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**RICHLAND FIVE-YEAR
02 R&D PROGRAM**

HIGHLY ENRICHED FUEL PROGRAM

**RICHLAND OPERATIONS OFFICE
DOUGLAS UNITED NUCLEAR, INC.
GENERAL ELECTRIC COMPANY - HAPD
ISOCHEM INC.
BATTELLE - NORTHWEST**

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HIGHLY ENRICHED FUEL PROGRAM

SUMMARY

INTRODUCTION

As the military needs for plutonium decline, the Commission's production-oriented reactor complex should develop and demonstrate pure non-defense operating modes to serve non-defense needs. The alternative is production reactor shutdowns coupled with needless expense in producing necessary products elsewhere or doing without them. Savannah River already has secured such pure non-defense capability, at least from a technical feasibility standpoint, in their fully-enriched, isotope-producing loadings and in their Pu-burning, partial loadings. Although the use of fully enriched fuel (partial or complete loadings) for the Richland production reactors was once practiced, renewed study and research to reestablish practical incentives, demonstrate technical feasibility (including reactor safety), and firm up product capability must be undertaken.

It is unquestionably true that fully-enriched uranium can be utilized as fuel for the Hanford reactors. It is the purpose of this mission to develop the conditions under which this may be accomplished safely and productively, to determine the necessary plant modifications required to achieve this processing capability, to disclose the inherent plant limitations to such processing, and to evaluate the incentives for operating on fully enriched uranium fuel.

INCENTIVES

The purpose of most of the Hanford R&D programs is to develop methods of diverting the productive capacity of the Hanford plant to the production of non-weapons products; i.e., products other than weapons-grade plutonium and tritium. To accomplish this, the reactors must be fueled so as to produce an excess of neutrons, beyond those required to sustain the fission process, which are not absorbed in U-238 or in Li-6. This is most readily and straightforwardly accomplished by eliminating these absorbers from the reactor core, and substituting target atoms which will produce more desirable products upon neutron absorption. For complete diversion of production to non-defense products, fully-enriched fuels, either U-235 or plutonium, must be utilized.

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A variety of neutron-irradiation-produced isotopes have been found to be of interest to those engaged in developing thermal-electric power sources. These isotopes range from the fission products such as strontium and cesium, to the transplutonium isotopes such as curium. They include several which are produced by simple neutron capture in target atoms, such as cobalt-60, plutonium-238, thallium-204, polonium-210, and thulium-170.

The degree of interest in these isotopes varies, depending on the potential application being studied, and perhaps the state-of-the-art in a particular applications area. Studies of production methods also vary; in somewhat direct proportion to the interest evidenced, these studies range from simple estimates of production capability to scoping studies of possible irradiation techniques to full-scale design studies and test irradiations. It appears inevitable, and not undesirable, that several or all of these isotopes will ultimately find their way into production programs, since the variety of power source missions for which they are required will result in finding that each isotope is optimum for certain special applications.

If one makes the (hopefully correct) assumption that the production complex will be faced with an unstable and flexible product demand, it is only logical that a single, or a few, modes of reactor fueling be developed which will suffice to produce any product mix in a near-optimum manner. It does not appear logical, and would impose an unrealistic demand on fuels fabrication and processing facilities, to attempt to accomplish more than this.

Setting up tentative criteria:

- a) Elimination of weapons-grade plutonium production,
- b) Irradiation of a number of different target isotopes,
- c) Possible production of fast reactor feed grade plutonium plus transplutonium isotopes, and
- d) Production of electric power (in the case of N-Reactor).

One concludes that fuel systems of either U-235 or some plutonium isotopic mix should be investigated. Should these then prove infeasible, the maximum feasible concentration of U-235 in uranium, and of plutonium mixed with a resonance absorber, should be sought. Therefore, trusting that suitable target isotopes will be provided when required, these studies will be undertaken.

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SCOPE AND OBJECTIVES

1. Reactors

A basic assumption is made that other programs will develop the need for, and the technology of producing, the target materials. This mission is designed to prevent "reinventing the wheel" for each target proposal. However, one must recognize that the targets must be put somewhere - irradiation space must be provided, which leads inevitably to the question of how much. Since the only reasonable premise is; "as much as possible," the program must begin on this note. The goal of developing "high power density" fuel is another mission - unfortunately, it is not practical to attack the problem serially within the time allotted (assumed to be four years). An arbitrary engineering judgment (commonly known as a guess) is therefore made that one-third of the irradiation volume can be devoted to target space, while operating the remaining two-thirds at a power density which will deliver the reactor design power level. (This is not necessarily rational - there must be some optimum combination of target-fuel ratio and power level.) Variations in irradiation volume and reactor power level are therefore excluded from the scope of this study. The scope of the study will be confined to:

- a) Physics behavior
 - 1) "Prompt" effects
 - 2) Total reactivity effects
 - 3) Transient reactivity effects
- b) Fuel materials (shared with High Power Density Mission)
- c) Reactor effects
 - 1) Moderator temperature and life considerations
 - 2) Zirconium tube temperature and life considerations
- d) Fuel handling problems - critical mass considerations (shared with Fuels Preparation and Reprocessing Sections).

