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RICHLAND, WASHINGTON

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DOUGLAS UNITED NUCLEAR, INC.
ANNUAL REPORT OF
RESEARCH AND DEVELOPMENT

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Manager, Technology
TECHNICAL DIVISION

June 12, 1970

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TABLE OF CONTENTS

	<u>Page</u>
INTRODUCTION	3
BASIC PRODUCTION	6
K Reactor Zircaloy Hydride	9
Reactor Plant Life	13
Plant Cost Reduction	16
PRODUCT FLEXIBILITY	19
Enriched Fuel Program	21
Pu-238	24
Other Isotopes	29
ENVIRONMENTAL AND REGULATORY TECHNOLOGY	34
Regulatory	37
WASTE MANAGEMENT	51

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INTRODUCTION

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DOUGLAS UNITED NUCLEAR, INC.
ANNUAL REPORT OF RESEARCH AND DEVELOPMENT

GENERAL

In FY-1970, the former ten missions which had existed under the Richland Five-Year 02 Research and Development Program were consolidated into four program areas as follows:

Basic Production
Product Flexibility
Environmental and Regulatory Technology
Waste Management

Each program area has included one or more separate distinct programs, some of which were basic to sustaining and improving production operations while others were directed to broadening plant versatility for producing potentially marketable products. Midway through FY-1970, the shutdown of KW Reactor with subsequent curtailment of funding resulted in a substantial reduction in the level of effort in both of these directions. Consequently, during the balance of FY-1970 improvement effort has primarily been concentrated on N Reactor operation, and plant versatility effort was virtually stopped.

The research and development reported herein represents progress made during the period January 1, 1970, through May 30, 1970, and, contrary to recent annual and semiannual reports, includes only that work primarily funded with 02 R&D funds. No process technology funded activities are included, these being reported regularly in Douglas United Nuclear Monthly Reports.

FINANCIAL PERFORMANCE

The expenditure pattern of 02 program research and development funds for FY-1970 is summarized below and compared to the budget for FY-1971.

(Dollars in Thousands)	<u>FY-1970</u>		<u>FY-1971</u>
	<u>Budget</u>	<u>Expenditures</u>	<u>Budget</u>
<u>Basic Production</u>	\$ 480.0	\$ 579.0	\$ 250.0
K Reactor Zircaloy Hydride			
Reactor Plant Life			
Plant Cost Reduction			
<u>Product Flexibility</u>	200.0	178.0	-
Enriched Fuel Program			
Pu-238			
Other Isotopes			
<u>Environmental and Regulatory</u>	720.0	725.0	675.0
<u>Technology</u>			
Regulatory			
<u>Waste Management</u>	125.0	43.0	-
Total	\$1525.0	\$1525.0	\$ 925.0

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HIGHLIGHTS OF R&D PROGRESS

As mentioned earlier, R&D progress reported herein has been affected significantly by two factors: the decision to report only that work funded with O2 R&D funds, and the marked reduction in the level of effort as the result of manpower and funding limitations imposed subsequent to the shutdown of KW Reactor. Highlights of progress made include:

Successful development and demonstration in a ten-tube reactor test of an alternate case hydride removal process involving an anodic stripping operation followed by an electrolytic etch to supplant the grit blasting process developed earlier.

Prediction of a turnaround of K reactor graphite moderator growth in 1973-74 based on completion of irradiation of 18 KW Reactor graphite samples in the ETR.

Completion of laboratory development of a process to apply oxidation-resistant coatings to graphite moderators using organosilanes in the gas atmosphere.

Satisfactory progress in the program to develop and demonstrate oralloy technology up to the point of program cancellation.

Development of fabrication techniques and procedures for advanced thin annulus target elements containing thin-walled hollow cylinders of pressed and sintered NpO_2 -aluminum cermet for the production of medical-grade Pu-238.

Determination that tritium production in supplemental control splines is not feasible based on irradiation of spline samples containing three forms of lithium.

Continuation of successful metal-water reaction tests until deferred to FY-1971 by funding limitations.

Further refinement of the travel time for liquid effluents from the 1301-N crib to the river at 9±1 days.

Studies indicate that the wind regime around 100-N Area is of a complex nature such that a release of material would pass over the Wahluke Slope 43 percent of the time.

All reactor building air system filter cells are operating at a level of 99.95+ for iodine removal capability.

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BASIC PRODUCTION PROGRAM

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U. S. ATOMIC ENERGY COMMISSION

Additional Justification for Operating Costs

HAN-

SCHEDULE 189

RICHLAND
OPERATIONS OFFICE

02 Research and Development
PROGRAM

1. Contractors: Douglas United Nuclear		Contract No. AT(45-1)-1857	Task No.
2. Project Title: Basic Production		189 No.	3. Budget Activity No. 02-50-00-00
4. Date Prepared: 4-8-70	5. Method of Reportings: Semiannually	6. Working Locations: 100 and 300 Areas	
7. Person in Charge: R. K. Robinson Principal Investigators: J. R. Bolliger, P. A. Carlson		8. Project Term: Continuing From: To:	

9. Man Years:	<u>FY 1970</u>	<u>FY1971</u>	<u>FY1972</u>
(a) Scientific and Technical	12.0	1.00	1.00
(b) Other	3.0	.25	.25
Total	<u>15.0</u>	<u>1.25</u>	<u>1.25</u>
10. Costs:	<u>FY 1970</u>	<u>FY1971</u>	<u>FY1972</u>
(a) Direct Labor (Including Continuity of Service)	320.0	34.0	34.0
(b) Materials, Technical Services, Subcontracts	81.0	171.0	171.0
Total Direct	<u>401.0</u>	<u>205.0</u>	<u>205.0</u>
(c) Indirect Expenses	79.0	45.0	45.0
Total Operating Cost	<u>480.0</u>	<u>250.0</u>	<u>250.0</u>
(d) Equipment (New Obligational Authority)			
(e) Irradiation Unit Cost			

INTRODUCTION

Research and development in support of the Basic Production Program for the Hanford reactors is aimed at developing and maintaining the technology required for support of the continued economical operation and enhancement of the productive capability of the K and N reactors. This is based upon continuing utilization of the Hanford reactors as efficient, economical and long-life producers not only of weapons and nonweapons grades of plutonium, but also other special nuclear materials for which the basic production processes have been identified.

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K REACTOR ZIRCALOY HYDRIDE

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I. SCOPE AND OBJECTIVES

The objective of this program is to determine the causes and mechanisms contributing to the hydriding of K reactor Zircaloy process tubes, promote measures to minimize additional hydriding, and develop methods and techniques for hydride case layer removal for restoration of the base metal.

II. MAJOR ACCOMPLISHMENTS TO DATE

Investigations have established that Zircaloy process tubes in low pressure reactors become hydrided in the warm water spacer section behind the active charge due to the action of a galvanic couple between aluminum spacers and the tube. In new tubes or in tubes that have been completely cleaned of hydride, the formation of hydride can be prevented by the use of anodized spacers. The rate of permeation of hydride into the base metal, with consequent embrittlement, has been determined and analysis completed of the probable future course of tube damage. A technique for removing hydride, already formed on the tube surfaces, has been developed and demonstrated at KE Reactor.

