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MONTHLY TECHNICAL REPORT
OCTOBER, 1963

Reactor Physics
Research and Engineering
N-REACTOR DEPARTMENT

Paul F. Nichols
and
Reactor Physics Staff

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MONTHLY TECHNICAL REPORT
OCTOBER, 1963

REACTOR PHYSICS

I. RESEARCH AND DEVELOPMENT ACTIVITIES

A. Lattice Parameter and Spectral Index Experiments

Details of the designs of special fuel elements for the lattice parameter and spectral index experiments have been completed. Work orders for uranium machining and fuel element assembly have been written and the uranium machining is well underway. Progress of the various phases of this work is indicated in Table I.

Both of the special fuel element designs discussed last month have been used for different parts of the tests. The conventional technique of putting sector foils or pins in short sections of fuel is being used for the cold tests. These short sections will be assembled with spaces and canned in aluminum instead of special zirconium tubes. End-drilled production line fuel elements are being used for the hot tests. Uranium rods machined to fit the holes will carry pins of the radioactivants and will be inserted in the end-drilled fuel elements. This latter technique yields a finished fuel element having nearly the same integrity as obtained on the production line.

Integrity of these special fuel elements has been discussed with Chemistry and Metallurgy Subsection personnel. A letter (D. H. Curtiss to R. A. Bennett, October 10, 1963) authorizes and specifies acceptance tests. The aluminum clad elements will be autoclaved

TABLE I

SPECIAL FUEL ELEMENT FABRICATION PROGRESS
(as of October 15, 1963)

Item	Cost Estimate	Work Orders To Date	Percent Completion	Completion Date (Estimated)
COLD TEST	$8960-Total	$7500-Total	25%-Total	December 1, 1963
Design			90%	December 1, 1963
Fuel Machining	(4500)		50%	"
Canning	2000		25%(Design plus sizing)	"
Clad - outside vendor	1500	Not required - local supply		
Radio Activant Procurement	(HL)		20%	"
Graphite	(300)		0%	"
Nickel F.P. Catchers	(0)	500	50%	"
Water Pin Holders	660		0%	"
HOT TEST	$7840-Total	$5500-Total	10%-Total	December 1, 1963
Design			90%	December 1, 1963
Fuel Drilling, Fuel Machining	(5500)		0%	"
Canning	(1000)		10% (Design only)	
Radio Activant Procurement	(HL)		20%	"
Water Pin Holders	(1340)		0%	"
Miscellaneous	-	150		
TOTALS	$16,800	$13,650		

at standard IPD autoclave conditions. This will provide an oxide film capable of better resistance to the 20 C - 50 psi-pH-10 water planned for the cold tests. Helium leak tests will also be made where possible. The Zircaloy-2 clad elements will be re-autoclaved at standard NRD autoclave conditions except for those containing cadmium. A reduction of autoclave conditions to 550 F - 1500 psi for 72 hours has been authorized for elements containing cadmium. This will prevent melting or distortion of cadmium covers. The melting point of cadmium is ~610 F.

A summary of the irradiation specifications for these experiments has been submitted to the Supervisor of Process Physics in the Process Evaluation and Control Subsection for inclusion in the "N-Reactor Startup Physics Test Program - Test Procedures," (HW-77917). A document outlining the technical details of the individual experiments is in final rough draft form and is being reviewed prior to publication as document HW-79237.

Irradiation of the special fuel elements during the "hot tests" is presently scheduled in the middle of the temperature coefficient tests. Period measurements during the temperature coefficient experiments at varying temperatures prior to and after the irradiation will produce undesired radioactivity in all radioactivants installed in the special fuel elements. This problem has been investigated with respect to its effect on counting rates of the base activities of Lu^{177} ($\tau_{1/2}$ = 6.7 days), Au^{198} ($\tau_{1/2}$ = 2.7 days), and Eu^{152} ($\tau_{1/2}$ = 9.3 hours), which are to be used to obtain spectral

[redacted]

indices in the method not utilizing cadmium. In all activants this additional activity is of the order of 5% of the total activity. This amount is significant, and because it will be induced at temperatures different from those during the primary irradiation, it must be subtracted from the measured activity. Error analysis indicates that 1% uncertainties in counting rates induce 10% and 3% uncertainties in inferred values of Westcott's r values and neutron temperatures, T_n, respectively. The correction for the undesired activity is expected to yield about a 1% uncertainty in activity ratios. Statistical counting uncertainties in the activities are expected to be approximately 1% also, hence T_n and r are expected to have maximum errors of $\pm$ 15 C and $\pm$ 0.02 respectively, from the measurements which do not utilize cadmium. It is anticipated that the value of r alone will be obtained with better accuracy from the cadmium covered measurements.

One method of interpreting these experiments will involve normalization of counting rates of detectors irradiated in N-Reactor to counting rates of detectors irradiated in a thermal column in TTR or PCTR. For these ratios to be independent of the respective flux-time irradiation curves, the latter must be identical. Making the thermal column irradiations identical to the hot test irradiation will be difficult in view of all the period measurements. This problem becomes acute with respect to activity ratios of fission product samples, since decay rates depend upon exposure history.

The solution to this problem is under investigation. A special hot test for these experiments without multiple period measurements would of course circumvent these difficulties and may be advisable.

B. T^3 and Np^{237} Production Rates for a Highly Enriched Fuel Loading

The production rates of T^3 and Np^{237} for highly enriched fuel loading have been estimated for inclusion in HW-79306, "Production of Special Nuclear Materials in N-Reactor," D. L. Condotta and E. G. Pierick, October 21, 1963 (Secret). Also included in that document is a discussion of the physics problems involved and the research and development program in physics which would be required for such a loading.