1.1 Physics Behavior

1.1.1 Reactor Control - "Prompt" Effects

Once the U-238 resonance absorber is eliminated from the fuel composition, the benefit of the doppler effect as an aid in controlling reactor excursions is lost (except for the effect of the Pu-240

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resonance in plutonium fuels, which will be explored). However, the effect of delayed neutrons is (never fear) still available. If one recognizes that the fuel heating time constant is not zero in present fuels, it is conceivable that the lost doppler could be replaced by the rapid scram action of control elements, yielding an adequately controlled system. It seems likely that some combination of: (a) engineered safeguards to inhibit large, rapid additions of reactivity, (b) instrumentation for early detection of an incipient excursion, and (c) a reasonably rapid scram system, will solve this problem.

1.1.2 Reactor Control - Total Reactivity Effects

Highly enriched fuel loads will be characterized by large reactivity changes which are generated as the fuel burns, by (possibly) large startup transient reactivity effects, and (possibly) by adverse positive reactivity changes if coolant is lost. The burnup reactivity changes can be made manageable by: (a) Possessing or installing a very large capacity control system - hopefully being imaginative enough to fabricate it so as to produce useful products when irradiated, (b) Charging the fuel in timed increments so that only a small part of the burnout transient is suffered between charging. This, of course, must be adjusted to be within the available control capacity. Differences in channel power inhibit this activity. (c) Periodically removing part of the target poisons as burnout progresses. One can imagine either changing the density of a particular target isotope, or periodically changing to a new target isotope, depending on product demand. (d) Utilizing "burnable" target isotopes (which, of course, they are anyway). This confines the target candidates to reasonably high cross section materials, probably diluted with inert materials, so that just the right fraction of targets will burnout to counterbalance the fuel burnup. Tricky but possible. (e) A combination of all of these - to some extent inevitable. One must guard against wandering around in permutations forever, and make astute choices based on survey studies.

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1.1.3 Reactor Control - Transient Effects

Transient reactivity effects and their control are difficult to define, since they tend to be very sensitive to fuel and target compositions and orientation, and to fuel exposures. Calculations have disclosed that, when target isotopes are lumped so as to be thermally black, and the target pieces are placed in channels separate from the drivers, the temperature transients are large and negative. This is also true if the target is placed concentrically inside the fuel element. However, if the target is placed immediately outside the fuel, it can be made to effectively isolate the fuel from the spectral effects of moderator temperature changes, and positive reactivity transients will result. Thus, at least one mechanism is available to tailor the reactivity effects to controllable values. Supplemental reactivity control mechanisms are already installed at the older Hanford reactors, and are under development for the N-Reactor. These mechanisms can be arranged to compensate for large swings - up to 5-7 percent k - in reactivity.

The reactivity change resulting from reactor coolant loss can be positive or negative, depending on the particular reactor load configuration. A positive reactivity change is perfectly acceptable, provided that the response of the control system is fast enough to overcome the positive reactivity ramp, and that the worth of the control system is greater than the dry excess reactivity. The latter requirement may restrict the loading designs to those with higher fissile material concentrations, and consequently to "blacker" target systems. Calculations indicate that the worth of control elements and targets does increase markedly with coolant loss due to the increased neutron migration area in the absence of coolant. By "lumping" the target into well-spaced "control elements" this effect may be enhanced, and should provide adequate control strength. Designs which depend on target control worth under coolant loss conditions also restrict the target design to materials which will remain intact under these high temperature conditions.

1.2 Fuel Materials

Many of the problems which must be considered under this subject may be assigned to the High Power Density Program. However, some problems would not arise at all but for the attempt to irradiate fully enriched fuels, and are sponsored by this program.

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Oralloy fuels have traditionally consisted of alloys of uranium and aluminum which are readily fabricated and reprocessed. However, it is not at all certain that these alloys will be compatible with modernized fission product confinement philosophies; certainly, studies of fission product retention in melting metallic uranium would not apply.

Certain fundamental criteria will govern the choice of fuel materials; (a) The actual fissile material must be diluted many-fold by inert substances, else fission density and accompanying exposure-dependent effects could not be tolerated. Dilutions by a factor of 20-100 are expected to be required. (b) The diluent must have a low neutron capture cross section. (c) The resulting fuel alloy must not pass through phase changes involving volume differences between shutdown and operating points. (d) The fuel alloy must resist swelling with exposure; such changes will upset the thermal-hydraulic balance of multi-coolant-channel elements and further, may endanger the integrity of the fuel cladding. (e) Upon coolant loss, the fuel assembly must retain fission products for a time, or to an extent, which satisfies criteria relative to the release of fission products to the environs. (f) The fuel must lend itself to mass-production fabrication and canning, casting and canning, etc. Fabrication processes which inherently generate large amounts of fine fuel particles are less desirable, because of critical mass control problems. (g) The fuel must be reprocessable in some existing plant, at a rate compatible with large-scale reactor irradiations. Plant modifications to insure critical mass control are considered inevitable, and separate from the fuels material problems. However, it is to be hoped that the choice of fuel materials can be guided by existing chemical technology in the dissolving and purifications steps.

