following first wall in the inner blanket where breeding blanket does not exist. The power core uses an advanced 'dual cooled' breeding blanket with flowing $Li_{17}Pb_{83}$ and He-cooled ferritic steel structures. Tritium breeding zone following immediately first wall is situated with a thickness of 100 cm in the outer blanket. He manifolds and low temperature shield follow thereafter.

In ARIES-ST design, ferritic steel(with 9% Cr, 2% W, 0.25% V, 0.07% Ta, 0.1% C ) is chosen as main structure material which has low activation property. Borated stainless steel is also used as structural material in the low temperature shield. After the decommissioning of the fusion power plant, these materials end up with class C nuclear waste according to the 10 CFR 61 regulations (8), suitable for shallow burial purposes. Table 1 gives the material compositions and densities of the ARIES-ST.

Table 1. Composition and dimension of materials in the zones of the blanket

| Zone | Material | Species | Atomic densities ($10^{24}cm^{-3}$) |
|---|---|---|---|
| High temperature shield | 80 % Ferritic steel 20% Helium | Cr | 5.980-3 |
| | | W | 1.329-3 |
| | | V | 1.661-4 |
| | | Ta | 4.651-5 |
| | | C | 6.645-6 |
| | | Fe | 5.886-2 |
| | | He | 2.167-4 |
| First wall (inner) | 70 % Helium 30 % Ferritic steel | Cr | 2.243-3 |
| | | W | 4.984-4 |
| | | V | 6.229-5 |
| | | Ta | 1.744-5 |
| | | C | 2.492-6 |
| | | Fe | 2.207-2 |
| | | He | 7.586-4 |
| First wall (outer) | 60 % Helium 40 % Ferritic steel | Cr | 2.990-3 |
| | | W | 6.645-4 |
| | | V | 8.306-5 |
| | | Ta | 2.326-5 |
| | | C | 3.322-6 |
| | | Fe | 2.943-2 |
| | | He | 6.502-4 |
| Helium manifold | 10 % Ferritic steel 90 % Helium | Cr | 7.475-4 |
| | | W | 1.661-4 |
| | | V | 2.076-5 |
| | | Ta | 5.814-6 |
| | | C | 8.306-7 |
| | | Fe | 7.357-3 |
| | | He | 9.753-4 |

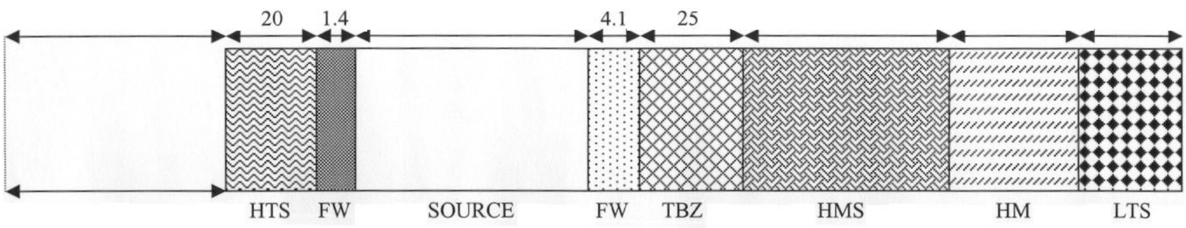


**Figure 1. One-dimensional hybrid modeling of the ARIES-ST reactor**
**(HTS = High Temperature Shield, FW=First wall, TBZ = Tritium Breeding Zone, HMS = Heavy Molten Salt, HM = Helium Manifolds, LTS= Low Temperature Shield, ).**
**Dimensions are given in cm, not in scale.**

The breeding zone has four quadratic channels each of which has a thickness of 25 cm. For simple change from the basic ARIES-ST design, the outer three breeding zone channels are replaced with molten salt containing flibe and $ThF_4$ to breed fissile fuel, without making any change elsewhere in the pure fusion ARIES-ST reactor. Figure 1 depicts the one-dimensional hybrid modeling of the ARIES-ST.

Using two different fluids in the quadratic channels may lead to the requirement of more complicated design of ARIES-ST. However, a hybrid reactor due to its fissile fuel breeding capability and more higher fusion energy multiplication is more advantageous than a fusion reactor with a low performance.

