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SUBJECT: Internally Cooled Molten-Salt Reactors

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Abstract

The initial and long-term nuclear characteristics of two internally cooled heterogeneous, graphite-moderated, two-region, molten-salt reactors have been studied. These reactors have doubling times of 22.5 years and 27.5 years. Methods of decreasing the doubling times by removing the Pa^{233} from the core and by increasing the specific power of the reactor are described.

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The initial and long-term nuclear characteristics of internally cooled heterogeneous, graphite-moderated, two-region, molten-salt reactors have been studied. The reactor considered in the calculations consists of a cylindrical core, 19.15 ft. in diameter and 19.15 ft. high, surrounded by a 2.5 ft. thick molten salt blanket contained in a 1.5 in. thick INOR-8 pressure shell. The core is penetrated by 0.975 in. cylindrical fuel-and-coolant passages arranged on a 1.475 in. triangular lattice parallel to the core axis. Each passage contains a coolant tube. The coolant tube has an inside diameter of 0.5 in. and a wall thickness of 0.1875 in. The fuel is contained in the 0.050 in. annulus that is formed by the moderator and the coolant tube.

The nuclear performance of the reactor described above was studied to determine the effects of changing the Th concentration in the fuel-coolant-blanket complex. Two cases were considered, 7 and 13 mole % Th. The compositions listed below were specified to be identical in the coolant, the blanket, and the fuel carrier. In each case sufficient $U^{235}F_4$ was added

	Case B1148 02142	Case B7149 01144
ThF ₄ (mole %)	7	13
LiF (mole %)	67.25	71
BeF ₂ (mole %)	25.75	16

to make the system critical. The ORACLE program Cornpone was used to calculate k and the Sorghum code was used to calculate the long-term performance of the reactor. The calculations were performed on a sphere having the same volume as the cylindrical core. Mean, "homogenized" densities (atoms/cm³) were used for each element. The fuel was processed approximately 90 times per year to remove fission products. An extract from the calculations is given in Tables 1 and 2 for the 7 and 13 mole % Th cases, respectively. The regeneration ratio, inventory and negative cumulative net burnup as functions of cumulative power are given in Figs. 1 and 2.

The 7 mole % Th²³² case gave a doubling time of approximately 27.5 years and the 13 mole % Th²³² case gave a doubling time of approximately 22.5 years.

These doubling times are lengthened by the build-up of Pa²³³ in the fuel. From the neutron balances, the Pa²³³ accounts for a loss of approximately 6.75% and 4.5% of the regeneration ratios of the 7 mole % and the 13 mole % cases, respectively. Removal of the Pa²³³ from the core would reduce these losses along with an accompanying reduction of the critical inventory. If the Pa²³³ were diluted to half of its concentration, the doubling times would be reduced to approximately 15 years from 27.5 years and to approximately 17 years from 22.5 years.

The Pa^{233} could be diluted by mixing the fuel carrier-salt with the blanket and the coolant fluids. This could be accomplished without an increase in the inventory by removing the UF_4 from the fuel-salt before the mixing and returning the UF_4 to the fuel. A flow diagram of a scheme to accomplish this is given in Fig. 3. As can be seen in Fig. 3, both the Pa^{233} and fission products are mixed with the blanket and coolant fluids. If the dilution of the fission products does not sufficiently reduce their concentration to an acceptable level, the CeF_3 removal of the rare earths can be introduced.

This processing scheme has several distinct advantages in addition to the removal of Pa^{233} from the core region. The most outstanding of these are:

1. No solids are required to be handled; e.g., solid UF_4
2. Rapid and accurate adjustment in the fuel concentration can be effected by controlling the flow of UF_6 to the reduction tank.
3. Leaks from the blanket to the fuel are automatically processed from the fuel.
4. Unforeseen leaks of blanket to fuel; e.g., such as, a broken fuel tube, can be tolerated by increasing the fuel processing rate.

To further reduce the doubling time of the internally cooled reactor, an arrangement of the coolant tube and fuel annulus is proposed that will allow the fuel annulus to be cooled on both sides and thereby more than double the heat removal capacity of a particular reactor. This arrangement consists essentially of two concentric tubes with fuel in the annulus and the coolant flowing through the inside tube and on the outside of the outside tube. An arrangement of this nature is shown in Fig. 4. The header for this arrangement is shown in section in Figs. 5 - 8. A reactor of any size can be fabricated from these unit cells. The tube headers can be made as large as the graphite fabrication will allow. If the single piece header is not large enough to fit the desired reactor size, several headers can be used and manifolded with graphite pipes.