1.3 Reactor Effects

1.3.1 Moderator Effects

A small fraction of the energy of each fission is released as gamma radiation. This energy is then deposited in each position of the lattice according to the gamma absorption cross section of each material, and its density. It follows that the removal of the uranium-238 from the fuel greatly reduces the mass; the gamma energy deposition will be redistributed, and a large share will fall to the graphite moderator. Since the heat generated must then be conducted back to the coolant heat sinks, it follows that the moderator temperature will rise until a new equilibrium heat balance is

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established. The temperature of the moderator may or may not be compatible with minimizing graphite growth or contraction phenomena, or with graphite burnout limits in cases where oxygen or oxides of carbon are present as part of the reactor atmosphere.

About 30 percent of the gamma energy generated in standard metallic uranium fuels is "prompt" gamma released upon neutron capture in uranium-238. These neutrons will now be captured in some substitute target material. In some cases, the prompt energy release will not be in the form of gamma radiation, but will appear as recoil energy in alpha particles. In these cases, the recoil energy is confined to the directly cooled target, and the fraction of the fission energy deposited in the graphite moderator is reduced.

1.3.2 Zirconium Tube Effects (N-Reactor)

If the net heat transfer from the graphite moderator to the coolant stream is increased, the temperature rise across the zirconium process tube will increase proportionately. A significant increase (more than, say, 10 percent) may cause the outer surface of the tube, at some points, to reach temperatures incompatible with the prevention of gas phase hydriding with the existing reactor atmosphere. Research into inhibiting atmospheres will probably be an essential part of this program if a "universal" fuel system, compatible with all possible targets, is required.

Both of the above problem areas are "local" effects, related to local power density conditions, not gross reactor power. The magnitude of the problem may therefore be reduced by simply reducing the ratio of maximum to average power density. This has been accomplished for years in the radial reactor dimensions by selective fuel loadings. Flattening in the axial dimension has not been significantly explored, and may prove to be the simplest solution. The degree of axial flattening possible tends to be inhibited by the tendency of these large reactors toward flux cycling due to cyclic variations in xenon poison concentration. Such tendencies are also sharply dependent on the temperature coefficients of reactivity, and application of flux flattening could probably be accomplished only in those reactor loads which exhibit net negative temperature coefficients of reactivity.

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1.4 Critical Mass and Fuel Handling Problems

Critical mass considerations will generate handling problems throughout the fuel handling complex. That these problems can be contained has, however, been demonstrated at other sites which process fully enriched fuels.

A significant alleviating feature in the processing of fully enriched fuels is the reduced volumetric throughput rate. With partially enriched uranium fuels, severely low critical masses are accompanied by high mass and volume throughputs, since only modest fuel exposures are sought. Fully enriched fuels will ordinarily be aimed to achieve a high percentage burnout of fissile material, which will greatly reduce throughput. For example, using the crude rule of thumb that the fissile material input concentration will remain the same as in metallic fuel, but that 50 percent burnout will be achieved, the residence time of the fuel will be increased by a factor of about three. Further the mass of fuel handled will be reduced by roughly an order of magnitude due to the removal of the uranium-238. While these changes will not eliminate the fuel handling problems, they will become more manageable.

It is conceivable that special fuel-target designs can be utilized to reduce the critical mass management problem throughout much of the fuel cycle. By building an integral driver-target fuel, the critical values can be reduced to those of common, metallic uranium fuel until it is necessary to separate the components. This could probably be accomplished just before driver reprocessing, eliminating critical mass problems inherent in the wet storage of irradiated fuel. Many target systems will not be amenable to this treatment, which is, therefore, not a cure-all. It is undoubtedly true that considerable equipment design will be required to provide geometrically favorable arrays and handling equipment.

2. Reprocessing Plant

The scope and objectives of studies of the reprocessing plant are by-and-large implicit in those outlined for the reactors. However, in order to prevent a serialization of the entire mission, or "tailgating" the reactor studies by the reprocessing studies, it is necessary that the latter be approached first from the standpoint of inherent plant and process limitations, rather than "go - no go" studies of reactor fueling proposals.

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Two major considerations guide the study of reprocessing plant capabilities:
 (a) Chemical compatibility of possible fuels with the plant flow sheet, and
 (b) Criticality prevention at all points in the stream.

Basic technology for reprocessing a limited number of types of fuel is available; these include metallic uranium clad with aluminum or zirconium, uranium oxide, thorium, mixed oxides, etc.

It is quite possible that fuel models developed to satisfy reactor requirements will fit within one of these reprocessing schemes. Initial reactor fuel investigations will be guided by the available reprocessing technology so that, if possible, an extensive research program in this area can be avoided.

PROGRESS DURING REPORT PERIOD

Highly enriched fuel processing has only recently been designated a separate research and development mission. Work specific to this mission is scheduled to begin in FY-1967; thus, no progress toward the goals and objectives outlined has been made during FY-1966.