Table 1 continued

| | | | |
|---|---|---|---|
| Tritium breeding zone | 76 % $Li_{17}Pb_{83}$ (containing 60 % enriched $Li^6$) | $Li^6$ | 2.538-3 |
| | | $Li^7$ | 1.692-3 |
| | | Si | 4.336-3 |
| | 12 % SiC (75 % dense) | C | 4.337-3 |
| | | He | 6.502-5 |
| | 6 % Helium | Cr | 4.485-4 |
| | 6 % Ferritic steel | W | 9.967-5 |
| | | V | 1.246-5 |
| | | Ta | 3.488-6 |
| | | Fe | 4.414-3 |
| Heavy molten salt | 76 % molten salt | Th | 1.429-3 |
| | | $Li^6$ | 1.572-3 |
| | $(Li_2BeF_4)_{0.9} \cdot (ThF_4)_{0.1}$ | $Li^7$ | 1.943-2 |
| | | Be | 1.050-2 |
| | 12 % SiC (75 % dense) | F | 4.669-2 |
| | | Si | 4.336-3 |
| | 6 % Helium | C | 4.337-3 |
| | 6 % Ferritic steel | He | 6.502-5 |
| | | Cr | 4.485-4 |
| | | W | 9.967-5 |
| | | V | 1.246-5 |
| | | Ta | 3.488-6 |
| | | Fe | 4.414-3 |
| Low temperature shield | 60 % $H_2O$ | H | 4.015-2 |
| | 25 % Borated stainless steel | O | 2.008-2 |
| | | Mn | 4.397-4 |
| | 15 % Ferritic steel | Si | 2.199-4 |
| | | Cr | 5.298-3 |
| | | Ni | 2.198-3 |
| | | B | 2.198-4 |
| | | Fe | 2.577-2 |
| | | W | 2.492-4 |
| | | V | 3.115-5 |
| | | Ta | 8.721-6 |
| | | C | 1.246-6 |

## 3. NUMERICAL CALCULATIONS

Numerical calculations are performed with the help of the SCALE4.3 SYSTEM using the 238 groups library, derived from ENDF/B-V (9). The neutron transport calculations are performed by solving the Boltzmann transport equation with transport code XSDRNPM (10) in $S_8$-$P_3$ approximation by using Gaussian quadratures (11). The numerical output of XSDRNPM is processed with XSCALC (12) to determine the main reactor parameters.

*The resonance calculations in the fissionable fuel element cell are performed with*

- BONAMI (13) for unresolved resonances

and

- NITAWL-II (14) for resolved resonances.

CSAS control module (15) is used to produce the resonance self-shielded weighted cross-sections for XSDRNPM.

In order to have tritium self-sufficiency, tritium breeding ratio should be higher than 1.05. TBR (Tritium Breeding Ratio) value in the investigated blanket is 1.12 means that tritium self-sufficiency is maintained for DT fusion driver.

One-dimensional SCALE calculations give a fissile fuel breeding rate of $^{232}Th(n,\gamma)^{233}U = 0.123$ per incident fusion neutron at start-up conditions, which corresponds to 3742 kg $^{233}U$/year by a full fusion power of 2740 MW. However, previous studies on long-term plant operation of hybrid blankets point out that this high fissile fuel production rate would decrease rapidly, due to the burn-up of the new fissile fuel *in situ* (16-18). Therefore, the reduction in fissile fuel breeding can be more than by a factor of 2 after a plant operation of 1 year (19).

Neutron leakage out of the blanket is $1.02 \cdot 10^{-6}$ that is very low.

## 4. NEUTRON SPECTRUM

Figure 2 illustrates the neutron spectrum at selected locations in the outer blanket. Along with the neutron penetration in the blanket, a continuous degradation of the fusion neutron peak and a spectrum softening is observed. Especially, the sharpest softening of the neutron spectrum occurs in the molten salt zone due to inelastic scattering between neutrons and thorium. The fast neutron fluxes decrease in the radial direction whereas the lower energy group fluxes increase. In other words, the neutron flux curves show a variation towards the outer boundary from the harder neutron spectrum shapes to the softer ones.

