High Temperature Gas-Cooled Reactor Criticality Research Program

by
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June 1976

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Report sponsored by the Thorium Utilization Program Office,
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EXECUTIVE SUMMARY

The objective of this report is to define the nuclear criticality research program needed in support of the High Temperature Gas-Cooled Reactor (HTGR) fuel cycle outside reactors. To meet this objective, an experimental plan, and the supportive calculational efforts, are presented herein, which, over a seven-year period, will provide sufficient information to permit maximizing plant equipment and operations.

To minimize the restrictions placed on HTGR plant equipment and operations by criticality considerations, experimental criticality data are needed over a sufficient range of operating variables to permit either direct application to plant conditions or verification of calculational techniques used in establishing safe limits. Currently, some data on well-moderated ^{235}U and ^{233}U exist which are applicable to the thorium-uranium fuel mixtures expected to be encountered in nonreactor facilities handling HTGR fuels. However, experimental data are needed for undermoderated ^{235}U and ^{233}U systems. Data are needed also on ^{235}U and ^{233}U containing soluble neutron poisons and in the presence of fixed neutron poisons. Also, few, if any, useful experimental data exist for thorium-uranium mixtures. However, this lack of data on Th-U mixtures is not of immediate concern, since currently envisioned processes are such that criticality limits must be established on the basis of uranium alone. For future fuels and processes in which the nonfissile diluent thorium and the fissile uranium cannot be separated, there would be a large economic incentive for including the thorium in the criticality considerations.

The research program presented herein covers all aspects of the experiments needed to provide data in the above areas. This includes the form, procurement, and processing of the experimental fuel in addition to the funding levels currently estimated for accomplishing the various phases of the program.

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HIGH TEMPERATURE GAS-COOLED REACTOR
CRITICALITY RESEARCH PROGRAM

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INTRODUCTION

The objective of the High Temperature Gas-Cooled Reactor (HTGR) Fuel Recycle Development Program is to develop recycle technology such that commercial plants for reprocessing and refabrication of HTGR fuels can be built and operated economically.⁽¹⁾ However, it is also necessary that these plants be operated such that nuclear criticality accidents do not occur. To assure criticality safety in such operations, limits are required on the equipment design and on the amount of fuel that may be handled or processed at any one location at any one time. These limits or controls present the main restrictions to plant throughput and, thus, the overall plant economy. The criticality control limits may dictate operations which are far from optimum, technically and economically, and the greater the uncertainty in a knowledge of criticality for a given operation, the more severe the restrictions become.

Flow diagrams for the HTGR Fuel Fabrication Pilot Plant (ORNL) and the HTGR Fuel Reprocessing Demonstration Plant (Idaho Falls) are shown in Figures 1 and 2, respectively. The complexity of criticality prevention in these operations becomes apparent when considering the fuel forms encountered. These forms and the anticipated conditions to be encountered are presented in the Appendix.

To minimize the restrictions placed on HTGR plant equipment and operations by criticality considerations, experimental criticality data are needed over a sufficient range of operating variables to permit either direct application to plant conditions or verification of calculational techniques used in establishing safe plant limits. The criticality data currently available on thorium-uranium fuel mixtures are listed in Table I, as well as in the Appendix. Of this currently available data, essentially only the well-moderated data for ^{235}U and ^{233}U are applicable to the thorium-uranium fuel mixtures expected to be encountered in nonreactor facilities handling HTGR fuels. Experimental data are needed for undermoderated ^{235}U and ^{233}U systems. Data are needed also on ^{235}U and ^{233}U containing soluble neutron poisons and in the presence of fixed neutron poisons. Also, few, if any, useful experimental data exist for thorium-uranium mixtures. However, this lack of data

HTGR FUEL FABRICATION PILOT PLANT-ORNL

(BASIC FLOW DIAGRAM)

FIGURE 1

The flowchart illustrates the U.S. DOE Fuel Cycle Strategy, detailing the process from initial fuel element preparation to final waste management and recycling.

Initial Processing: 14" X 31" Hex Elements (12/day) are processed by 3 crushers in series (3/16" size) and weighed in a hopper. The material is then transferred via air to a collection transfer hopper, then to a primary burner feed hopper.