BUDGET PERIOD PLANS AND EXPECTED RESULTS

1. FY-1967

1.1 Reactors

Work during FY-1967 will be first directed toward establishing the basic feasibility of reactor operation with fully enriched fuel, and identifying specific limitations imposed by the inherent reactor design characteristics. As such, this work will be initially confined to physics studies. These studies will follow the basic assumptions outlined under Scope and Objectives, designing a series of reactor fuels, varying (a) the fissile material concentration, while compensating the reactivity changes with added target concentration, (b) the fueling geometry, from integral elements containing both fuel and target to "supercell" arrays where fuel and targets are placed in separate channels, and (c) fuel burnout. Calculations will be performed to determine: (a) Reactivity changes with fuel exposure, and accompanying fuel channel relative power changes, if "continuous" charging is assumed. (b) Reactivity

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changes with reactor temperature. (c) Reactivity changes upon loss of coolant. (d) The worth of the available reactor control systems, their ability to control the reactor under all normal and accident conditions, or the amount of additional control capacity required. (e) The kinetic response of the reactors to the maximum credible positive reactivity ramps, assuming that available control systems are scrambled, and any required engineered safeguards to limit the maximum credible ramp. Assuming that the initial physics studies reveal a feasible method of utilizing fully enriched fuels, studies in the latter part of FY-1967 will be directed toward defining a fuel material which will satisfy reactor criteria with regard to irradiation behavior, fission product confinement, and fabricability. As this work progresses, coordination with the reprocessing studies will permit the adaptation of existing technology to the proposed fuels, or development of new technology.

1.2 Reprocessing

Work during FY-1967 will be first directed toward determining the inherent limits of the existing separations plant in the processing of fully enriched fuel. Fuel models will not yet have been defined by reactor studies; initial reprocessing studies will therefore be largely confined to: (a) assessing the criticality restrictions, based on a series of assumed fissile material throughput assumptions, (b) making preliminary studies of existing flow sheets, thus providing assistance to reactor studies of possible fuel materials choices, (c) determining plant capabilities in those positions where such capacities will be relatively independent of the fuel models finally chosen, i.e., reduction of separated products to final form, separation of products after dissolution, and (d) determining the extent by which the plant capacity may be enhanced through modifications to provide critically safe equipment, improved instrumentation systems, and other engineered safeguards.

2. FY-1968

2.1 Reactors

Reactor fueling systems which will develop from the FY-1967 studies will probably be quite dissimilar from any previous reactor loadings. While the development of sophisticated analytical models has enhanced confidence in the accuracy of calculated reactivity parameters, it

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will be necessary that some experimental verification be obtained. One or two loading schemes will be chosen which will provide "landmark" tests of the analytical techniques. Experiments will be performed to determine the lattice parameters of the loading k_{00} , thermal utilization, and flux distribution, and the worth of the reactor control systems in these loadings. In parallel with these tests, in-reactor irradiations will be performed to verify fuel performance characteristics, fuel burnout predictions, and lattice productivity. Planning and design of these tests, obtaining materials and fabricating fuel and targets, will consume most of FY-1968; testing will be initiated late in FY-1968.

Design of the prototypic tests should be completed by March 1968. It is assumed that a full-reactor irradiation mission, identifying a specific product and quantity, will have been defined. Based on these data, design of a full reactor irradiation campaign will be initiated. This design will be completed by mid FY-1969, and a Hazards Summary Report will be prepared by about March 1969. The results of the prototypical irradiations, which will be started in about June 1968, will become available about June 1969, thus providing the experimental verification required for full-scale loading approval.

2.2 Reprocessing

A spectrum of reactor fuel candidates will have been obtained during FY-1967. Separations process details will be investigated in depth during FY-1968 for these fuels. As flow sheets are developed, equipment and materials requirements to provide the required plant capacity will be defined. Pilot plant investigations to demonstrate flow sheet flexibility will be performed. By the end of FY-1968 sufficient detail will be developed to permit budget studies for facility modifications and additions.

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Statistical Summary Schedule

	<u>Fiscal Year</u>				
	<u>1967</u>	<u>1968</u>	<u>1969</u>	<u>1970</u>	<u>1971</u>
<u>Dollars (in Thousands)</u>					
General Electric (N-Reactor)	35	35	100	175	250
Douglas United Nuclear	30	100	140	170	250
Isochem	<u>158</u>	<u>191</u>	<u>200</u>	<u>250</u>	<u>300</u>
Total	223	326	440	595	800

Man Years

General Electric (N-Reactor)	.5	.5
Douglas United Nuclear	0	1.3
Isochem	<u>1.6</u>	<u>2.2</u>
Total	2.1	4.0

Equipment

General Electric (N-Reactor)	0	0
Douglas United Nuclear	5	10
Isochem	<u>0</u>	<u>0</u>
Total	5	10

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U. S. ATOMIC ENERGY COMMISSION

RICHLAND
OPERATIONS OFFICERESEARCH AND DEVELOPMENT STATISTICAL SUMMARY
02 PROGRAM

HAN- _____

1. MISSION NO. AND TITLE <u>No. 17, Highly Enriched Fuel Program</u> MISSION SUB-PROGRAM TITLE _____		2. BUDGET ACTIVITY NO. _____
3. CONTRACTOR AND CONTRACTOR NUMBER	A. <u>General Electric Company</u> B. _____ C. _____ <u>AT(45-1)1350</u>	4. PERSON IN CHARGE: <u>M. C. Leverett</u> INVESTIGATOR: <u>W. S. Nechodom,</u> INVESTIGATOR: <u>M. Lewis</u>