TABLE I

Initial and Long-Term Nuclear Performance With Fuel Processing
of Two-Region, Graphite-Moderated, Unreflected, Blanketed,
Th²³²-Conversion Molten-Salt Reactors

Core Diameter: 21.92 ft.
Power: 1563 Mw (th)
Fuel Volume: 461 ft³
Plant Factor: 0.8
Th²³²-F₄ Concentration: 7 mole %

Duration of Operation, Years	0		1		5	
Cumulative Power Gen., Mwy	0		1250		6250	
	Inv., kg	Abs. Ratio	Inv., kg	Abs. Ratio	Inv., kg	Abs. Ratio*
Fissionable Isotopes						
U-233 (fuel)	500	1.0000	533.8	.9934	547.3	.9448
U-233 (blanket)	0		3.62		3.51	
U-235	0	0	3.91	.0066	35.53	.0552
Fertile Isotopes						
Th-232 (fuel)	25,550	1.0265	25,550	.9819	25,550	.9374
Th-232 (blanket)	131,700	.0852	131,700	.0824	131,700	.0796
U-234	0	0	65.2	.0193	246.4	.0678
Fuel Carrier						
Li-7+F-19		.0527		.0496		.0466
Li-7	7270		7270		7270	
Be-9	3660		3660		3660	
F-19	44,000		44,000		44,000	
Moderator						
C-12	249,200		249,200		249,200	
Be-9+C-12		.0811		.0752		.0694
Fission Products	0	0	4.778	.001	4.778	.0011
Parasitic Isotopes						
U-236, etc.	0	0	.190	.00002	10.67	.00132
Miscellaneous						
Pa-233 (fuel)	0	0	54.06	.0349	51.86	.0317
Pa-233 (blanket)	0	0	4.70		4.55	
Core Vessel + Leakage		.0193		.0187		.018
Blanket Carrier Salt	232,900				232,900	
Neutron Yield, η	2.2649		2.2632		2.2528	
Total Fuel Inv., kg	500		541.4		586.3	
Cumulative Net Burn-up, kg	0		30.5		-81.5	
Net Fuel Requirement, kg	500		571.9		504.8	
Regeneration Ratio	1.1117		1.0487		1.0530	

*Neutrons Absorbed per Neutron Absorbed in Fuel.

B1148
Case No. 02142

TABLE I (Continued)

Duration of Operation, Years	10		15		20	
Cumulative Power Gen., Mwy	12500		18750		25000	
	Inv., kg	Abs. Ratio	Inv., kg	Abs. Ratio	Inv., kg	Abs. Ratio*
Fissionable Isotopes						
U-233 (fuel)	555.2	.9120	561.6	.8971	567.3	.8901
U-233 (blanket)	3.43		3.4		3.37	
U-235	59.84	.088	72.1	.1029	78.58	.1099
Fertile Isotopes						
Th-232 (fuel)	25,550	.9096	25,550	.8946	25,550	.8852
Th-232 (blanket)	131,700	.0778	131,700	.0768	131,700	.0762
U-234	360.3	.0943	416.9	.106	446.7	.1117
Fuel Carrier						
Li-7+F-19		.0446		.0436		.043
Li-7	7270		7270		7270	
Be-9	3660		3660		3660	
F-19	44,000		44,000		44,000	
Moderator						
C-12	249,200		249,200		249,200	
Be-9+C-12		.0659		.0639		.0627
Fission Products	4.778	.001	4.778	.001	4.778	.001
Parasitic Isotopes						
U-236, etc.	41.57	.0050	82.25	.0099	127.5	.0152
Miscellaneous						
Pa-233 (fuel)	50.48	.0297	49.74	.0287	49.27	.0280
Pa-233 (blanket)	4.46		4.41		4.38	
Core Vessel + Leakage		.0176		.0174		.0173
Blanket Carrier Salt	232,900		232,900		232,900	
Neutron Yield, η	2.2456		2.2422		2.2404	
Total Fuel Inv., kg	618.5		637.1		649.3	
Cumulative Net Burn-up, kg	-222.6		-357.2		-482.4	
Net Fuel Requirement, kg	395.9		279.9		166.9	
Regeneration Ratio	1.0519		1.0488		1.0450	

*Neutrons Absorbed per Neutron Absorbed in Fuel

Case No. B1148
02142

TABLE II

Initial and Long-Term Nuclear Performance With Fuel Processing
of Two-Region Graphite-Moderated, Unreflected, Blanketed,
Th²³²-Conversion Molten-Salt Reactors

Core Diameter: 21.92 ft.
Power: 1563 Mw (th)
Fuel Volume: 461 ft³
Plant Factor: 0.8
Th²³²F₄ Concentration: 13 mole %