Primary Burner and Air Classifier: The primary burner removes the outer pyrophoric C coating, producing fines and off-gas. The remaining ash is sent to an air classifier (WT. & Cross Section). The classifier separates the material into three streams: 233U Recycle Particles, Fines Hopper, and Fissile/Fertile Particles (only FSVR sic coated).

Secondary Burner and Batch Dissolver: The Fines Hopper and Fissile/Fertile Particles streams are sent to a secondary burner (900°C) which removes remaining C. The output is sent to a batch dissolver (120°C NP) where U₃O₈ + ThO₂ ash from the O₂ burner is added. The resulting slurry is centrifuged, separating solid disposal (SiC etc.) from a Th, U, FP stream.

Solvent Extraction and Recovery: The Th, U, FP stream is sent to an account-ability collection tank, then to a collection feed tank. It then undergoes steam strip to remove XS acid, followed by solution adjustment and solvent extraction. The resulting aqueous U, Th, and FP stream is sent to the 1st cycle of solvent extraction.

Recovery and Waste Management: The aqueous U, Th, and FP stream is processed through a series of scrubbers and strippers. The first stage recovers organic U and Th, while the second stage recovers organic U and Th. The remaining aqueous U + Th is sent to a calciner. The final waste stream (FP disposal) is sent to a waste storage tank.

Diagram Details: The diagram includes a detailed view of a fuel element cross-section, showing a central SiC core surrounded by a low density pyrolytic C layer and an outer high density pyrolytic C layer.

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TABLE 1 - SUMMARY OF AVAILABLE CRITICALITY DATA APPLICABLE TO THE HTGR FUEL CYCLE

Type of System	Type of Data	Source of Data
^{233}U - ^{232}Th -C	k_{∞} measurement on oxide rods in graphite at ~ 50 Th/U and 13756 C/U.	BNW in HTLTR
	k_{∞} measurement on oxide rods in graphite at ~ 11 Th/U and 2750 C/U.	BNW in HTLTR
^{235}U - ^{232}Th -C	k_{∞} measurement on oxide rods in graphite at ~ 30 Th/U and 6000 C/U.	BNW in HTLTR
^{239}Pu - ^{232}Th -C	k_{∞} measurement on oxide rods in graphite at ~ 30 Th/Pu and 7916 C/Pu.	BNW in HTLTR
^{233}U	Measured critical sizes in spherical and cylindrical geometries for homogeneous aqueous UN solutions at concentrations ranging from 13 g U/g to 600 g U/g.	ORNL
	Measured reflected and bare arrays of UN solution at 333 g U/g and 204 g U/g.	ORNL
	Measured critical size for ^{233}U metal sphere.	LASL
	25, 30 and 42 cm bare and reflected cylinders at 27 g/g to 200 g/g.	Saclay
^{235}U -C	Measured reflected and bare arrays of UN solution at 330 g U/g.	BNW
	Reactivity worth measurements on rods in graphite at 432 C/U to 5000 C/U.	GGA
^{233}U -C	Reactivity worth measurements on rods in graphite at 432 C/U to 5000 C/U	GGA
^{235}U	Measured critical size for ^{235}U metal sphere	LASL
^{235}U	Measured critical sizes in cylindrical geometry for homogeneous aqueous UO_2F_2 solutions at concentrations ranging from 78 g/g to 532 g/g	ORNL

on Th-U mixtures is not of immediate concern, since currently envisioned processes are such that criticality limits must be established on the basis of uranium alone. For future fuels and processes in which the nonfissile diluent thorium and the fissile uranium cannot be separated, there would be a large economic incentive for including the thorium in the criticality considerations.

The primary area in which criticality data are needed is that of under-moderated ^{235}U and ^{233}U . In the fuel fabrication plant, uranium loaded resins having H/U atomic ratios ranging from about 7 to 60 are possible between the loading of the weak acid resin, as feed stock to the carbonization furnaces, and the coater furnaces. Experimental criticality data exist for metal spheres of ^{235}U (2) and ^{233}U (3) but none are available for either of these nuclides in the presence of a small amount of moderation. Experimental data on low-moderated, about 20 H/U, uranium are also needed to support the design and operation of head-end equipment in the chemical reprocessing plant. The situation of most concern occurs when a small fraction of dissolver solution is slurried with fissile particles. Overly restrictive limits on the primary and secondary burners can also be crucial to the reprocessing plant's throughput capacity.