FIVE YEAR FORECAST

	CURRENT BUDGET YEARS *					5 YEAR TOTAL
	FY 19 67	FY 19 68	FY 19 69	FY 19 70	FY 19 71	
5. MAN-YEARS						
A) SCIENTIFIC AND TECHNICAL	.4	.4				
B) OTHER	.1	.1				
TOTAL	.5	.5				
6. COSTS (\$000'S)	\$	\$	\$	\$	\$	\$
A) DIRECT LABOR (INCL. COST OF SERV.)	16	15				
B) MATERIALS, TECH. SERV., SUB CONTRACTS	14	13	37	65	93	
C) SPONSORED WITH BNW						
TOTAL DIRECT	30	28				
D) INDIRECT EXPENSES	5	7				
TOTAL OPERATING COSTS	35	35	100	175	250	
E) EQUIPMENT (NEW)						
F) IRRADIATION UNIT COSTS						

7. COMMENTS:

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GENERAL ELECTRIC COMPANY

MISSION NO. 17, HIGHLY ENRICHED FUEL PROGRAM

SCOPE AND OBJECTIVES

The objective of this mission is to devise one or a few highly enriched fuel models for the N-Reactor which will serve as neutron sources for the irradiation of a broad spectrum of target materials. Studies will be made of the use of U-235 and plutonium isotopic mixes as fuel. Since another program (High Power Density) will undertake to define the fraction of the reactor irradiation space which must contain fuel, it will be initially assumed that one-third of the space must be reserved for target material, and that the remaining two-third must operate the reactor at full design power.

The scope of this study will include:

- a) Physics Behavior of highly enriched fuels
 - 1) "Prompt" kinetic effects
 - 2) Total reactivity effects and control system worths
 - 3) Transient reactivity effects
- b) Fuel Materials
- c) Reactor Effects
 - 1) Moderator temperature
 - 2) Zirconium tube temperature
- d) Fuel Handling - critical mass considerations.

1.1 Physics Behavior

1.1.1 "Prompt" Kinetic Effects

The N-Reactor Mark I fuel load possesses a prompt negative temperature coefficient of reactivity, made up of a doppler effect due to fuel heating, and a net reactivity defect due to coolant heating. The reactor power is therefore self-damping, and accidental excursions are limited by this effect. Fully enriched fuel loads would not possess a doppler defect. However, the negative coolant temperature coefficient of reactivity would be retained in most models. In addition, the major factor

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contributing to adequate reactor control is the existence of delayed neutrons, which minimize the reactor power response to small changes in reactivity. While the kinetic behavior of all proposed models must be evaluated, it is likely that the combination of negative coolant temperature coefficient of reactivity, delayed neutron fraction, sensitive scram instrumentation, and accelerated safety rods, will provide adequate reactor control.

1.1.2 Total Reactivity Effects and Control System Worth

The worth of the N-Reactor primary control system is limited to 6-8 percent k, depending on the fuel model and reactor condition. The secondary backup system contributes an additional 5-7 percent k. These systems must control the reactivity swings of startup transients, including xenon poison buildup, leaving little capacity for controlling fuel burnup reactivity transients. Studies of fuel loads which will be subject to large burnup transients will include evaluation of the need for supplementary control devices, incremental fuel charging to minimize the reactivity swing, arranging the target isotopes as burnable poisons, and intermittent target element replacement with lower cross section materials.

1.1.3 Transient Reactivity Effects

Reactor startup net reactivity transients and the reactivity transient associated with coolant loss can be either negative or positive, depending on the loading model. The transients can be adjusted to some extent by target design and placement. Since the control system must compensate for both transient and fuel burnup reactivity effects, these problems interact, and the total reactivity swing determines the required control capacity.

1.2 Fuel Materials

The design of the N-Reactor, utilizing the confinement concept of fission product release control, requires that the fuel element perform as a fission product container at least through the limited time period while the primary loop is undergoing depressurization following coolant supply failure. The current, conservative, hydraulic analysis of this accident indicates that fission products must be retained within the fuel for at least 220 seconds after coolant is lost. In the case of the Mark I fuel, this is accomplished automatically, since transient analyses indicate that

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the fuel does not exceed the uranium melting point until at least 220 seconds have elapsed. A similar resistance to gross fuel failures will have to be provided for other fuel models. This could again be accomplished by choosing a high melting point fuel matrix, or by insuring that the fuel cladding will remain intact for the required time.

1.3 Reactor Effects

1.3.1 Moderator Effects

The N-Reactor graphite temperature, when the reactor is loaded with Mark I fuel, ranges up to 1150 F. The maximum graphite temperature is limited primarily by considerations of graphite burnout rates in the presence of oxidizer agents, and the limit is therefore a function of the concentration of these agents. It is found to be necessary to maintain a certain minimum concentration of water vapor in the reactor atmosphere to prevent hydriding of the zirconium process tube; this concentration in turn establishes a graphite temperature limit. The current practical limiting case occurs with a water vapor concentration between 200 and 600 ppm, a zirconium tube temperature limit of 700 F, and a graphite temperature limit of about 1550 F.