III. PROGRESS DURING REPORT PERIOD

Mechanisms

The study of the hydriding mechanisms was terminated with the shut-down of KW Reactor. Analyses of the coupons which had been exposed in the outlet section of KW Reactor tubes for up to 14 months have been completed, and have led to the following conclusions:

1. Bare aluminum has the greatest hydriding potential. Anodized aluminum will cause some hydriding if good electrical contact is maintained, and stainless steel will not cause hydriding itself, but is not able to prevent all further hydriding when a case layer is already present.
2. The presence of a case layer of hydride platelets on the inner surface enhances the continued buildup of hydrogen in the base metal.
3. Removal of all traces of case layer from previously hydrided surfaces restores them to a condition where anodized aluminum spacers will not cause further hydriding.

Case Layer Removal

An alternate hydride case layer removal process, to supplant the grit blasting process developed earlier, was developed and demonstrated at KE Reactor. This process consists of two steps: first, an anodic

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stripping operation in which approximately 0.1 amp per cm² current is applied in a 3 percent nitric acid electrolyte; and second, an electrolytic etch with a HNO₃:HF electrolyte. The first step selectively attacks the hydride case layer and removes it at a rate of about 0.1 mil per minute. It removes all of the massive continuous layer but does leave some of the fine needles which normally are found at the case layer-base metal interface. The second step uniformly etches the surface leaving it in a highly polished condition, free of all traces of the case layer. Removal of 1 to 2 mils by this second step is adequate to complete the cleaning operation. Although this combined process results in a micropolished surface, it does leave a macro-roughness identical to the interface between the case layer and the base metal. These on-reactor tests demonstrated that the depth of case layer formation varies greatly within individual tubes and between tubes.

The test was very successful, and there appears to be no obvious reason why the technique cannot be adopted for full-scale reactor use.

Tube Examination

The downstream 66 inches of 25 C Reactor Zircaloy tubes were removed to be used in the electrochemical cleaning development. Sampling of these tubes revealed that five of them were heavily hydrided, with the balance only lightly hydrided. One of these tubes, 1076-C, did contain the greatest amount of hydrogen ever found in a Zircaloy tube, 458 ppm base and 16 mils maximum case layer thickness. For unknown reasons, the hydriding of the C Reactor tubes is even more erratic than at the K reactors. Although there was no evidence that the above-mentioned tube had failed due to hydriding, it was found that its physical properties more nearly resembled a cermet than a metal.

IV. EVALUATION OF EFFORT

Work progressed on schedule and the electrolytic method of cleaning the tubes was demonstrated successfully at KE Reactor. This technique was firmly established. Monitoring of the results will require approximately a year to determine the ultimate value of cleaning as a means of controlling the hydride problem. Further optimization of the process, originally planned for FY-1970, was deferred due to budget cuts associated with the KW Reactor shutdown.

V. FUTURE PLANS

The on-reactor demonstration test involved anodic stripping of five tubes, together with anodic stripping and electrolytic etch of five additional tubes to allow evaluation of the need and effectiveness of the stripping step alone or in combination with the etching

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step. One of each of the two groups of tubes was removed immediately after cleaning to determine the effectiveness of the cleaning process. The other eight will be removed by pairs at intervals over the next 12 to 18 months to determine if any deleterious effects have resulted from the operation, and to determine if there has been any resumption of hydriding.

The commitment of additional funds and programs for full-scale removal of the hydride case layers from K reactor Zircaloy tubes depends on AEC long-range plans for the reactors.

VI. DOCUMENTS ISSUED

DUN-6387, "K Hydriding Failure Analysis," WK Alexander, November 3, 1969. (Secret)

DUN-7000 KO, "K Reactors Zircaloy Hydriding Program Plan," JM Lutton, October 8, 1969.

DUN-6680, "Production Test Authorization-204, Electrochemical Hydride Removal and Evaluation," WK Alexander and CH Gydesen, January 30, 1970. (Confidential)

Letter, CH Gydesen to Distribution, "Interim Report, Production Test Authorization-204, On-Reactor Experience and Equipment System Evaluation," March 9, 1970.

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REACTOR PLANT LIFE

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I. SCOPE AND OBJECTIVES

Assure that the ability of the Hanford reactors to produce irradiation products economically does not become constrained by irreplaceable facilities. Irreplaceable facilities are defined as those which cannot be replaced without essentially rebuilding the entire complex (the graphite stack or reactor shields for example) or those which are so expensive to replace that economic considerations suggest reactor decommissioning as the more palatable alternative (external piping complex or water treatment system for example).

II. MAJOR ACCOMPLISHMENTS TO DATE

Irradiation of both N and K reactor graphite to exposures well beyond those current in the reactors, as well as continued observation and measurement of the graphite stacks, has provided bases for the development of procedures and methods for coping with future problems, particularly those posed by contraction, thus insuring continued operability.

III. PROGRESS DURING REPORT PERIOD

Irradiation of the 18 graphite samples removed from KW Reactor in 1966 was completed in the ETR. Samples irradiated at normal K reactor stack temperatures experienced turnaround at an exposure of about 10^{22} n/cm² and with a total contraction of 3 percent. This is equivalent to about 20 years of K reactor operation, which would indicate turnaround of the K reactor stacks in 1973-74.

The irradiation of large blocks and small samples in N Reactor to determine the effects of size and stress on dimensional changes continued. Latest results indicate the contraction rate of large blocks has been 32 percent greater than that of small samples at exposures up to 1.5×10^{21} n/cm².

The laboratory development of a process to apply oxidation-resistant coatings to the graphite moderators with organosilanes in the gas atmosphere was completed and documented. Coated graphite showed improved oxidation resistance by about a factor of 7 compared to standard graphite controls in tests performed at 900 C in simulated N Reactor atmospheric conditions. Effective coatings can be applied using tetraethyl or tetramethylsilanes at normal N Reactor stack temperatures. The coating reagents were found to be stable in the radiation environment. Coated graphite samples irradiated to 7.33×10^{21} n/cm² in the ETR at temperatures of 700 to 900 C experienced no visible deterioration. It was concluded that the probability of success in applying oxidation-resistant coatings to Hanford reactor moderators was good provided adequate application techniques could be developed.

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IV. EVALUATION OF EFFORT

The laboratory phase of the development of a graphite coating process which could be used to apply oxidation-resistant coatings to the existing reactor moderators was successfully completed as scheduled in FY-1970.

Satisfactory progress was made in graphite irradiation programs. These are continuing programs and progress is directly dependent on the rate of accumulation of exposure in the N and test reactors.

V. FUTURE WORK

The ETR data from the irradiation of graphite samples removed from KW Reactor will be analyzed to determine the value of this technique to predict long-range behavior of the reactor moderators. Assuming a positive result, TSX graphite samples will be removed from N Reactor for irradiation in the ETR to establish the long-range distortion behavior of the N Reactor moderator. In addition, the irradiation of samples of TSX graphite, irradiated in the GETR in 1964-1965, will be completed in ETR for comparison with the core samples.

The irradiation of large and small graphite samples in N Reactor to evaluate the "size-effect" on graphite distortion will be completed.