Since soluble neutron absorbers, such as cadmium, boron, and gadolinium, will almost certainly be required, as neutron poisons, to achieve an adequate fuel dissolution capacity in the reprocessing plant, criticality data are also needed for these low-moderated systems containing representative amounts of these neutron absorbing materials. Because of dilution during and following fuel dissolution, criticality data on these neutron poisons are also required for the more moderated systems up to H/U atomic ratios of about 200.

Plant throughput capacity and storage capacity can also be enhanced by the use of fixed neutron absorbers such as borated stainless steel raschig rings.

Data on these types of systems are lacking and needed over the H/U atomic ratio range of 50 to 300.

GENERAL DESCRIPTION OF CRITICALITY EXPERIMENTS AND DATA CORRELATIONS

Criticality measurements would be made on highly enriched homogeneous uranium systems to determine the effect of uranium concentration on the critical mass and volume. A concentration range from about 85 g U/liter (300 H/U) to about 1100 g U/liter (12 H/U) will be covered in the experiments to provide the needed data on undermoderated systems (~ 10 to 50 H/U) and the data needed for systems containing soluble and fixed neutron poisons. Critical masses and volumes will be determined over a concentration range from 85 g U/liter to 450 g U/liter in a cylindrical vessel filled with 4 wt% borated stainless steel raschig rings. Criticality data will be obtained over the full 85 to 1100 g U/liter concentration range with soluble boron, cadmium, and gadolinium compounds to measure the poisoning effects these nuclides have on the criticality of ^{235}U and ^{233}U . Measurement data will be obtained also on mixtures of these soluble poisons, since calculations have indicated that these nuclides are more effective neutron absorbers at different portions of the neutron spectrum.⁽⁴⁾

The experiments would be carried out in well-defined cylindrical geometry to permit a maximum amount of direct correlation to be made between the needed data, and to minimize the number of fuel composition adjustments required during the course of the experiments. As near as possible, identical sets of measurement data will be obtained for ^{235}U and ^{233}U .

EXPERIMENTAL FUEL

Several fuel forms were considered for use in the experiments. These were:

1. Uranyl fluoride solution
2. Uranyl nitrate solution
3. Uranium oxide-polystyrene compacts
4. Loaded resin microspheres
5. Coated microspheres in hydrogenous compacts

Fuel in the form of compacts was eliminated from consideration because of its high fabrication cost and the restrictive time interval imposed by radiation exposures associated with ^{233}U fuel. Also, wherever possible, fuel in solution form is preferred for the homogeneity not achievable in any other fuel form.

Loaded resin microspheres were eliminated from consideration as an experimental fuel because of the difficulty associated with particle flow, achievement of uniform bulk densities, wide H/X ranges required, and uniform addition of neutron absorbers.

Of the two solution forms, uranyl nitrate was considered to be the best choice. Uranyl nitrate hexahydrate, UNH, has an H/U atomic ratio of 12 and is in the liquid state at 60°C . Saturated aqueous uranyl fluoride at 60°C has an H/U atomic ratio of only 17, in addition to some viscosity problems and corrosion problems as saturation is approached. Also, UNH can be easily diluted, with relatively-easy-to handle nitric acid or with water, to achieve the H/U ratios required. Consequently, UNH was chosen as the fuel form to be used throughout the experiments.

Because of the difficulties currently associated with cross-country shipment of liquid radioactive material, Oak Ridge National Laboratory (ORNL) will provide the uranium fuel in the form of a uranium salt or as UO_2 enriched to at least 93 wt% in ^{235}U and ^{233}U . The fuel, after shipment to the Battelle

Pacific Northwest Laboratory (BNPL), will be dissolved and reconcentrated to dry UNH by BNPL to achieve a 12 H/U atomic ratio. The UNH will be received as a solid at the Critical Mass Laboratory (CML), reliquified, by heating to approximately 60°C, and transferred to geometrically safe storage tanks maintained at 60°C to 65°C. H/U variations will be obtained by dilution with nitric acid and water.