The graphite temperature is a function of the heat generation rate in the moderator, and the conductivity of the heat transfer paths from the graphite to the reactor coolant streams. The conductivity is not upwardly adjustable; helium is used as a heat transfer medium, and possesses the best conductivity of available gaseous mediums. The graphite temperature could be raised, but not lowered, by modifying the gas composition. The heat generation rate in the graphite is a function of reactor (actually regional) power, and of the fuel composition. Heat is generated by moderation of neutrons and by capture of gamma radiation. Changes in fuel models can result in changing the ratio of water to graphite volume, thus changing the relative amount of neutron moderation, and can also result in changing the relative amount of gamma radiation absorbed in the fuel and moderator. In addition, since part of the gamma radiation results from radiative neutron capture in target materials, the total gamma energy generation may change as a result of changing fuel models. In general, it is found that lighter fuel weights result in higher graphite temperatures, while changing the target material from U-238 to an alpha-particle-producing material will reduce graphite temperatures. Each highly enriched fuel model will therefore have to be evaluated in combination with several target materials, to establish a range of heat generation rates as a function of power level.

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The most direct method of changing the maximum graphite temperature with any fuel model is to adjust the maximum local power level. This may be accomplished by selective placement of enriched, "spike," fuels, or by control rod manipulation. It has been found that the Mark I fuel load can readily be operated with a radial peak-to-average tube power ratio of about 1.25. However, it is found that attempts to reduce the axial ratio to this value result in xenon cycling instabilities. Thus, the axial ratio is maintained at about 1.4 or greater, resulting in an overall peak-to-average power ratio of greater than 1.75. The xenon cycling constraint will not necessarily apply to all loadings; those which exhibit strong negative temperature coefficients of reactivity will not cycle, and may be flattened as much as economically feasible. An overall ratio of about 1.55 or less may be possible. Designs of fully enriched fuel loads will therefore be biased in the direction of negative reactivity coefficients in order to establish this flexibility.

1.3.2 Zirconium Tube Temperature Effects

The N-Reactor zirconium process tube outer surface temperature must be controlled below a limit of about 700 F to prevent zirconium hydriding. The maximum tube surface temperature with Mark I fuel ranges up to 660 F, being established by the operating coolant temperature and the rate of heat transfer from the moderator to the coolant stream (and hence, the heat generation rate in the graphite). Since the tube temperature is a function of heat generation rate in the graphite, the discussion in the preceding section also applies to zirconium temperature control. Here, however, the current operating point is found to be closer to the technical limit. In addition to attempts to control within the existing limit, it may be necessary to undertake studies designed to increase the limit, either by a more accurate and precise experimental program, or by devising inhibiting reactor atmospheres which permit operation at higher temperatures.

1.4 Fuel Handling - Critical Mass Considerations

While little or no new research will be required, application of existing critical mass technology to the present plant process will require an extensive and thorough design and equipment development effort.

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Current N-Reactor criticality control criteria require that all fissile material arrays be assumed optimally moderated, for purposes of evaluating criticality potential. This is feasible with the relatively low fuel enrichments which have been utilized. However, experience indicates that fuel handling limits will become excessively restrictive at enrichments in excess of 2-3 w/o U-235. It will be necessary that credit be taken for the absence of moderator, and that the process be designed with this end in mind, in order to efficiently process fully enriched fuel. This is currently standard practice at other sites.

Critical mass problems will be partially alleviated when processing fully enriched fuel by the reduced volume of fuel required. Such fuel will be irradiated to much higher fissile material burnups than is the current practice with metallic uranium fuels. It can be expected that the fully enriched fuel volume throughput will be roughly a factor of three less than that of current fuel models. In addition, the mass of fissile fuel handled will be reduced by roughly an order of magnitude due to the removal of the U-238. Both of these factors will tend to make easier the solutions to critical mass handling problems.

The most difficult problem foreseen, from the standpoint of modifying existing systems (it is assumed a different fuel fabrication plant will be utilized) is the irradiated fuel handling and storage system. For reactor efficiency, fuel must be discharged in large batches. The existing fuel handling and storage arrangement is incompatible with efficient handling of fully enriched fuels. The system will have to be modified to handle the discharged fuel more rapidly, in small quantities or a continuous stream, and to package and store the fuel in safe arrays. A system design which is safe by design and geometry, not by procedure, is considered a necessity where a moderating medium is always present.

PROGRESS DURING REPORT PERIOD

Mission 17 is now being initiated, and has progressed only into the planning state.

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BUDGET PERIOD PLANS AND EXPECTED RESULTS

1. FY-1967

The basic physics feasibility of N-Reactor operation with fully enriched fuels will be established. Parametric calculations will be performed, varying: (a) fissile material concentration, (b) fuel element geometry, from integral elements containing both fuel and target to "supercell" arrays, and (c) fuel burnout. The calculations will determine: (1) reactivity and fuel relative power variations with fuel burnup, (2) reactivity changes with reactor temperature and power, (3) reactivity changes with coolant loss, (4) worth of the available control systems, and amount of required supplemental control, and (5) the kinetic response of the reactor to rapid reactivity insertions.

Assuming that feasible methods of fully enriched fueling are defined, studies in the latter part of FY-1967 will be directed toward defining fuel material(s) which will satisfy criteria for irradiation behavior, fission product confinement, and fabricability. Conversely, incentives may be revealed for engineering modifications to the N-Reactor confinement system to make it compatible with an otherwise desirable fuel model. Coordination with the reprocessing studies is an essential feature of this effort.

