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UNION CARBIDE NUCLEAR COMPANY

Division of Union Carbide Corporation

Post Office Box X

Oak Ridge, Tennessee

ORNL
CENTRAL FILES NUMBER

59-4-88

DATE: April 20, 1959

SUBJECT: Thermal Performance of UO_2 in Existing
and Planned Reactors

TO: Listed Distribution

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THERMAL PERFORMANCE OF UO_2 IN EXISTING AND PLANNED REACTORS

Reactor designers differ in their opinions as to the thermal performance that can be achieved with UO_2 pellets. This variance is undoubtedly due to uncertainties concerning the thermal conductivity of UO_2 and the amount of fission gases released from sintered UO_2 pellets.

In order to get a picture of how various reactor designers regard the capabilities of UO_2 fuel, the average specific power of reactors fueled with solid UO_2 pellets has been plotted against pellet diameter in Fig. 1. Shown also in Fig. 1 are curves of the equation

$$q = \frac{16 k \Delta T}{d^2}$$

for $\Delta T = 4000, 3000, 2000, 1000,$ and $500^\circ F$. The thermal conductivity, k , of UO_2 is taken as $1.0 \text{ Btu/hr}\cdot\text{ft}\cdot^\circ F$

It is evident that, of the reactors plotted in Fig. 1, the designers of NPD-2 are the most optimistic about the capabilities of UO_2 . The Westinghouse-designed reactors are the most conservative. In the case of FWR-1, even at the peak heat flux of 5 times the average, the ΔT is not much higher than the average ΔT for AECL's NPD-2 reactor. If the peak heat flux of NPD-2 were as much as double the average flux, the center temperature of the UO_2 pellets would be above the melting point.

Each reactor having cored UO_2 pellets has its own set of curves for specific power vs diameter. The curves follow the relation,

$$q = \frac{2 k \Delta T}{\left[\frac{r_2^2 - r_1^2}{2} - r_1^2 \ln \frac{r_2}{r_1} \right]}$$

In Fig. 2, curves are plotted for four cored-pellet reactors. Of these, the UO_2 is being worked the hardest in the Carolinas-Virginia pressurized-water reactor. The points for the rest of the reactors, including GCR-2, indicate a moderate conservatism on the part of the designers.

Figure 1
Pellet Diameter vs.
Average Specific Power
for Reactors Fuelled
with Solid VO_2 Pellets

Reactor Average Specific Power, Mw heat/Tonne U

Pellet Outside Diameter, In.

Figure 2

Pellet Diameter Vs. Average Specific Power for Reactors Fuelled with Cored UO_2 Pellets

The thermal conductivity of UO_2 , of course, affects the positions of the curves of Figs. 1 and 2. A value for k of 0.85 (WAPD-200) instead of 1.0 would shift the curves of constant peak fuel temperature downward to specific power values 85% of those shown.

Additional information about the reactors included in Figs. 1 and 2 is given in Table 1. Of particular interest in this table are the calculated maximum UO_2 temperatures in the average fuel elements and in the hottest elements. UO_2 surface temperatures were selected, quite arbitrarily, on the following basis:

- (a). In the case of water-cooled reactors, the UO_2 surface temperature was taken as 100°F above the coolant outlet temperature of PWR-1.
- (b). For the two sodium-cooled reactors, a temperature 100°F above the coolant outlet of the Hallam reactor was chosen.
- (c). A temperature 300°F above the gas outlet temperature of GCR-2 was selected for gas-cooled reactors.

Errors in these choices for UO_2 surface temperature probably are minor compared to other errors in the calculations.

The ΔT 's in the UO_2 pellets were determined from the formulae given above. The ratios of maximum-to-average specific power have been published for PWR-1, Dresden, GCR-2, and PEGCPR. For the rest of the reactors, a ratio of 2 has been used.

REFERENCES

1. J. W. Ullmann, Reactor Fuel Data Sheets, ORNL CF-56-10-142 (October 1956).
2. Atomic Energy Facts, Atomic Energy Commission, U. S. Government Printing Office, September 1957.
3. A. W. Kramer, Boiling Water Reactors, Addison-Wesley, New York, 1958.
4. The ORNL Gas-Cooled Reactor, ORNL-2500 (Apr. 1, 1958).
5. Fuel Costs for the Helium-Cooled Prototype Enriched Gas-Cooled Power Reactor, ACF Job Report No. 3-210-8.0 (Oct. 21, 1958).

TABLE 1. THERMAL PERFORMANCE CHARACTERISTICS OF REACTORS FUELED WITH UO_2 PELLETS

Reactor	Type	Designer	References	Fuel Element Geometry	UO_2 Pellet Diameter, in.	Specific Power, Mw/Tonne U		Assumed UO_2 Surface Temp., °F	Calculated ΔT in UO_2 Pellet, °F		Calculated Maximum UO_2 Temperature, °F	
						Average	Maximum		Average	Maximum	Average Element	Hottest Element
FWR-1 (Blanket)	FWR	Westinghouse	1, 2	120-rod bundle	0.357	9.3	47.9 ^a	640	520	2660	1160	3300
Dresden	BWR	General Electric	1, 2, 3	36-rod bundle	0.5	11.9	34.8 ^a	640	1300	3800	1940	4440
Yankee - Core 1	FWR	Westinghouse	1, 2	68,70,80 or 81-rod bundle	0.29	18.3	36.6 ^b	640	670	1340	1310	1980
Yankee - Core 2	FWR	Westinghouse	1, 2	68,70,80 or 81-rod bundle	0.26	19.8	39.6 ^b	640	590	1180	1230	1820
Chugach	SDR	Nuclear Development Corporation	1, 2	7-rod cluster	0.73	4.6	9.2 ^b	1030	1070	2140	2100	3170
Consumers' (Hallam)	SGR	Atomics International	1, 2	43-rod cluster	0.439	10.5	21.0 ^b	1030	880	1760	1910	2790
Northern States ^c	BWR	Allis-Chalmers	1, 2	8-tube plate	0.335	24.2	48.4 ^b	640	1180	2360	1820	3000
N.S. Savannah	FWR	Babcock-Wilcox	1	41-rod bundle	0.422	10.6	21.2 ^b	640	800	1600	1440	2240
BR-3 (Belgium)	FWR	Westinghouse	1	110-rod bundle	0.30	21.5	43.0 ^b	640	840	1680	1480	2320
NPD-2 (Canada)	PDR	Atomic Energy of Canada, Ltd.	1	7-rod cluster	0.975	5.6	11.2 ^b	640	2320	4640	2960	5280
GCR-2	GCR	Oak Ridge National Lab.	1, 4	7-rod cluster	0.75x0.32	5	10 ^a	1300	620	1240	1920	2540
FEBCPR	GCR	KE-ACF	1, 5	7-rod cluster	0.71x0.32	7.5	12.6 ^a	1300	790	1340	2090	2640
Carolinas-Virginia	FWR	General Atomics	1	19-rod cluster	0.445x0.12	17.8	35.6 ^b	640	1130	2260	1770	2900
Florida West Coast-East Central	GCR D ₂ O Mod.	Nuclear Development Corporation	1	19-rod cluster	0.54x0.34	17.5	35.0 ^b	1300	680	1360	1980	2660

(a). Published ratio of maximum: average specific power.

(b). Assumed ratio of maximum: average specific power = 2.

(c). Calculations were made as though fuel elements were made of rods. Since elements are essentially finned rods, temperatures would be lower.