VI. DOCUMENTS ISSUED

DUN-6807, "Graphite Coating Studies for Prolonged Reactor Life, R. Cooperstein, April 30, 1970. (Secret)

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PLANT COST REDUCTION

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I. SCOPE AND OBJECTIVES

The Plant Cost Reduction Program was intended to adapt existing technology or develop new technology which, when implemented, would result in a direct reduction in plant operating cost or avoidance of costs which can be projected on the basis of current experience and conditions.

This program concentrated on adapting process computer technology for use at N Reactor. The plant process surveillance system utilizing a central processing unit, peripheral equipment, and data acquisition systems was directed toward three main objectives:

1. Direct plant cost reduction.
2. Increased plant reliability and dependability.
3. Enhanced attainment of licensability status.

II. MAJOR ACCOMPLISHMENTS TO DATE

The feasibility study which was completed and approved in July 1968 concluded that adapting process computer technology to N Reactor operations was indeed feasible and showed attractive returns on invested capital. The systems study was completed. In selecting systems or components for application of automatic surveillance, principal emphasis was placed on Process Standards, systems critical to nuclear safety, prestartup checks and consolidation of control in the 105-N control room. Software programs were identified and program requirements specified. Hardware requirements were established based upon performance and reliability requirements.

III. PROGRESS DURING THE REPORT PERIOD

There was no progress during the report period. The completed systems study was submitted to the Atomic Energy Commission for approval and implementation. No further effort was expended on this program as the result of curtailment of funds.

IV. EVALUATION OF EFFORT

The in-depth study of detailed application of automatic data processing technology to N Reactor was successfully completed. Specific systems and process parameters were identified. System and equipment requirements were established. Cost and other benefits were determined. The study concluded that procurement and installation of a process computer system was justified, even under the austere budgetary climate existing, and recommended approval of the program.

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V. FUTURE PLANS

There are no future plans for this program.

VI. DOCUMENTS ISSUED

DUN-6356 REV1, "Plant Process Surveillance System for N Reactor -
Systems Study," Staff of Instrument and Automation Technology,
November 8, 1969. (Confidential)

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PRODUCT FLEXIBILITY PROGRAM

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INTRODUCTION

The Product Flexibility Program was aimed at broadening plant versatility for producing potentially marketable products in addition to defense and nondefense plutonium, and demonstrating that Hanford production reactors are economically capable of such versatility.

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I. SCOPE AND OBJECTIVES

The objective of the Enriched Fuel Program was to develop and demonstrate the use of oralloy and plutonium bearing fuels for the production of weapons or nonweapons material in the Hanford reactors. Principal effort was devoted to utilization of a low enrichment alloy driver (<5 percent oralloy) at the single pass reactors, with some design work directed at a fuel-target combination with driver concentrations of oralloy up to 15 percent. Reduced funding levels resulted in cancellation of the program following shutdown of KW Reactor.

II. MAJOR ACCOMPLISHMENTS TO DATE

Preliminary nuclear safety analyses, gamma heating calculations, PCTR measurements and preliminary test design studies were discussed in the Douglas United Nuclear, Inc. Semiannual Research and Development Report. The major accomplishments since that period and prior to cancellation of the program include:

1. All pertinent nuclear safety, physics, thermal hydraulics, and fuel design calculations for a small scale K reactor test involving fuel with less than 5 percent oralloy were completed and issued as a technical bases document.
2. A temperature and burnup dependent fuel swelling analysis was completed and documented as part of the transient behavior analysis.
3. A final production test draft for the small scale low enrichment oralloy bearing fuel test was completed.
4. Core casting and hot-die-sizing parameters were established for solid and I&E oralloy-aluminum fuel designs containing up to 15 percent oralloy.
5. The production test draft and program status as of the program termination date were documented for possible future reference.
6. An analysis of plutonium bearing fuel utilization in a K reactor with respect to maximum throughputs and core fraction which could be tolerated without introducing unanalyzed nuclear safety considerations was completed. Conclusions were transmitted to the Atomic Energy Commission through participation on a Hanford-wide Task Force.

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III. PROGRESS DURING REPORT PERIOD AND EVALUATION OF EFFORT

The program progressed satisfactorily during the current reporting period to the time of cancellation. Fuel fabrication efforts were halted just prior to beginning work on the small scale test requirements. The test authorization document was ready for authorizing signatures in preparation for scheduled May 1, 1970, reactor charging. The program status has been documented to minimize the effort required in the event future developments promote its reactivation.

IV. FUTURE PLANS

There are no future plans for further development of an Enriched Fuel Program.

V. DOCUMENTS ISSUED

DUN-6618, "Technical Bases of K Reactor Oralloid Production Test," GR Correll, January 12, 1970. (Secret)

DUN-6699, "The Calculation of Volume Increases in Fuel Elements during Transient Temperature Excursions," HA McDonald, February 26, 1970.

DUN-6757, "PTA-Draft, K Reactor Oralloid Fuel Irradiation," GR Correll, February 27, 1970. (Secret)

ARH-AOP-18, "Plutonium Fueling at Richland, Interim Report," Task Force: OF Hill, GC Fullmer, WK Woods, et al, January 16, 1970. (Secret)

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Pu-238

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I. SCOPE AND OBJECTIVES

The objective of the program is to increase the national production of Pu-238 in as efficient and economical a manner as possible by maximizing the production of Np-237 in basic production loads and by identifying optimum methods for efficient Np-237 utilization and relatively rapid conversion to Pu-238 in the DUN-operated reactors. The production of clean Pu-238 (low Pu-236 content) will be further developed. The technology for producing Pu-238 by irradiating Am-241 will also be developed.

To maximize the production of neptunium, principal effort is placed on better quantification of the conversion rates of Np-237 from U-236 fuel management techniques for maximizing the U-236 in the fuel streams within economic limitations, and optimum utilization of uranium fuel in the reactors consistent with production requirements.

In producing Pu-238 from the irradiation of Np-237, several factors must be considered in accelerating the conversion rate while maintaining efficiency. Emphasis will be placed on optimizing the neutron flux level, the related target and reactor load design, and the reactor exposure per cycle.

Development work will continue on processes for producing clean Pu-238. Investigations will involve defining the effects of reactor loading patterns as well as the design of alternate target geometries, e.g., the containment of the neptunium in a thin annulus.

The significant availability of Am-241 forecasted for the 1970s from power reactor plutonium tails suggests the irradiation of Am-241 as an attractive supplemental route for producing Pu-238. The incentives for developing the technology for producing Pu-238 from Am-241 will be assessed and development carried out if feasibility is established.

II. MAJOR ACCOMPLISHMENTS TO DATE

- A. Predictions of Np-237 production over a wide range of exposures in unirradiated uranium fuel and in recycled uranium fuel with moderate levels of U-236 have been confirmed by measurements.
- B. Pu-238 production rates from irradiation of neptunium have been established by experimental measurements.
- C. An improved method for irradiating neptunium utilizing a higher neutron flux level and a facility for the charge and discharge of the target elements during operation was successfully demonstrated in a K reactor.

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- D. A moderate scale Pu-238 production proposal for immediate implementation using existing Hanford facilities was submitted to the AEC.
- E. The production of Pu-238 containing 0.3 ppm Pu-236 was demonstrated by irradiating neptunium in a graphite matrix.
- F. An irradiation test to investigate the growth of NpO₂-graphite matrix targets during irradiation has been completed and the technology for use of this type of target is now available if required.
- G. A 2x2 column cluster test of advanced thin-annulus neptunium targets has been charged into the reactor.