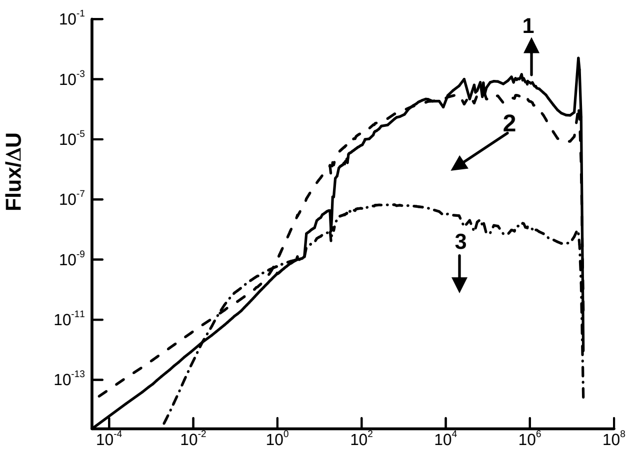


Figure2. Neutron spectrums at different locations of the outer blanket

1)First wall, 2) the outer end of the tritium breeding zone, 3) the outer end of the heavy molten salt zone

## 5. HEAT GENERATION

Total energy release in blanket can be calculated as:

Total energy release in blanket = $200 \cdot \sum_f + 4.784 \cdot T_6 - 2.467 \cdot T_7$ (1)

where, $\sum_f$ is total fission rate, $T_6$ is tritium produced by $^6Li(n,t)$ reaction and $T_7$ is tritium produced by $^7Li(n,n't)$ reaction. As it can be seen from equation(1), heat generation mainly depends on fission ratio and tritium production reactions of Li isotopes.

The energy multiplication factor, M, is defined as the ratio of the total energy release in the blanket to the incident fusion neutron energy. The M values are 1.41 and 1.39 for the original ARIES-ST reactor and hybrid ARIES-ST reactor, respectively. Therefore the replacement of heavy molten salt containing $ThF_4$ does not increase the M value of the reactor because of its very low fission ratio.

Figure 3 shows the heat generation profile of the external driven blanket. Heat generation decreases exponentially from inner beginning of the tritium breeding zone to outer end of the molten salt region. Because in the molten salt region, $\sum_f$ ($6.57 \cdot 10^{-4}$) is very small. And also lithium density in this region is much lower than in the tritium breeding zone. There is a slight increase in heating at the outer end of the tritium breeding zone due to the fact that increasing low energy neutron fluxes at the outer end of this zone lead to increase the $^6Li(n,t)$ reaction having high cross sections with low energetic neutrons

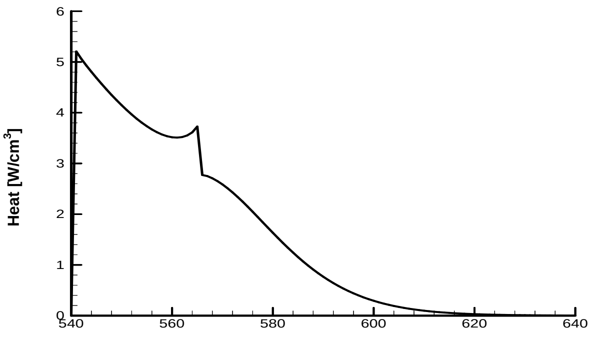


Figure 3. Heat generation profile for neutron load of 4.2 MW/m$^2$ at outer first wall

## 6. CONCLUSIONS AND RECOMMENDATIONS

The main conclusions can be outlined as follows:

- The use of molten salt containing $ThF_4$ leads to a significant amount of fissile fuel production in ARIES-ST reactor.
- Tritium production is self-sufficient for the DT driver.
- Heat production decreases towards to outer end of the molten salt zone. Due to very low fission ratio energy production is low and the replacement of heavy molten salt does not raise the energy multiplication of the reactor.

As a result, substantial amount of fissile fuel production is possible by using heavy molten salt in the ARIES-ST fusion reactor with maintaining sufficient TBR value for DT driver. For further studies, fissile fuel breeding by using heavy molten salt containing $UF_4$ in ARIES-ST fusion reactor can be considered. And also, radiation damage to structural materials, especially inner and outer first walls, can be investigated.

## REFERENCES

1. Mogahed, E.A., and the ARIES Team, Loss of Coolant Accident (LOCA) Analysis of the ARIES-ST Design, 13th Topical Meeting on the Technology of Fusion Energy, June 7-11, 1998, Nashville TN.