Duration of Operation, Years	0		1		5	
Cumulative Power Gen., Mwy	0		1250		6250	
	Inv., kg	Abs. Ratio	Inv., kg	Abs. Ratio	Inv., kg	Abs. Ratio*
Fissionable Isotopes						
U-233 (fuel)	982.6	1.000	1036.2	.9981	1069.7	.9748
U-233 (blanket)	0	0	3.22		3.16	
U-235	0	0	2.4	.0019	133.2	.0252
Fertile Isotopes						
Th-232 (fuel)	46,324	1.0888	46,324	1.0566	46,324	1.0249
Th-232 (blanket)	243,200	.0741	243,200	.0727	243,200	.0713
U-234	0	0	66.4	.0102	287.1	.0418
Fuel Carrier						
Li-7+F-19		.0322		.031		.0298
Li-7	7634		7634		7634	
Be-9	2212		2212		2212	
F-19	45,290		45,290		45,290	
Moderator						
C-12	249,200		249,200		249,200	
Be-9+C-12		.0377		.0355		.0333
Fission Products	0	0	4.778	.0007	4.778	.0006
Parasitic Isotopes						
U-236, etc.	0	0	.0585	.0000	4.647	.0005
Miscellaneous						
Pa-233 (fuel)	0	0	59.38	.0236	57.77	.0221
Pa-233 (blanket)	0	0	4.18		4.10	
Core Vessel + Leakage		.0161		.0158		.0154
Blanket Carrier Salt	225,000		225,000		225,000	
Neutron Yield, η	2.2489		2.2461		2.2398	
Total Fuel Inv., kg	982.6		1041.8		1106.1	
Cumulative Net Burn-up, kg	0		-0.3		-249.9	
Net Fuel Requirement, kg.	982.6		1041.5		856.2	
Regeneration Ratio	1.1629		1.1158		1.1159	

*Neutrons Absorbed per Neutron Absorbed in Fuel.

Case No. B7149
01144

TABLE II (Continued)

Duration of Operation, Years	10		15		20	
Cumulative Power Gen., Mwy	125,000		18,750		25,000	
	Inv., kg	Abs. Ratio	Inv., kg	Abs. Ratio	Inv., kg	Abs. Ratio*
Fissionable Isotopes						
U-233 (fuel)	1093.7	.9470	1112.0	.9281	1128.1	.9155
U-233 (blanket)	3.11		3.08		3.05	
U-235	74.2	.0530	105.0	.0719	127.4	.0845
Fertile Isotopes						
Th-232 (fuel)	46,324	.9967	46,324	.9772	46,324	.9629
Th-232 (blanket)	243,200	.0700	243,200	.0691	243,200	.0684
U-234	479.1	.0664	611.1	.0816	704.5	.0915
Fuel Carrier						
Li-7+Be-9		.0288		.0280		.0275
Li-7	7634		7634		7634	
Be-9	2212		2212		2212	
F-19	45,290		45,290		45,290	
Moderator						
C-12	249,200		249,200		249,200	
Be-9+C-12		.0315		.0302		.0293
Fission Products	4.778	.0006	4.778	.0006	4.778	.0006
Parasitic Isotopes						
U-236, etc.	23.1	.0023	52.7	.0053	89.91	.0089
Miscellaneous						
Pa-233 (fuel)	56.37	.0208	55.4	.0200	54.68	.0193
Pa-233 (blanket)	4.04		4.00		3.96	
Core Vessel + Leakage		.0152		.0150		.0148
Blanket Carrier Salt	225,000		225,000		225,000	
Neutron Yield, η	2.2322		2.2269		2.2233	
Total Fuel Inv., kg	1171.0		1220.1		1258.5	
Cumulative Net Burn-up, kg	-555.9		-850.7		-1134.4	
Net Fuel Requirement, kg.	615.1		369.4		124.1	
Regeneration Ratio	1.1122		1.1079		1.1035	

*Neutrons Absorbed per Neutron Absorbed in Fuel

Case No. B7149
01144

FIG-1 U-233 FUELED - Th^{232} CONVERTER 7 MOLE % Th^{232}

FIG. 2 U-233 FUELED - Th^{232} CONVERTER 13 MOLE % Th^{232} .

FIG. - 3
PROCESS SCHEME
FOR INTERNALLY
COOLED MOLTEN
SALT REACTORS

FIG. - 4
UNIT CELL -
INTERNALLY COOLED
MOLTEN SALT REACTOR

FIG.-5
SECTION A-A

FIG. - 6
SECTION B-B

FIG. - 7
SECTION C-C

FIG. - 8
SECTION D-D

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