The anticipated cost and timing associated with fuel preparation and handling is itemized in Table II.

CRITICALITY RESEARCH PROGRAM

The research program is directed towards providing the reliable technological basis needed for assuring safe and economical criticality control throughout the HTGR fuel cycle, including the design and operation of equipment and facilities in fabrication, shipping, and processing operations. All the activities required to meet these objectives are covered in this program. Both experimental and analytical work are involved. Specifically, the program includes:

- 1) Identification and procurement of the data required for criticality safety in HTGR fuel cycle operations.
- 2) Data testing and analysis to provide validated analytical methods with ENDF/B cross sections (to guide selection and implementation of improved analytical methods).
- 3) Development of standards or guides on criticality prevention and control for fuel cycle operations covering limits for fuel fabrication and reprocessing.
- 4) Performance of survey calculations and development of general criticality data which serve as a basis for evaluating the criticality safety aspects of advanced processes in fuel cycle.

TABLE II

FUEL PREPARATION SCHEDULE AND CURRENT ESTIMATE OF COSTS INVOLVED
HTGR CRITICALITY EXPERIMENTS⁽¹⁾

	<u>FY-1977</u>	<u>FY-1978</u>	<u>FY-1979</u>	<u>FY-1980</u>	<u>FY-1981</u>	<u>FY-1982</u>	<u>FY-1983</u>
1. Preparation of 50 kg Natural UNH by BPNL	BPNL	---	---	---	---	---	---
2. Preparation of 50 kg 93 wt% ²³⁵ U Enriched UO ₂ by ORNL ⁽²⁾	---	\$50 k	---	\$60 k	---	---	---
3. Shipment of 50 kg 93 wt% ²³⁵ U Enriched UO ₂ From ORNL to BPNL	---	BPNL	---	BPNL	---	---	---
4. Dissolution of 50 kg 93 wt% ²³⁵ U Enriched UO ₂ and Concentration to UNH	---	BPNL	---	BPNL	---	---	---
5. Concentrate Aqueous ²³⁵ U(93)NH to UNH	---	---	---	BPNL	---	---	---
6. Removal of Cd from Aqueous ²³⁵ UNH Solution and Concentration to ²³⁵ UNH	---	---	---	---	BPNL	---	---
7. Concentrate Aqueous ²³⁵ UNH Solution Containing Gd	---	---	---	---	---	BPNL	---
8. Remove Gd and B from Aqueous ²³⁵ U(93)NH Solution and Return to ORNL	---	---	---	---	---	---	BPNL
9. Preparation of 50 kg 93 wt% ²³³ U Enriched UO ₂ by ORNL	---	---	\$100 k	\$100 k	---	---	---
10. Shipment of 50 kg 93 wt% ²³³ U Enriched UO ₂ from ORNL to BPNL	---	---	BPNL	BPNL	---	---	---
11. Dissolution of 50 kg 93 wt% ²³³ U Enriched UO ₂ and concentration to UNH	---	---	BPNL	BPNL	---	---	---
12. Purify and Concentrate Aqueous ²³³ U(93)NH Solution to UNH	---	---	---	BPNL	BPNL	---	---
13. Removal of Cd from Aqueous ²³³ U(93)NH Solution, Purify and Concentrate to ²³³ U(93)NH	---	---	---	---	BPNL	---	---
14. Removal of Gd from Aqueous ²³³ U(93)NH Solution, Purify, and Return to ORNL	---	---	---	---	---	BPNL	---

(1) Cost to be determined by Battelle Pacific Northwest Laboratory where indicated by BPNL and by Oak Ridge National Laboratory where indicated by ORNL.

(2) For planning purposes it is assumed the uranium will be shipped from ORNL as UO₂.

To accomplish these objectives in a timely and logistically feasible manner the various program activities have been categorized on a fiscal year basis beginning with fiscal year 1977. Each year's activities are based on an anticipated funding level, projected work load at the Critical Mass Laboratory, and the scheduling involved with fuel adjustment, purification, and storage. Some flexibility is possible in the time schedule; however, there is very little room for compression.