2. FY-1968

Reactor fueling systems which will develop from the FY-1967 studies will probably be quite dissimilar from any previous reactor loadings. While the development of sophisticated analytical models has enhanced confidence in the accuracy of calculated reactivity parameters, it will be necessary that some experimental verification be obtained. One or two loading schemes will be chosen which will provide "landmark" tests of the analytical techniques. Experiments will be performed to determine the lattice parameters of the loading - k_{00} , thermal utilization, and flux distribution, and the worth of the reactor control systems in these loadings. In parallel with these tests, in-reactor irradiations will be performed to verify fuel performance characteristics, fuel burnout predictions, and lattice productivity. Planning and design of these tests, obtaining materials and fabricating fuel and targets, will consume most of FY-1968; testing will be initiated late in FY-1968.

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Design of the prototypic tests should be completed by March 1968. It is assumed that a full-reactor irradiation mission, identifying a specific product and quantity, will have been defined during FY-1967. Based on these data, design of a full reactor irradiation campaign will be initiated. This design will be completed by mid FY-1969, and a Hazards Summary Report will be prepared by about March 1969. The results of the prototypical irradiations, which will be started in about March 1968, will become available about June 1969, thus providing the experimental verification required for full-scale loading approval.

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Schedule of Activities

Establish feasible reactor loading design.	January 1967
Define fuel material.	March 1967
Establish a full-reactor target irradiation mission.	June 1967
Initiate PCTR tests.	September 1967
Complete PCTR tests.	March 1968
Initiate prototype reactor irradiations.	March 1968
Complete full-reactor test design.	January 1969
Complete full-reactor hazards summary report.	March 1969
Complete and analyze prototype irradiations.	June 1969
Obtain full-scale loading approval.	July 1969

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U. S. ATOMIC ENERGY COMMISSION

RICHLAND
OPERATIONS OFFICE

RESEARCH AND DEVELOPMENT STATISTICAL SUMMARY
02 PROGRAM

HAN- _____

1. MISSION NO. AND TITLE <u>No. 17, Highly Enriched Fuel Program</u> MISSION SUB-PROGRAM TITLE _____		2. BUDGET ACTIVITY NO. _____	
3. CONTRACTOR AND CONTRACTOR NUMBER		4. PERSON IN CHARGE:	
A. <u>Douglas United Nuclear, Inc.</u> <u>AT(45-1)1857</u>		<u>T. W. Ambrose</u>	
B. _____		<u>R. W. Reid</u>	
C. _____		INVESTIGATOR: _____	

FIVE YEAR FORECAST

	CURRENT BUDGET YEARS *					5 YEAR TOTAL
	FY 19 67	FY 19 68	FY 19 69	FY 19 70	FY 19 71	
5. MAN-YEARS						
A) SCIENTIFIC AND TECHNICAL	0	1.3				
B) OTHER	0	0				
TOTAL	0	1.3				
6. COSTS (\$000'S)	\$	\$	\$	\$	\$	\$
A) DIRECT LABOR (INCL. COST OF SERV.)	0	26				
B) MATERIALS, TECH. SERV., SUB CONTRACTS		12				
C) SPONSORED WITH BNW	28	50	70	90	120	
TOTAL DIRECT	28	88				
D) INDIRECT EXPENSES	2	12				
TOTAL OPERATING COSTS	30	100	140	170	250	
E) EQUIPMENT (NEW)	5	10				
F) IRRADIATION UNIT COSTS						

7. COMMENTS:

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DOUGLAS UNITED NUCLEAR

MISSION NO. 17, HIGHLY ENRICHED FUEL PROGRAM

SCOPE AND OBJECTIVE

Prior to 1955 highly enriched fuel in the form of uranium-aluminum alloy was used as spike enrichment and as reactivity support for a tritium production load. The use of highly enriched fuel was discontinued because of unresolved questions concerning safety of the loadings and economic disadvantages.

The objective of this mission is to investigate the safety and economic feasibility of full and partial highly enriched fuel-target loadings in the five production reactors operated by Douglas United Nuclear, Inc. The availability of high speed computer codes, improved experimental methods and facilities, and proven calculational techniques will allow resolution of the questions on the safety of highly enriched fuel-target loadings.

The stimulus for such loadings is that effectively no Pu-239 is produced and all production can be diverted to other isotopes.

The predominate characteristic of the highly enriched fuels which requires safety analysis is the lack of U-238, which results in decreased negative doppler coefficient and the low melting temperatures of uranium-aluminum alloys.

PROGRESS DURING RECENT PERIOD

This mission has just been established, and hence, there is no progress to report.

BUDGET PERIOD PLANS AND EXPECTED RESULTS

Reactor engineering and physics analysis computer codes will be used to determine fuel design and operating characteristics, lattice physics parameters, temperature coefficients, water loss reactivity, and critical mass data. Reactor kinetic codes will be used to determine operating safety limits and melting times, especially as concerned with "speed-of-control"

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and "confinement" criteria. Modifications necessary in safety system, fuel fabrication facilities and handling, storage, and shipment facilities will be determined. Calculated yields and operating characteristics will be used to obtain preliminary cost estimates. From the calculated data a preliminary Hazards Summary Report can be prepared.

If the preliminary Hazards Summary Report shows feasibility of highly enriched fuel, experiments will be performed to confirm calculations and in-reactor test loads will be designed to obtain data on fuel element and target fabrication and separation, reactor performance, and yield data. These data will then be used to design full and partial reactor loadings.