III. PROGRESS DURING REPORT PERIOD

Neptunium Irradiation

An irradiation test was conducted to determine if nominal changes in design and processing parameters could cause the observed dimensional instability of graphite matrix target elements during irradiation. The one-column test of eight targets was discharged from KW Reactor on February 1 after 20 days of operation. A preliminary analysis of the data shows that (1) graphite filler wafers grew an average of 3 percent in length, (2) one group of NpO₂-graphite matrix wafers which were designed to operate approximately 100 C higher than other matrix wafers grew 2 percent, and (3) there was little difference between the growth of NpO₂-graphite matrix wafers fabricated using different process variables, the average growth being about 6 percent. It appears that earlier matrix targets were irradiated successfully because of provision of adequate expansion space and use of annealed cladding (which would yield uniformly) to accommodate the growth that occurred. A final report on this work is being prepared.

Fabrication techniques and procedures were developed for advanced thin-annulus target elements containing a series of thin-walled hollow cylinders of pressed and sintered NpO₂-aluminum cermet. Twenty elements, ten with a graphite core and ten with a water core, were fabricated and tested. These elements were charged into a 2x2 column cluster in KE Reactor on May 10. Ten columns of support enrichment were provided. Approximately 3 months of irradiation time will be required to reach goal exposure.

A comparison was made between the design of a thin-annulus neptunium target element and a target design in which neptunium pins were circumferentially located in a graphite core. Calculated absorption rates relative to a thin-annulus element varied from

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83 percent for a 4-pin element to 97 percent for a 16-pin element. The pin-type design was abandoned because of higher fabrication cost and computed excessive pin temperature during irradiation.

The thin-annulus test has three objectives: (1) to demonstrate the technology for fabrication and irradiation of NpO_2 targets which eliminate the problems associated with dissolution of graphite matrix elements, (2) to demonstrate production of medical-grade Pu-238 with a Pu-236 contamination level near 0.2 ppm at a neptunium conversion of 10 percent, and (3) to evaluate the relative advantages of graphite core versus water core elements. Calculations indicate that the conversion efficiency of the elements with the removable graphite core will be higher than that of the water core element.

All of the 30 neptunium elements irradiated during CY-1969, including the ruptured and sampled elements, have been shipped to Battelle-Northwest for dissolution and product recovery. The targets are to be dissolved in three batches, permitting separate chemical analysis to be performed on each batch. The three batches consist of the 12 Np-Al elements irradiated in the core of KW Reactor, the 12 neptunium-graphite elements from the core, and the 6 neptunium-graphite elements irradiated in the fringe. In addition, the NpO_2 -graphite matrix wafers irradiated during January 1970 have been shipped to Battelle-Northwest for processing.

Americium Irradiation

The irradiation of the 150 grams of americium oxide, charged into KE Reactor during May 1969, is continuing. Discharge of these elements is planned during the first quarter of FY-1971.

IV. EVALUATION OF EFFORT

Progress has continued on this program at a relatively low level consistent with funding and manpower limitations. Although the program received an unexpected setback with the failure of the neptunium oxide-graphite target elements a year ago, a graphite matrix target element irradiation program could now be carried out successfully in the Hanford reactors if required.

Development of advanced thin-annulus technology has gone ahead at a rapid pace. In the fabrication phase, procedures and techniques were worked out to consistently produce elements that would meet all test requirements. With some equipment additions, relative large quantities of material could be fabricated in these geometries. Irradiation proof testing is now in progress and at this time these types of targets, which are advantageous from a separations standpoint, appear to be best suited for production of medical-grade Pu-238 in the Hanford reactors.

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The graphite matrix neptunium target elements which were irradiated in a single tube produced Pu-238 containing 0.29 to 0.33 ppm Pu-236. Calculations indicate that irradiation of these targets in a cluster of four tubes should lower the Pu-236 concentration to about 0.20 ppm.

V. FUTURE PLANS

The shutdown of the KW Reactor was accompanied by substantial reduction of funds available for research and development. At this time, it has been decided to discontinue further work on development of Pu-238 technology, despite the favorable capability of KE Reactor as a low cost producer of Pu-238 low in Pu-236.

Arrangements have been made for the irradiated thin-shell neptunium targets to be processed at Battelle-Northwest during the second quarter of FY-1971 with funds provided by the Atlantic Richfield Hanford Company. If the production of several tens of grams of Pu-238 containing only 0.2 ppm of Pu-236 can be demonstrated, it is possible that further financial support may be obtainable from interested customers.

No provision has been made for analyzing or processing the americium targets after discharge from the reactor. It has been planned all along to let these elements cool until most of the Cm-242 had decayed to Pu-238 (2 or 3 years) before processing.

VI. DOCUMENTS ISSUED

DUN-6812, "Production Test Authorization 209, Improved Neptunium-237 Target Element Configurations," May 5, 1970. (Secret)

DUN-6841, "Conversion Calculations for Thin-Annulus Neptunium Element Design," AF Kupinski, March 23, 1970. (Secret)

DUN-6939, "Comparison of Neutron Absorption in Pin Versus Thin-Annulus Targets," SP Roblyer, March 26, 1970.

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OTHER ISOTOPES

I. SCOPE AND OBJECTIVES

The objective of the Other Isotopes Program is to enhance the capability of the Hanford reactors to produce various isotopes and to determine the cost and capability of production for those isotopes not included in other activities.

II. MAJOR ACCOMPLISHMENTS TO DATE

A. Tritium production has been demonstrated in the K reactors using 94 Metal. Both the conversion rates and the economics of the process have been established.

B. The N Reactor coproduct demonstration program was completed and achieved all of its objectives.

It should also be noted the initial portion of the test demonstrated 4800 MW reactor power level capacity for the reactor operating parameters and critical reactor systems. This was proved conclusively by operation at 4800 MW for 31 hours.

C. The buildup rate of the higher plutonium isotopes and trans-plutonium up to Cm-244 has been predicted and normalized to results from a small scale irradiation.

D. The Cm-244 production and cost model has been completed and is available for planning optimized production campaigns.

E. A Pu-Al fuel model capable of operation of 100 kw/ft for extended periods of time has been designed and fabricated.

F. A cobalt-bearing process tube was irradiated in KW Reactor for the production of Co-60, but unexplained and unacceptable release of Co-60 radioactivity to the coolant water was encountered.

G. Cobalt strips have been irradiated in N Reactor, providing a potential source of data on N Reactor capability as a Co-60 producer.

H. N Reactor capability as a producer of thulium-170 was established.

I. The plutonium deep burn program has continued for 6 months at design level specific power.

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III. PROGRESS DURING REPORT PERIOD

A. Cobalt-60

The cobalt base alloy tube (Haynes alloy 25, nominally 51 percent cobalt) had been irradiated for 8 months at the time KW Reactor was shut down. This irradiation should have resulted in a cobalt-60 formation of approximately 18 curies/gram of cobalt, or 20,000 curies for the tube. The coolant from this tube was analyzed periodically to determine the corrosion rate, and subsequent release of cobalt-60 to the effluent stream. During the first 6 months of operation, the cobalt-60 content of the effluent from the tube built up slowly to about 4 pCi/ml as compared to a normal level in K reactor effluent of 0.2 pCi/ml. However, after the third charge-discharge, the outlet activity suddenly increased by a factor of 10 where it remained until the reactor was shut down. There is no explanation for this sudden change. The tube has been left in the reactor. It was monitored after shutdown for heat generation, but its temperature was only slightly above the normal Zircaloy tubes.