The fuel element design considered consisted of an outer tube of Li-Al, a central driver tube of highly enriched uranium in zirconium, and an inner target rod of Li-Al alloy. The driver tube contained $\approx$30 wt% U of which $\approx$93% was U^{235}. The estimates of the Np^{237} production were made in hand calculations with the use of effective cross sections based on Westcott r and T_n values. The production rates of the Np^{237} for this fuel geometry were found to be heavily dependent upon the self-shielding effects of U^{236} for the case in which the fuel is recycled. These effects were estimated from self-shielding factors supplied by Dr. P. F. Gast.

C. Graphite Heat Generation

Graphite heat generation calculations were also made for the same target-driver-target geometry assumed in the studies of the production rates of T^3 and Np^{237} for a highly enriched loading. The specific results of the heat generation calculations are of some general interest and are discussed here.

The results show that the heat generation in the graphite would be about 1.51 times that which is expected to result from the Phase I tube-in tube fuel geometry. This amounts to 13.2 mev/fission as compared to 8.74 mev/fission[1] for the Phase I geometry. The calculations assumed that the neutron energy contribution from slowing down would be about the same as calculated for various other fuel geometries[2], i.e., ~4.2 mev/fission. The γ-ray energy contribution was 9.0 mev/fission which is about twice the γ-ray contribution for Phase I geometry. This can be broken down further into 3.78 mev/fission from prompt fission γ energy, 3.19 mev/fission from fission product γ energy, 1.69 mev/fission from U-Zr-2 fuel (n,γ) γ energy, and 0.34 mev/fission from Zr-2 cladding and process tube and graphite (n,γ) γ energy. As was expected from the above results, the γ-ray energy dissipated in the Zr-2 process tube was calculated to be ~2.02 mev/fission or about twice that of Phase I geometry.

The γ energy dissipation was calculated with the aid of the IBM 7090 code COLPRO which will yield a Monte Carlo calculation of the collision probability in all regions for a γ-ray of given energy originating in a given region. In order to simplify the calculation with a small sacrifice in accuracy, the lattice was assumed to

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(1) HW-76343, "NPR Heat Generation - Startup Fuel Geometry," R. H. Meichle, July 17, 1963 (Secret).

(2) HW-59075, "NPR Heat Generation During Operation," R. Nilson, February [illegible], 1959 (Secret).

[redacted]

consist of four homogenized regions whose diameters were 1.473 inches (inner target and its cladding, inner water annulus, and inner cladding of fuel tube), 1.772 inches (U-Zr-2 fuel only), 2.708 inches (outer cladding of fuel, central water annulus, outer target and its cladding and outer water annulus), and 3.658 (process tube only).

The calculations were carried through two γ-ray collisions which accounted for almost all of the γ-ray energy which would be given up in the graphite. Of the (n,γ) reactions which occur in the N lattice only the Zr-2, U^{235} and graphite (n,γ) reaction were considered. All γ-ray energy resulting from graphite (n,γ) reactions were assumed to be lost entirely in the graphite, and all γ-rays which escaped through the process tube into the graphite were assumed to lose the remainder of their energy in the graphite.

Prompt and fission product γ-ray energy spectra, total γ-ray attenuation cross sections and γ-ray energy absorption cross sections were obtained from ANL-5800[3]. U^{235}, Zr-2 and graphite (n,γ) γ-ray energy spectra and total γ-ray attenuation cross sections for Zr-2 were taken from the previously referenced HW-76343.

D. Technical Bases

The hazards associated with hot startups are being investigated at this time. The consequences of an incident during this period of operation would probably be more severe than for normal startup

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(3) ANL-5800, "Reactor Physics Constants," Argonne National Laboratory.

or equilibrium incidents because 1) of high inlet water temperature, 2) startup ramps of 1 mk/sec are still obtainable, and 3) the pressurizer has not restored full system pressure by the time rod withdrawal must be initiated.

E. Physics Study of Single Tube Fuel Element

A further examination of the data assembled for the single tube fuel element reveals that the selection to date of engineering constraints such as pressure and flow rates has not been fully appropriate. The principal fault with present data is the appearance of unreasonably large coolant flow values. Apparently the cause of this problem is the improper use of the relationship between ΔP (pressure drop across the active zone) and the total reactor coolant flow in determining appropriate input for the FLEX 2 code. A well-founded selection of ΔP (across the active zone), ΔT (temperature rise in the coolant across the active zone), and use of the proper options in FLEX 2 should solve the problem. Use of the code in this manner is underway. A delay of perhaps two weeks in the completion of the survey study is anticipated as a result of this difficulty.

F. Thirty-six Tube Special 1.25 w/o Fuel Test Loading

A fuel has been designed that will meet the requirements of the special loading providing that the self-supports are similar in hydraulic behavior to those in the base load. Upon confirmation from the Chemistry and Metallurgy Subsection a report will be issued or the fuel re-designed.

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G. Code Development FLEX 3

Further improvements and modifications in the FLEX code are in progress. The capacity to handle the higher Pu isotopes in fuel regions and further flexibility in the handling of non-fuel materials in fuel regions are being incorporated. The first trial run of FLEX 3 has uncovered many errors in both the mechanics of programming and in logic. It is anticipated that the debugging of the modified program will be rather slow and it may take several months to make the code operable.

II. REPORTS PUBLISHED

HW-79183, "Monthly Technical Report, September, 1963 - Reactor Physics," Paul F. Nichols and Reactor Physics Staff, October 1, 1963, (Secret).

HW-78968, "FLEX 2 - A Fuel Element Design-Analysis Code," R. J. Shields, September 18, 1963 (Unclassified).

HW-79028, "Estimated Losses in Pu Production Rates from Simultaneous Irradiation of Np^{237} Target Material," E. E. Mills, September 18, 1963, (Secret).

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