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Schedule of Activities

Calculate reactor physics and engineering parameters - DUN and BNW.	January 1967
Determine modifications to existing facilities - DUN.	March 1967
Preliminary Hazards Summary Report - DUN.	June 1967
Physics and Engineering tests - BNW.	September 1967
In-reactor demonstration load design and irradiation - DUN.	March 1968
Separations and analysis - Isochem.	June 1968
Define full and partial reactor load parameters - DUN.	July 1968

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RICHLAND
OPERATIONS OFFICERESEARCH AND DEVELOPMENT STATISTICAL SUMMARY
02 PROGRAM

HAN- _____

1. MISSION NO. AND TITLE MISSION SUB-PROGRAM TITLE	<u>No. 17, Highly Enriched Fuel Program</u> _____	2. BUDGET ACTIVITY NO.	_____
3. CONTRACTOR AND CONTRACTOR NUMBER	A. <u>Isochem, Inc.</u> B. _____ C. <u>AT(45-1)1851</u>	4. PERSON IN CHARGE: INVESTIGATOR: INVESTIGATOR:	_____ _____ _____

FIVE YEAR FORECAST

	CURRENT BUDGET YEARS *					5 YEAR TOTAL
	FY 19 67	FY 19 68	FY 19 69	FY 19 70	FY 19 71	
5. MAN-YEARS						
A) SCIENTIFIC AND TECHNICAL	1.2	1.7				
B) OTHER	.4	.5				
TOTAL	1.6	2.2				
6. COSTS (\$000'S)	\$	\$	\$	\$	\$	\$
A) DIRECT LABOR (INCL. COST OF SERV.)	19	27				
B) MATERIALS, TECH. SERV., SUB CONTRACTS	25	45				
C) SPONSORED WITH BNW	103	100	135	175	200	
TOTAL DIRECT	147	175				
D) INDIRECT EXPENSES	11	16				
TOTAL OPERATING COSTS	158	191	200	250	300	
E) EQUIPMENT (NEW)						
F) IRRADIATION UNIT COSTS						

7. COMMENTS:

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ISOCHEM, INC.

MISSION NO. 17, HIGHLY ENRICHED FUEL PROGRAM

SCOPE AND OBJECTIVES

Detailed separations requirements for reprocessing fully enriched fuels or plutonium fuels will depend on: (a) The fuel description---materials, dimensions, U-235 or (Pu) content; and (b) The throughput requirement. Presumably, incentives for maximum throughput will be great in order to assure accommodation of other Hanford reprocessing programs. Thus, the general requirements that govern the detailed scope and objectives include this assumption.

The scope of the Isochem program will include the chemical reprocessing steps commencing with the receipt of irradiated fuels. Following the basic separations processing, the product streams, including neptunium values, if any, will be purified and the waste streams will be disposed of in accordance with the requirements of acceptable waste management practices. The UNH (or Pu nitrate) product from the separations process will be converted to a form acceptable for recycle at Hanford, or for any other indicated use.

The Isochem research and development objectives for such a program would be to provide the technology required by the program as scoped. Basic technology for reprocessing certain types of these fuels is now available. The available technology and experience would provide a basis for the R&D required for application at Hanford. Because of the current status of Hanford reactor and fuel technology, it is anticipated that initial programs would stress engineering studies of separations processes that would complement studies by General Electric and DUN.

During this period, laboratory R&D efforts would be of a preliminary exploratory nature. Subsequently, at about mid FY-1967, the laboratory effort would be expected to accelerate, as the program needs become better defined.

PLANS AND EXPECTED RESULTS

1. FY-1967

During FY-1967, R&D efforts will initially include engineering studies to define the most promising approaches for Isochem reprocessing efforts. Processes consistent with the program scope and objectives

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would be included. To the extent compatible with reactor studies, preliminary studies of (a) flow sheets, (b) criticality requirements (c) safety, (d) equipment design, and (e) materials would be included. Laboratory programs will initially be exploratory in nature. These studies will be concerned with (a) dissolution of conceptual fuel types and materials; (b) dissolver concepts; (c) reduction of the separated products to a useful form; and (d) plant instrumentation requirements.

As the results of the engineering studies become available, definition of the supporting laboratory programs will improve, and more specific program needs will be indicated. By the end of FY-1967, preliminary process selection and flow sheet definition can be made for the simpler candidate fuel forms; and the more complicated fuel materials will require considerable laboratory and engineering support to develop suitable processes of high reliability and flexibility.

2. FY-1968

During FY-1968, the better definition of fuel elements will permit the investigation of separations process details in depth. Flow sheet problem areas will be investigated and equipment and material needs will be defined. Supporting pilot plant investigations to provide process demonstration for advanced fuel forms will be performed. By the end of FY-1968, Isochem requirements for the Hanford program can be defined on a practicable, working basis, to permit the preparation of budget studies to support necessary planning for facility modifications and/or additions.

3. Post FY-1968

Subsequent R&D programs will be aimed at (a) process optimization, and (b) the investigation of promising alternative processes. Information necessary for design study and facilities scoping will be completed.

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Schedule of Activities

Establish process and flow sheet for DUN and GE primary
fuel candidates.

September 1967

Complete pilot plant process demonstration.

June 1968

Complete planning of facility modifications.

January 1969

(Have modifications in place)

June 1970

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