The four cobalt strips irradiated in N Reactor graphite cross-coolant tubes were discharged and moved to the fuel storage basin. Because of declining interest and lack of funding for cobalt irradiations, the element will simply be stored until a suitable customer can be found. Radiation readings through the side of the cask (~ 200 mR/hr) indicate the total production on the four strips is around 13,000 curies, or in the range of 50 curies/gram.

B. Thulium-170

Final calorimetric results were received on the 176 grams of thulium-169 charged into an N Reactor graphite cross-coolant tube in the fall of 1967. Extrapolation of calorimetry data back to time of discharge from the reactor indicated that an average production of 126 curies/gram was realized on the thulium. Based upon the actual irradiation history for the thulium, calculations imply that, in order to obtain this production, the thulium experienced an average Maxwellian flux of 3.1×10^{13} neutrons/cm²/second.

C. Plutonium Burning

The 13 PuAl columns in the single-column-width slab in KE Reactor continued to operate at design levels during the third and fourth quarters of FY-1970. Decreasing blackness of PuAl elements and changing reactivity requirements dictated periodic replacement of thoria and E-metal with natural uranium in order to maintain the desired specific power in the PuAl elements.

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D. Lithium Bearing Spline Development

Spline samples containing lithium in three different forms were irradiated for 3 months in KE Reactor. The three forms were: (1) lithium-magnesium alloy particles dispersed in an aluminum powder metallurgy matrix, (2) lithium-carbonate crystals dispersed in an aluminum powder matrix, and (3) lithium-fluoride crystals in an aluminum powder metallurgy matrix. All samples were extrusion clad with 3 to 6 mils of aluminum. All samples contained 3.2 wt% lithium which had been enriched to 95 percent lithium-6.

Samples of each composition were irradiated to three levels corresponding to 1, 1-1/2 and 2 months at the center of the reactor. After irradiation, it was found that all of the samples containing the lithium-magnesium had swollen at the end where they were exposed to the coolant. All of the samples containing the two lithium salts were extensively blistered, indicating release of gas under the cladding. On the basis of these tests, and the previously reported test on lithium-aluminum alloy splines, it is concluded that tritium production in supplemental control splines is not feasible.

IV. EVALUATION OF EFFORT

Additional valuable information on N Reactor production capabilities for cobalt-60 and thulium-170 was obtained.

Satisfactory progress is being made on the plutonium burning test. The data obtained while increasing the element power levels will enhance the calculational model used for predicting exposure and reactivity requirements. The successful completion of this test irradiation will be a demonstration of the Hanford capability for the irradiation of plutonium as a fuel material for californium-252 or other higher isotope programs.

The overall program is at a minimum level of effort restricted to those isotopes which warrant further attention due to long-term market potential. Other isotope programs, such as the coproduct study, which may also have future market potential, have been completed.

V. FUTURE PLANS

The shutdown of KW Reactor was accompanied by substantial reduction of funds available for research and development. Work on this program was terminated at the end of February 1970.

An initial 10 kilograms of plutonium is being irradiated in KE Reactor. It is planned to continue this irradiation for about 3 more years, discharging some of the tubes every 6 months. The

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first tube is to be discharged during the first quarter of FY-1971, but no plans have been made for the chemical processing and isotopic analysis of the irradiated plutonium.

VI. DOCUMENTS ISSUED

DUN-5479 SUPA, "Total Control Calculations for the PuAl Production Test (PTA-150) - Supplement A," AF Kupinski, JF Nemec, December 15, 1969. (Secret)

DUN-6789, "Final Report - PT-NR-89, Thulium-170 and 171 Production in N Reactor," AF Kupinski, March 6, 1970. (Secret)

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ENVIRONMENTAL AND REGULATORY
TECHNOLOGY PROGRAM

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U. S. ATOMIC ENERGY COMMISSION

HAN-

SCHEDULE 189

Additional Justification for Operating Costs

RICHLAND
OPERATIONS OFFICE02 Research and Development
PROGRAM

1. Contractors Douglas United Nuclear		Contract No. AT(45-1)-1857	Task No.
2. Project Title: Environmental and Regulatory		189 No.	3. Budget Activity No. 02-50-00-00
4. Date Prepared: 4-29-70	5. Method of Reporting: Semiannually		6. Working Locations: 100 and 300 Areas
7. Person in Charge: R. K. Robinson Principal Investigators: D. W. Peacock, R. H. Shoemaker, P. A. Carlson		8. Project Term Continuing From To:	

9. Man Years:	<u>FY 1970</u>	<u>FY1971</u>	<u>FY1972</u>
(a) Scientific and Technical	17.0	5.6	5.6
(b) Other	4.3	1.4	1.4
Total	<u>21.3</u>	<u>7.0</u>	<u>7.0</u>
10. Costs:	<u>FY1970</u>	<u>FY1971</u>	<u>FY1972</u>
(a) Direct Labor (Including Continuity of Service)	459.0	171.5	171.5
(b) Materials, Technical Services, Subcontracts	151.0	382.0	382.0
Total Direct	<u>610.0</u>	<u>553.5</u>	<u>553.5</u>
(c) Indirect Expenses	110.0	121.5	121.5
Total Operating Cost	<u>720.0</u>	<u>675.0</u>	<u>675.0</u>
(d) Equipment (New Obligational Authority)			
(e) Irradiation Unit Cost			

INTRODUCTION

The Environmental and Regulatory Program is designed to insure that Hanford production reactors are maintained and operated safely, within limits which are established upon sound technical bases, to obtain a better understanding of the contributing factors and consequences of chronic as well as postulated accidental releases of fission products, and to ultimately attain a reactor safety status which meets the same basic safety criteria which are currently applied to licensed power reactors.

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METAL-WATER REACTIONS WITH FULL SIZE FUEL ELEMENTSI. SCOPE AND OBJECTIVES

Experimentally determine the nature and extent of metal-water reactions with full size fuel elements to allow comparison, and to provide information to justify the use of an analytical model derived for the calculation of metal-water reactions in an N Reactor accident condition.

II. MAJOR ACCOMPLISHMENTS TO DATE

Seventeen metal-water reaction tests and three clad expansion tests have now been completed since the first irradiated fuel test of this type on April 15, 1969. Validity of the experimental technique was confirmed during earlier tests in 1968 and 1969. Measured hydrogen evolution from the Zircaloy-steam reaction on unirradiated fuel and process tubes was found to agree within 10 percent of the Baker-Just parabolic rate law that is being used throughout the nuclear industry.

Results from tests with irradiated fuel above the clad failure temperature indicate an initial reaction rate equation of parabolic form, changing towards a linear equation in the latter phases of the tests (in the period 8 to 60 minutes). Test after test has shown the same form of reaction equation, with the magnitude of hydrogen generation varying in a manner consistent with the observed physical characteristics of the failed fuel.

Measured hydrogen evolution is less than that predicted by analytical models used in current N Reactor safety analyses.