2. ARIES Team, Tillack, M.S., Wang, X.R., Pulsifer, J., Malang, S., Sze, D.K., ARIES-ST Breeding Blanket Design and Analysis, Fusion Engineering and Design, vol. 49-50, pp. 689-695, 2000.

3. Sze, D.K., et al., Blanket System Selection for the ARIES-ST, Fusion Engineering and Design, vol. 48, pp. 371-378, 2000.

4. El-Guebaly, L.A., and The ARIES Team, Three-Dimensional Neutronics Study for ARIES-St Power Plant, 13th Topical Meeting on the Technology of Fusion Energy, June 7-11, 1998, Nashville TN.

5. El-Guebaly, L.A., et al., Need for Inboard Shield to Protect the Center Post of ST Power Plants, 13th Topical Meeting on the Technology of Fusion Energy, June 7-11, 1998, Nashville TN.

6. Yapıcı, H., Übeyli, M., Yalçın, Ş, Neutronic Analysis of Prometheus Reactor Fuelled with Various Compounds of Thorium and Uranium, Annals of Nuclear Energy, (to be published).

7. Şahin, S., et al.,Neutronic Investigation of a Hybrid Version of the ARIES-RS fusion reactor, Annals of Nuclear Energy, (submitted).

8. Fetter, S., Cheng, E.T., Mann, F.M., (1988). Long-Term Radioactivity in Fusion Reactors, Fusion Engineering and Design, 6, 123.

9. Jordan W. C., S. M. Bowman, Scale Cross-Section Libraries, NUREG/CR-0200, Revision 5, 3, section M4, ORNL/NUREG/CSD-2/V3/R5, Oak Ridge National Laboratory, 1997.

10. Greene N. M., L. M. Petrie, XSDRNPM, A One-Dimensional Discrete-Ordinates Code For Transport Analysis, NUREG/CR-0200, Revision 5, 2, Section F3, ORNL/ NUREG/ CSD-2/V2/R5, Oak Ridge National Laboratory, 1997.

11. Şahin, S., Radiation Shielding Calculations For Fast Reactors (in Turkish), Gazi University, Publication # 169, Faculty of Science and Literature, Publication number 22, Ankara, Turkey, 1991.

12. Yapıcı, H., XSCALC for Interfacing Output of XSDRN to Calculate Integral Neutronic Data, Erciyes University, Kayseri, TURKEY, 2001.

13. Greene N. M., BONAMI, Resonance Self-Shielding By The Bondarenko Method, NUREG/CR-0200, Revision 5, 2, section F1, ORNL/NUREG/CSD-2/V2/R5, Oak Ridge National Laboratory, 1997.

14. Greene N. M., L. M. Petrie, R. M. Westfall, NITAWL-II, Scale System Module For Performing Resonance Shielding and Working Library Production, NUREG / CR-0200, Revision 5, 2, Section F2, ORNL/NUREG/ CSD-2/V2/R5,Oak Ridge National Laboratory, 1997.

15. Landers N. F., L. M. Petrie, CSAS, Control Module For Enhanced Criticality Safety Analysis Sequences, NUREG/ CR-0200, Revision 5, 1, Section C4, ORNL/ NUREG/ CSD-2/V1/R5, Oak Ridge National Laboratory, 1997.

16. Şahin, S., Baltacıoğlu, E., Yapıcı, H., Potential of a Catalyzed Fusion Driven Hybrid Reactor for the Regeneration of CANDU Spent Fuel, Fusion Technology, 20, 26, 1991.

17. Şahin, S., Yapıcı, H., Baltacıoğlu, E., Rejuvenation of LWR Spent Fuel in a Catalyzed Fusion Hybrid Blanket, Kerntechnik, 59, 270, 1994.

18. Şahin, S., Yapıcı, H., Rejuvenation of LWR Spent Fuel in Fusion Blankets, Annals of Nuclear Energy, 25, 1317, 1998.

19. Şahin, S., Yapıcı, H., Neutronic Analysis of a Thorium Fusion Breeder with Enhanced Protection Againts Nuclear Weapon Proliferation, Annals of Nuclear Energy, 26, 13, 1999.