The anticipated funding level does not include cost associated with the preparation and handling of the experimental fuel. These costs are delineated in the preceding section on Experimental Fuel. A total projected funding schedule is broken down and shown in Table III.

FY-1977

ANTICIPATED FUNDING LEVEL: \$85 k

Pre-experiment planning and design will be continued, including procurement arrangements for the various fissile materials required to perform the experiments. Storage vessels and experimental equipment will be procured and installed. Natural uranium as UNH will be obtained for use in checking out the experimental facilities. A series of survey calculations will be completed in support of the planned experiments.

FY-1978

ANTICIPATED FUNDING LEVEL: \$190 k

Procurement of experimental criticality data will be initiated for developing the technology on nuclear criticality control in HTGR fuel cycle operations. In both fuel reprocessing and fuel fabrication plants, one of the most restrictive stages to plant throughput involves furnace type processing of normally undermoderated fuel. In fuel reprocessing plants this includes such areas as the ~ 24-inch ID cylindrical primary burners where the outer pyrophoric carbon is removed from crushed fuel assemblies. In the fuel fabrication plant the 9-inch ID cylindrical carbonization and coater furnaces normally process dry

fuel in the form of microspheres. Either of these type systems are extremely sensitive to external reflection and water addition; consequently, a thorough knowledge of their criticality parameters is particularly required to prevent their unduly restricting plant capacity.

A series of criticality experiments with uranyl nitrate solutions, having a ^{235}U enrichment of at least 93 wt% and H/U atomic ratios ranging from 12 to 300, will be initiated in bare and water-reflected cylinders. The measurements will be carried out in 12-inch ID and 24-inch ID cylinders. In the 50 to 300 H/U atomic ratio range (~ 450 g U/liter to ~ 85 g U/liter), criticality data will also be obtained on a 24-inch ID cylindrical vessel filled with 4 wt% borated steel raschig rings occupying about 20 vol% of the vessel. These latter measurements will provide the data needed for verifying calculations on ^{235}U solution storage vessels containing fixed neutron poisons.

FY-1979

ANTICIPATED FUNDING LEVEL: \$250 k

Criticality experiments with 93 wt% ^{235}U enriched (min.) uranyl nitrate solution will be completed. Data testing and analysis of the experiments to provide validated analytical methods with ENDF/B cross sections will be initiated and completed. The results of the experiments and data correlation will be reported on in the open literature. Development of standards or guides on criticality prevention and control for the HTGR fuel cycle will be initiated.

A series of criticality experiments with uranyl nitrate solutions, having a ^{233}U enrichment of at least 93 wt% and H/U atomic ratios ranging from 12 to 300, will be initiated in bare and reflected cylinders. The measurements will be carried out in 12-inch ID and 24-inch ID cylinders. In the 50 to 300 H/U atomic ratio range (~ 450 g U/liter to ~ 85 g U/liter), criticality data will also be obtained on a 24-in ID cylindrical vessel filled with 4 wt% borated steel raschig rings occupying about 20 vol% of the vessel.

FY-1980

ANTICIPATED FUNDING LEVEL: \$250 k

The criticality experiments with ^{233}U solutions will be completed. Data testing and analysis of the experimental data will be performed and reported on in the open literature.

Criticality experiments will be initiated and completed to measure the effect that cadmium as a soluble neutron absorber has on the criticality of ~ 93 wt% ^{235}U enriched uranyl nitrate. Measurements will be made on solutions having H/U atomic ratios ranging from 12 to 300 and containing up to 6 g Cd/liter.

FY-1981

ANTICIPATED FUNDING LEVEL: \$250 k

The experimental data obtained in previous years on ^{235}U and ^{233}U will be correlated and presented in the open literature. Development of standards or guides on criticality prevention and control for the HTGR fuel cycle will be continued.

Criticality experiments will be initiated and completed with 93 wt% ^{233}U enriched uranyl nitrate solution containing up to ~ 6 g Cd/liter, and having H/U atomic ratios ranging from 12 to 300.