III. PROGRESS DURING REPORTING PERIOD

The metal-water reaction test program proceeded at a rapid pace during the first 7 months of FY-1970, with completion of 15 tests. These included seven metal-water reaction tests at temperatures above clad failure, but below uranium melting temperature, five tests above uranium melting temperature, and three special clad expansion tests. Principal results from these tests are as follows:

Non-Molten Tests

Clad failure temperature is nominally 1850 to 1900 F with an indication of somewhat lower clad failure temperatures at higher fuel exposures (well above normal goal exposure).

The physical form of the fuel after clad reaction controls the magnitude of the metal-water reaction. Outer elements fail on their inner clad surface and inner elements fail on their outer clad surface. The combined effects of uranium

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extruding from both portions of the elements into the inner annulus indicate a high probability that steam access to the fuel will be greatly inhibited, and fission product release will also be inhibited. Hydrogen release measurements confirm the visual observations. Fission product release data are still being reduced, but are also expected to demonstrate the blockage effects of failed fuel.

The summation of hydrogen evolution measurements from inner and outer fuel tests indicates a total hydrogen evolution below the value predicted by the analytical molten pool model documented and reported to regulatory groups in 1967.

Molten Tests

Initial hydrogen evolution test results are above the values predicted by the molten pool model. However, direct visual observation of the tests (seen through clear quartz tubes) indicated that the molten uranium poured out into pools at each end of the fuel piece, with each pool having a large surface area for reaction. In the reactor, such an effect could occur only on the fuel piece at each end of each fuel column, and these pieces operate at such low powers that melting could not occur for several hours after a loss-of-coolant accident. Methods of providing physical restraint to each end of the fuel piece to simulate in-reactor conditions are being explored.

Clad Expansion Tests

The three clad expansion tests were done with high exposure fuel (more than twice normal goal) to investigate effects seen on earlier tests at intermediate exposures in which the outer clad of outer elements was seen to expand away from the uranium. Results from these tests supported the earlier evidence, and are now being studied and documented. Detailed metallurgical photographic analysis of these fuels has been completed and will be reported.

Fission product release data (in curies) have been obtained from several tests, but results have not been reportable due to questions about the validity of a computer program (RIBD) for calculating fuel inventories of certain long-lived fission products. These questions are largely resolved, but one final comparison will be made with results from a special fuel dissolution test performed at the ARHCO facility. Following this comparison, fission product inventories of the test fuel pieces will be recalculated so that the curie release numbers can then be reported in terms of percent of inventory.

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No tests have been performed since February 3, 1970, due to limitations in funds. The expanded scope (and costs) of tests already completed in FY-1970 utilized all available funds.

IV. EVALUATION OF EFFORT TO DATE

Experimental measurement of metal-water reactions progressed very satisfactorily in the first 7 months of FY-1970. Measured hydrogen releases are very consistent and there is exceptional agreement with analytical models. With the program progressing at a rapid pace, it was a major disappointment to halt testing for 5 months due to lack of funds. During these 5 months, nine specially irradiated fuels were discharged from the reactor but could not be used. Fission product decay in these fuels has now proceeded to the point that their usefulness for fission product release measurement is gone. Three special tests were performed with high exposure fuel to investigate the clad separation (from fuel) phenomena seen on several earlier tests, and five metal-water reaction tests were performed with fuel heated into the molten uranium range.

Results from the test program with non-molten fuels have been consolidated by Battelle-Northwest into an interim report that is being reviewed within DUN. The consolidated results substantiate earlier reports that the magnitude of the metal-water reactions for failed but non-molten fuel is less than that indicated by analytical models developed and reported to regulatory bodies.

V. FUTURE WORK

Metal-water reaction tests with molten fuel will be restarted in FY-1971. The molten fuel tests should be completed within the first 6 months.

An interim report on results from the non-molten metal-water reaction tests will be issued shortly. A draft copy prepared by Battelle-Northwest is now being reviewed with DUN.

Difficulties with analytical determination of fission product inventories will be resolved and all previous fission product release measurements will be placed on a consistent basis.

Fission product releases from molten fuel will be measured.

Detailed planning of fuel quench tests will be started so that tests simulating the core performance during transition from normal to emergency cooling can be started in FY-1971. Clad shattering is not predicted for N Reactor fuel during these quench tests, as clad temperature will be well below the temperatures at which this phenomena has been seen by others. However, confirmation of clad integrity under rapid cooldown conditions is essential for the unique fuel used at the N Reactor.

VI. DOCUMENTS ISSUED

None.

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METEOROLOGICAL STUDIESI. SCOPE AND OBJECTIVES

The meteorological program is designed to investigate the atmospheric transport, diffusion, and deposition of postulated fission product releases from the Hanford production reactor sites. The objectives of this program are:

1. To define and measure the meteorological parameters which would describe the path, speed and dilution of radioactive materials released from a production reactor following a postulated accident.
2. To experimentally determine the dilution rates of tracers released at production reactor sites.
3. To provide a valid model which describes the atmospheric transport, diffusion and deposition on the Wahluke Slope of fission products released from a production reactor following a postulated accident.

The scope of the program includes the measurement and evaluation of meteorological parameters in the vicinity of the Hanford production reactor sites and over the Wahluke Slope. Meteorological parameters will be measured which are expected to have an effect on the transport, diffusion, and deposition of a plume during its passage over the slope.

The evaluation of meteorological data will include calculations of joint probabilities of wind direction, speed and variability as a function of atmospheric stability as measured at the N Reactor meteorology tower. This evaluation will also provide correlations of meteorological data between the N Reactor tower, portable meteorological measurement systems located on the Wahluke Slope, and the Hanford meteorology tower located on the 200 Area plateau.

A series of experiments involving tracer release and sampling is planned to simulate the diffusion and deposition characteristics of postulated fission products released from the production reactors. These tests will be conducted at distances that will provide data that will characterize the atmospheric diffusion over Wahluke Slope. A diffusion model, selected for its applicability to the geography surrounding N Reactor, will be used to compute time-integrated air concentration values for comparison with observed values.

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II. MAJOR ACCOMPLISHMENTS TO DATE

A meteorological data acquisition system was installed and put into operation at 100-N Area. Installation and operation of five remote portable wind stations was completed, two of these being automatic punched tape systems. From data recorded at these stations and that from the 100-N Area site and Hanford tower, 159 streamline analyses were made. The dates that were analyzed were randomly chosen from a 21-month period--January 1968 to October 1969.

III. PROGRESS DURING REPORTING PERIOD

A report on the analysis of data from tetron releases and meteorological stations was issued. This report summarized tetron releases, streamline studies, and wind direction correlations for the various stations. A major conclusion from the research to date is that the wind regime around 100-N Area is of a complex nature, such that a release of material at 100-N Area would pass over the Wahluke Slope 43 percent of the time. During certain atmospheric conditions and wind circulations, a vortex is formed near 100-D Area.

IV. EVALUATION OF EFFORT

The report by Battelle-Northwest indicates much more study will have to be made to resolve peculiar wind streamline phenomena, especially that of the vortex, which usually occurs around 100-D Area at low wind speeds.

V. FUTURE PLANS

Continue collection of data from all stations. Complete N meteorological system software and generate statistical summaries of wind speed-direction frequencies at 100-N Area. Revise field station network of wind sensors to optimize knowledge of airflow patterns. Carry out balloon releases (piballs and tetrons) and aircraft soundings of temperature variation with height. Implement diffusion studies using smoke and ZnS tracers. Determine characteristics and causes of the "vortex" phenomenon, especially the height and frequency of the effect.

VI. DOCUMENTS ISSUED

Letter, MA Wolf to DW Peacock, "Climate Summary - Wahluke Slope," February 11, 1970.

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SOIL STUDIES

I. SCOPE AND OBJECTIVES

The objective of this program is to evaluate the adequacy of ground disposal of contaminated liquid wastes generated after an N Reactor accident. This program will characterize the depth and flow of ground water, as a function of river level and season, in the vicinity of N Reactor.

The capacity of the soil for permanent retention of radionuclides will be determined from laboratory evaluation of core samples taken from soil in the vicinity of N Reactor and from tracer studies in the field which will disclose the nature and rate of radionuclide transport in the soil and ground water at selected sites.

A detailed simulation model will be obtained of the ground water flow system in the vicinity of N Reactor in order to determine the optimum location of the proposed emergency waste disposal basin. Various input flow rates and ground water environments will be considered.

II. MAJOR ACCOMPLISHMENTS TO DATE

The minimum travel time for I-131 from the 1301-N crib to the river has been shown to be:

1. Approximately 5 days from qualitative correlation of plots of I-131 concentrations versus time for the 1301-N crib and springs.
2. On the order of 12 to 13 days from the application of a digital simulation model.
3. On the average of 9 ± 1 days from monitoring ground water concentrations of I-131 following a fuel element rupture.
4. Ten to 12 days from cross-correlation study of crib and spring I-131 concentrations.

The I-131 concentration in the spring effluent going into the river was found normally to be roughly a factor of 3 above the 10CFR20 limits for uncontrolled areas. Thus, it was decided to increase the size of the rupture effluent waste disposal facility (now known as the Contaminated Coolant Treatment Facility - CCTF) in order that normal waste streams may be decontaminated to well within the 10CFR20 criteria.

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III. PROGRESS DURING REPORT PERIOD

The minimum travel time from the 1301-N crib to the river bank springs was found to be 9 ± 1 days from field measurements of I-131 concentrations after a fuel failure at N Reactor.

IV. EVALUATION OF EFFORT

The question of the I-131 travel times from the 1301-N crib to river is considered to be resolved.

V. FUTURE PLANS

The second phase of the digital modeling study to evaluate the hydrological factors related to the emergency waste disposal (EWD) basin will be implemented as soon as funds are available. This study will involve the effects on the ground water of a transient discharge of liquid into the EWD basin. The effect of activation or inactivation of the 1301-N crib on the model will be evaluated. The ground infiltration rate and decontamination factors of contaminated water entering the EWD basin will be determined from soil samples.

VI. DOCUMENTS ISSUED

BNWL-CC-2326, "Analysis of Travel Time of I-131 from the 1301-N Crib to the Columbia River during July 1969," WS Crews, DD Tillson, December 15, 1969.

LATTICE HEATING TRANSIENTS AND
CONFINEMENT SYSTEM ANALYSIS

I. SCOPE AND OBJECTIVES

The program involves experimental and analytical evaluation of heat transfer and coolant flow as a function of time and location in the reactor after initiation of an accident. Results from this program should allow accurate calculation of fuel temperature transients following loss of adequate cooling in the reactor.

II. MAJOR ACCOMPLISHMENTS TO DATE

The experimental apparatus for the fuel heat transfer experiments was completed.

III. PROGRESS DURING REPORT PERIOD

Computerized models were developed for primary cooling system blow-down calculations and confinement system pressurization studies. The blowdown program consists of main programs and a group of library

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subroutines representing specific configurations of primary piping, steam generators, pumps, and other major primary loop components and subsystems. The subroutines are arranged to be used in a "building block" fashion to analyze blowdown transients for primary pipe ruptures. This will provide the capability of analyzing steam release and reactor core transients for a pipe rupture in any chosen location and of any given size. Since each rupture case requires a different model (or system), one or more main programs will be required to tie the building blocks together in the appropriate arrangement. The confinement system model will be used to calculate confiner pressure transients that correspond to the specific blowdown transients analyzed, incorporating steam release transients, fog spray condensation, and confinement system operational logic.

IV. EVALUATION OF EFFORT

The experimental program has been rewarding in supporting analytical techniques and the combination of efforts will allow timely inputs to the overall regulatory effort.

V. FUTURE PLANS

Blowdown and confinement pressurization studies will continue with sufficient pipe break accident cases analyzed to determine the consequences of key pipe breaks at specific locations and of specific sizes. For each case studied, the time at which filtered release can be established, and the subsequent flow of steam through the filters will be combined with established rates, concentration, and species of fission products released to the confinement system to provide data for calculations of offsite radiation dosage consequences. Results from these studies are expected to identify any needs for modifications to the confinement system or to the operational logic of the system.

Heat transfer experiments simulating loss-of-coolant accidents will be evaluated and, if necessary, the results will be applied for adjustment of the fuel temperature transient model to refine calculations of the rate and fraction of core meltdown for various defined accident cases.

VI. DOCUMENTS ISSUED

None.

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K REACTOR METAL-METAL STUDIES

I. SCOPE AND OBJECTIVES

The uranium-aluminum reaction during a fuel melt accident could have a significant effect on the release of fission products and the extent of reactor damage. The purpose of this program is to gain information on the reaction kinetics, and apply these results to reactor accident studies.

II. MAJOR ACCOMPLISHMENTS TO DATE

The results of experiments performed at Iowa State University by Dr. Chiotti on the heat formation of uranium-aluminum compounds ($UAl_{2,3,4}$) have been utilized in analyzing the kinetics of the reaction. From analyses performed to date, it appears that the reactions are slow in relation to the timing of temperature excursions which would result from partial loss of cooling events. If this could be verified by further study, the exothermic metal-metal reaction would not be a significant consideration in such accidents.

III. PROGRESS DURING REPORTING PERIOD

There was no progress on this program during the reporting period.

IV. FUTURE PLANS

There are no plans to continue the work in the future.

V. DOCUMENTS ISSUED

None.

K SEISMIC STUDIES

I. SCOPE AND OBJECTIVES

An evaluation of the seismic adequacy of critical K reactor safety systems was started in FY-1970 by John Blume and associates. The purpose of these studies was to evaluate K reactor systems against the 0.12/0.25 g earthquake criterion currently applied at N Reactor. The results of these studies were to be used in formulating plant modifications required as part of a K Reactor Regulatory Program.

II. MAJOR ACCOMPLISHMENTS TO DATE

Major accomplishments to date include a preliminary report on part of the primary piping system (reported in Douglas United Nuclear, Inc.

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Semiannual Research and Development Report, December 31, 1969). A large portion of the analysis on the stack and shields, piping and 105 Building has been completed.

III. PROGRESS DURING REPORTING PERIOD

Work on this project was terminated in February 1970 due to budgetary limitations. As a result, no significant progress was made during this reporting period.

IV. EVALUATION OF EFFORT

The cancellation of this contract occurred at a time when the bulk of the data acquisition and analysis had been accomplished. However, no results had been transmitted to DUN as of this time.