Initiate and complete experiments with ~ 93 wt% ^{235}U enriched uranyl nitrate solutions containing up to ~ 3 g Gd/liter and having H/U atomic ratios ranging from 12 to 300. Based on the results of these measurements, experiments will be initiated with ^{233}U enriched uranyl nitrate solutions containing Gd.

FY-1982

ANTICIPATED FUNDING LEVEL: \$250 k

Complete experiments with ^{233}U enriched uranyl nitrate containing gadolinium and report the results of the measurements in the open literature.

Calculations and experimental data on plutonium systems have indicated that mixtures of gadolinium and boron are more effective soluble neutron

poisons over a wide range of fissile concentrations than is either one alone.⁽⁴⁾ To provide confirmatory data in this area for highly enriched ^{233}U and ^{235}U systems, criticality experiments will be performed with 93 wt% ^{235}U enriched uranyl nitrate solutions containing B-Gd mixtures and having H/U atomic ratios ranging from ~ 50 to 300.

The experimental data obtained on ^{233}U and ^{235}U enriched uranyl nitrate containing soluble neutron poisons will be correlated and published in the open literature. Standards or guides on the use of soluble poisons for criticality prevention in the HTGR fuel cycle will be developed. Recommendations on additional (if any) experiments needed for the HTGR fuel cycle will be made.

FY-1983

ANTICIPATED FUNDING LEVEL: \$100 k

Correlation of data will be completed and a report issued covering criticality prevention guidelines applicable to the HTGR fuel cycle. Safe limits, serving as the basis for standards on criticality control, will be deduced including those covering the applications and use of fixed and soluble poisons. The report will be based on the analysis of the critical experiments utilizing ENDF/B cross section data and modern codes; the "validated" computational methods will then be used to extend the application (and usefulness) of the report beyond the range covered in the original experiments.

BEYOND FY-1983

Experiments would be designed as necessary to handle the criticality data needs resulting from any special conditions that may be encountered in fuel cycle operations. Additional experiments would be performed, if needed, for formulating standards of excellence on criticality control. For example, the effects of Th dilution and the benefits derived therefrom may become of more interest. An experiment of interest involves determining the limiting critical

U concentration in unmoderated (dry) Th+U for which criticality is possible. The possible segregation of the fuel in dissolver operations and burner operations may require further consideration and evaluation.

The currently projected funding levels required to accomplish all phases of the research program, during the time intervals specified, are indicated by fiscal year in the table below.

TABLE III
PROJECTED FUNDING SCHEDULE

	<u>FY-1977</u>	<u>FY-1978</u>	<u>FY-1979</u>	<u>FY-1980</u>	<u>FY-1981</u>	<u>FY-1982</u>	<u>FY-1983</u>
Experiments and Data Correlation	50	190	220	250	250	200	100
Experimental Vessels (3)	18	---	---	---	---	---	---
Load-In Station Transfer Lines, etc.	10	---	---	---	---	---	---
Safety and Interlock System	10	---	---	---	---	---	---
Storage Tanks (2)	--	---	45	---	---	---	---
Fuel Preparation and Reprocessing							

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APPENDIX
HTGR FUEL CYCLE
PROCESS STEPS AND MATERIAL FORM
PERTINENT TO CRITICALITY PREVENTION

PROCESS STEPS TO WHICH CALCULATIONAL OR EXPERIMENTAL WORK IS RELATED	MATERIAL FORM ENCOUNTERED IN PROCESSES WHICH ARE PERTINENT TO CRITICALITY	DATA AVAILABILITY AND NEEDS
I. FUEL FABRICATION		
1. RECEIPT AND STORAGE OF EITHER URANYL NITRATE SOLUTION OR UO ₃ POWDER FOLLOWED BY DISSOLUTION (IF UO ₃) PURIFICATION, AND LOADING AND DRYING OF CATION EXCHANGE RESINS. (IF SOL-GEL PROCESS IS USED IN MICROSPHERE PRODUCTION, THE SAME MATERIAL FORMS WILL BE ENCOUNTERED EXCEPT POSSIBLE UF ₆)	DRY UO ₃ , UN AND UF ₆ (IF LOAD RESINS FROM UF ₆) SOLUTIONS, AND LOW HYDROGENOUS UO ₂ AND U (U-C ₆ O ₂ H ₆)	EXPERIMENTAL DATA EXISTS FOR SOLUTIONS CONTAINING LESS THAN 333 g U/l (ORNL-4280, p. 55, AND ORNL 60-4-12) AND FOR A FULL DENSITY METAL SPHERE (LAMS-3067, p. 54). DATA IS NEEDED FOR SOLUTIONS ABOVE 333 g/l IF ENRICHED URANYL FLUORIDE IS TO BE USED IN LOADING THE RESIN. ALSO, DATA IS NEEDED FOR DRY AND LOW MODERATED OXIDES. SOME DATA EXIST FOR ARRAYS OF UN SOLUTION UNITS (ORNL-4280, p. 55, AND NUCLEAR APPLICATIONS, VOL 4, p. 136); HOWEVER, ADDITIONAL DATA ARE NEEDED ON SOLUTION ARRAYS AND ON ARRAYS OF DRY OXIDE UNITS REFLECTED WITH WATER AND CONCRETE.
2. STORAGE AND CARBONIZATION OF DRIED LOADED RESINS FOLLOWED BY INSPECTION AND STORAGE OF CONVERTED CARBONIZED PARTICLE	U-C ₆ O ₂ H ₆ AT 15 H/U (MIN.) AND NON-HYDROGENOUS CONVERTED CARBONIZED RESIN PARTICLE AT ABOUT 3 C/U (UC ₃ O _{0.5})	NO DATA EXIST WHICH ARE APPLICABLE TO THESE PROCESS STAGES. DATA ARE NEEDED ON LOW MODERATED OXIDE AND CARBIDE SYSTEMS
3. PARTICLE COATING, INSPECTION AND STORAGE	CARBON COATED UC ₃ O _{0.5} PARTICLES (~400 μ m) IN A HYDROCARBON GAS (C ₃ H ₆ OR CH ₄)	NO DATA EXIST WHICH ARE APPLICABLE TO THESE PROCESS CONDITIONS. DATA ARE NEEDED ON LOW MODERATED OXIDE AND CARBON SYSTEMS.
4. URANIUM-THORIUM PARTICLE BLENDING AND ROD FABRICATION FOLLOWED BY ROD INSPECTION AND STORAGE (IF SOLUTION BLENDING AND CO-MICROSPHERE PREPARATION ARE USED, MIXED Th-U WILL BE PRESENT THROUGHOUT FABRICATION.)	CARBON COATED U AND Th PARTICLES AT A Th/U RATIO BETWEEN ZERO AND 20 (NORMALLY 4.25), AND C/U OF FROM ~ 15 TO 500 (NORMALLY ~ 15 TO 115). (RODS ARE 1/2-5/8 INCHES IN DIAMETER AND 2-3 INCHES IN LENGTH)	DATA EXIST FROM REACTIVITY WORTH MEASUREMENTS OF RODS IN GRAPHITE, PERFORMED IN SUPPORT OF HTGR DESIGN (GA-8468, 1968); HOWEVER, THE FUEL USED IN THESE EXPERIMENTS CONTAINED NO THORIUM AND NO C/U RATIOS BELOW 432 (THE LATTER IS PROBABLY NOT IMPORTANT, SINCE GOOD CALCULATIONAL RESULTS WERE OBTAINED FOR C/U RATIOS BETWEEN 432 AND 2500). MEASURED k _∞ VALUES EXIST AT HIGH C/U (ABOVE 2500) AND Th/U RATIOS (ABOVE 10) FOR Th-U GRAPHITE MIXTURES (BNWL-1561, BNWL-1601)