V. FUTURE PLANS

No future work on this project is planned.

VI. DOCUMENTS ISSUED

None.

NOBLE GAS PARTITION STUDIES

I. SCOPE AND OBJECTIVES

The objective of this program is to determine the partition coefficients associated with noble gases in water to allow design of disposal basins in which accident produced radioactive wastes will be discharged. These studies will also give insight into the release path of the noble gases from the reactor confines.

II. MAJOR ACCOMPLISHMENTS TO DATE

Information has been obtained on the dissolution of krypton in water in a closed vessel, and the release of krypton from water to air after dumping the water to an open vessel. Data on the transport of krypton through water pools having varying degrees of turbulence have been gathered.

IV. EVALUATION OF EFFORT

Information to date has been useful for the definition of further tests that will allow a more meaningful interpretation of the studies with respect to the final disposition of accident produced liquid wastes.

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V. FUTURE PLANS

Investigation of the possibility of variation in krypton transport with pool depth will be made, as well as a study of the partition coefficients of radio-krypton when introduced to water with a carrier gas. The possibility of a large-scale partitioning test using existing tanks and piping at the N Reactor is being investigated.

VI. DOCUMENTS ISSUED

BNWL-CC-2395, "The Release of Kr-85 from Water - Interim Report," J Mishima, LC Schwendiman, February 2, 1970.

CONFINEMENT CHARCOAL FILTERS

I. SCOPE AND OBJECTIVES

Objectives of this program have been to develop a routine inspection system for the charcoal filters installed in the confinement systems of the operating reactors. Experience at Hanford and elsewhere has shown that it is not enough to test new charcoal filters individually before installation and then assume the installation to be sound.

The program for FY-1970 involved: (1) purchase as per specification of new stainless steel charcoal filter units of 1-inch bed depth, the media to consist of activated coconut shell charcoal impregnated for high ignition temperature, (2) Freon testing by vendor and DUN of the individual charcoal filter units, (3) installation of units in all operating reactor exhaust systems, (4) installation of permanent in-place test lines for annual in-place testing, (5) development and incorporation of a sample testing program quarterly to check the charcoal of the systems for premature poisoning and iodine capacity, and (6) incorporation and documentation of procedures in the Process Standards and Equipment Maintenance Standards.

II. MAJOR ACCOMPLISHMENTS TO DATE

The major accomplishments to date have been the receiving of the filter units, complete Freon testing of all units with no rejection, and installation of the charcoal filter units at KE Reactor, A and B cells and the C and B cells at N Reactor. A-cell, N Reactor, was reloaded with new coconut activated charcoal in old canisters in January 1970 because of borderline efficiency and the unavailability of new stainless filter units. D-cell at N Reactor, the standby cell, has not indicated sufficient canister corrosion or charcoal poisoning to warrant replacement.

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Permanent in-place test facilities have been installed in all operating reactor cells, and prototype removal partitioned charcoal filter units have been installed in A-cell, KE Reactor, and B-cell, N Reactor, to facilitate easy quarterly sample removal.

III. PROGRESS DURING REPORT PERIOD

Progress has been excellent with all projected plans of action for the current fiscal year on schedule:

- A. All cells are operating at a level of 99.95+ for iodine removal capability.
- B. All necessary replacement has been accomplished without incident.
- C. Permanent in-place test lines that save at least 1 man-week of time per filter cell when used for testing have been installed at all operating reactors.
- D. Incorporation of test procedures in the KE Reactor Process Standards has been accepted.
- E. Installation in two cells, one at KE Reactor and one at N Reactor, of a prototype sampling charcoal canister, has been accomplished. Where previously a sample cost \$1000 to retrieve, now only 2 hours of technician time is necessary.

IV. EVALUATION OF EFFORT

For the DUN-operated reactors with moist underground systems, the tests developed are the best that current technology will economically allow. Laboratory testing of charcoals at high temperature with steam to check poisoning effects, together with the extrapolation of results to system integrity in case of an accident condition, are not reported in open literature by any other installation.

V. FUTURE PLANS

This concludes the development work on this item. Future work will be accomplished utilizing operating funds.

VI. DOCUMENTS ISSUED

DUN-SA-105, "In-Place Testing of the Hanford Reactor Charcoal Confinement Filter Systems Using Iodine Tagged with Iodine-131 Tracer," JE Mecca, JD Ludwick, June 1969.

DUN-SA-116, "Activated Charcoal Filter History of the Hanford Single-Pass Production Reactors from March 1968 to the Present Time," JE Mecca, October 1969.

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BNWL-1046, "Iodine Collection Efficiency of Activated Charcoal from Hanford Reactor Confinement Systems: Methyl Iodide Retention by Activated Charcoal," JD Ludwick, July 1969.

HWS-8730, "Douglas United Nuclear, Inc. Specifications for Confinement System Charcoal Filters," EL Etheridge, JE Mecca, March 1969.

DUN-DC-1030, "Permanent In-Place Charcoal Filter Test Facilities," EL Etheridge, September 1969.

DUN-5870, "In-Place Test Procedure for Confinement Charcoal Filters," DL Hirschel, September 1969.

DUN-SA-128, "In-Place Test Results of the N Reactor Charcoal Confinement Filters Using Iodine Tagged with Iodine-131 Tracer," JE Mecca, JD Ludwick, September 1970 (to be issued)

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WASTE MANAGEMENT PROGRAM

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I. SCOPE AND OBJECTIVES

The objective of this program is to comply with executive orders regarding the reduction of pollution in the environment, including study, evaluation, and implementation of techniques that would allow compliance with governmental requirements and guidelines applicable to activities associated with waste disposal.

II. MAJOR ACCOMPLISHMENTS TO DATE

This is a new program established to reflect the increasing interest in waste disposal practices and the resultant environmental effect. Since it is not a previous continuing program, there is no previous record of major accomplishments to report.

III. PROGRESS DURING REPORT PERIOD

Pilot plant facilities to obtain information for the design of equipment to reduce the release of radioactive material to the environment were scoped for operating reactors.

IV. FUTURE PLANS

Two major areas of endeavor are envisioned for this program in the future:

1. Preliminary studies for a Contaminated Coolant Treatment Facility for N Reactor are planned. These studies will include pilot plant tests of an ion exchange processing system and facility processes. Information gained from these studies will be instrumental in allowing more detailed engineering studies leading to the design of the final configuration of the facility.
2. Monitor radioactive releases from the operating reactors to assure compliance with established appropriate regulations and guidelines.

V. EVALUATION OF EFFORT

Due to the limited time that specific program activities have been conducted, a meaningful evaluation of effort is not feasible. However, in the reporting period, general program definitions have been prepared, objectives established, and work started at a low level on some program facets. To date, progress on this program is considered to be adequate to meet overall needs. As more definitive policies regarding overall waste disposal practices are established by the Atomic Energy Commission, the program will be adjusted to meet these policies as needed within the constraints of available manpower and funding.

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VI. DOCUMENTS ISSUED

Letter, RG Geier to CW Kuhlman, "Waste Management Plans for DUN,"
October 13, 1969.

USAEC Construction Project Data Sheet, "N Reactor Confinement and
Contaminated Coolant Treatment Facilities - Richland, Washington,"
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