5. FUEL ELEMENT ASSEMBLY	HEXAGONAL GRAPHITE BLOCKS, 31-IN. LONG AND 14-IN. ACROSS THE FLAT, CONTAINING SOME 2000 FUEL RODS	DATA ARE AVAILABLE AS INDICATED ABOVE IN 1.4
6. FUEL ELEMENT CARBONIZATION AND ANNEALING	WATER IMMERSION OF FUEL ELEMENTS CONSIDERED POSSIBILITY	NO DATA
7. FUEL ELEMENT INSPECTION, STORAGE AND SHIPMENT	DRY FUEL ELEMENT	DATA EXIST FOR DRY FUEL ELEMENTS, AS COVERED IN 4 ABOVE; HOWEVER, NO DATA EXIST ON ARRAYS OF ELEMENTS SUCH AS WILL BE ENCOUNTERED DURING FUEL ELEMENT STORAGE
II. FUEL REPROCESSING		
1. RECEIPT AND STORAGE OF FUEL ELEMENTS	WATER REFLECTED DRY FUEL ELEMENTS	DATA EXIST FOR DRY FUEL ELEMENTS (SEE 1.4); HOWEVER NO INFORMATION ON ARRAYS, DRY OR FLOODED.
2. FUEL ELEMENT CRUSHING FOLLOWED BY WEIGHING, STORAGE, AND PRIMARY BURNING	CARBON COATED U AND CARBON COATED Th-U AND GRAPHITE AGGREGATES HAVING C/U RATIOS ~ 3 TO ~ 5000. (IF STRATIFICATION OCCURS TO THE EXTENT THAT FISSILE, FERTILE AND GRAPHITE PARTICLES SEPARATE, A MINIMUM C/U OF 3 IS THEORETICALLY POSSIBLE, ALTHOUGH A RATIO OF ABOUT 60 C/U IS CONSIDERED MOST CREDIBLE.)	NO DATA EXIST THAT ARE APPLICABLE TO THIS TYPE SYSTEM. SOME REACTIVITY WORTH MEASUREMENTS IN SUPPORT OF THE HTGR DESIGN EXIST (1.4); HOWEVER, THIS DATA DOES NOT COVER THE C/U RATIO RANGE BELOW 400 AND, THUS, THE NEUTRON ENERGY SPECTRUM IS MUCH SOFTER THAN THAT ENCOUNTERED IN THESE PROCESS STEPS. ALSO, THE EFFECTS THAT Th HAS ON REACTIVITY CANNOT BE VERIFIED FROM THESE HTGR EXPERIMENTS.
3. CLASSIFICATION AND STORAGE	CARBON-COATED U AND CARBON-COATED Th-U PARTICLES (INITIALLY, BOTH U AND Th-U PARTICLES WILL BE SIC COATED. EVENTUALLY, ONLY THE FISSILE U PARTICLES WILL BE SIC COATED) AT ~40 Th/U.	
4. CRUSHING AND SECONDARY BURNING	CARBON COATED U PARTICLES OR CARBON COATED Th-U PARTICLES (BOTH MAY HAVE SIC COATINGS ALSO, AS DISCUSSED IN PRECEDING PROCESS STEP). CARBIDE PARTICLES, UC ₃ O _{0.5} , ARE CONVERTED TO OXIDES IN THIS STEP.	
5. DISSOLUTION AND CENTRIFUGE	BURNER ASH FROM SECONDARY BURNER CONTAINING ThO ₂ AND UO ₂ . SLURRY OF DISSOLVED ThO ₂ AND UO ₂ AND PARTICLE HULLS, SIC HULLS AND C IN HNO ₃ -HF. NEUTRON POISONS MAY BE PRESENT.	
6. ACCOUNTABILITY TANK, ACID STRIP COLUMN, SOLUTION ADJUSTMENT AND DECONTAMINATION SOLVENT EXTRACTION COLUMNS	THORIUM-URANIUM NITRATE SOLUTIONS	NO EXPERIMENTAL DATA EXIST FOR Th-U SOLUTIONS
7. THORIUM-URANIUM PARTITIONING	THORIUM NITRATE SOLUTION AND URANIUM NITRATE SOLUTION	NO EXPERIMENTAL DATA EXIST FOR Th-U SOLUTIONS
8. URANIUM NITRATE SCRUBBING, CONCENTRATION AND STORAGE	URANYL NITRATE UP TO 350 g U/l	EXPERIMENTAL DATA EXIST FOR SOLUTIONS CONTAINING LESS THAN 333 g U/l (ORNL-4280 AND ORNL 60-4-12).
9. DENITRATION AND CALCINATION	URANYL NITRATE UP TO 350 g U/l AND DRY UO ₃	

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