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HEAVY WATER ORGANIC COOLED REACTOR

QUARTERLY TECHNICAL PROGRESS REPORT INCEPTION THROUGH JUNE 1965

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PREFACE

This report is the first in the series of quarterly technical progress reports of research and development work conducted under the Heavy-Water, Organic-Cooled Reactor (HWOCR) Program for the United States Atomic Energy Commission. It covers the period from contract inception on January 8, 1965, through June 30, 1965. The work is being performed jointly by Combustion Engineering and Atomics International as a single, fully coordinated program.

The report is divided into sections following the task and sub-task numbering system in use during Fiscal Year 1965. Tasks performed jointly by AI and CE during the year include those designated 03, 06, 09, and 10. Consequently, most of the sub-task titles in these sections are identical, since each company contributes a complementary portion of the work. Sub-task numbers with a "1" in the third digit designate Combustion Engineering, while a "5" in this position indicates Atomics International.

All figures referenced within each sub-task will be found at the conclusion of text material in that sub-task.

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HWO CR QUARTERLY REPORT	
TITLE: HWO CR Program Summary	SUBTASK NO.: 01
PROJECT ENGINEER: R. J. Rickert C. A. Trilling	REPORTING PERIOD: Inception through June 1965

I. PROGRAM OBJECTIVES

The HWO CR Program has been undertaken to develop an advanced converter reactor type which will generate energy at competitive costs in the immediate future, and which will yield increasing economic benefits as time goes on because of inherent superiority in those characteristics which will become most important as nuclear plants come to supply a large fraction of the nation's energy consumption. The most important of these characteristics is high fuel utilization--the ability to produce a large amount of energy per unit of fissile isotope used up. So far as is currently known, the resources of natural fissile isotopes are small in relation to the size of the expected nuclear power demand, but they can in principle be stretched to meet the demand comfortably through the judicious use of high-utilization converters--such as the HWO CR--and breeders. This circumstance imposes an urgency upon the development of the HWO CR, for with the rapidly expanding use of nuclear power, the industry is building up a momentum in the direction of low fuel utilization which it will find harder and harder to re-direct as time goes on.

In the accomplishment of this broad purpose, the role of the HWO CR Program is to meet the following specific objectives:

- a) Develop a basic design of an HWO CR capable of economic power generation in the near future when built in sizes which will lie in the upper range of near future utility practice (1000 Mwe and above).

b) Incorporate in that design the basic characteristics which may be exploited for continuing economic improvement in the expanding nuclear power industry. These are the characteristics which will make possible:

- 1) High fuel utilization
- 2) Low fuel cost
- 3) Large single unit capacity
- 4) Favorable economics for the dual-purpose production of electricity and energy for other purposes, such as water desalting.

c) Demonstrate these accomplishments by an operating 300-Mwe plant in the early 1970's followed by a second 1000-Mwe plant in the middle 1970's.

The objectives stated above pose two basic problems: the problem of designing a demonstration plant which will show favorable power costs in the economic climate of the early 1970's while demonstrating at the same time the advanced characteristics which may not yield economic benefits until a later date, and the problem of carrying out the development rapidly enough to meet the time objective.

The contradictions inherent may be illustrated by some considerations with regard to fuel utilization. High fuel utilization requires that the absorption of neutrons by parasitic materials in the reactor be reduced to a ~~minimum~~. The organic coolant of the reactor is itself a major parasitic sink of neutrons and, consequently, the utilization goal requires that it be used sparingly. On the other hand, all reactor experience to date has indicated that the attainment of high power density is a major factor in minimizing energy production costs. This requires the generous use of coolant. At some time in the future, when the limitations of fissile isotope supply are serious enough to affect the cost of feed uranium, an economic balancing of fuel utilization considerations against power density considerations will occur naturally, and the coolant content for

each particular reactor design will be determined by this balance. At present, while the installed nuclear capacity is small relative to the fissile isotope supply, there is little economic reward for high utilization, and the most economic reactor design will inevitably be one which uses coolant generously. In the HWOCR Program, whenever there is such a discrepancy between short-term economic performance and longer-range advances, ways will be sought to attain low power costs while still pointing the way to the future advances, so that they can be realized when future economic pressures prove the need for them.

II. PROGRESS DURING REPORT PERIOD

The first major fuel irradiation experiment of the HWOCR Program was designed, and fabrication was nearly completed by the end of this reporting period. Insertion in the organic-cooled, U-3 loop of the NRU reactor at Chalk River is planned for late July. The experiment (EXP-NRU-305) consists of an 8-ft string of five 18-rod fuel bundles. The lower two bundles are assemblies of UO_2 fuel in Zircaloy-4 cladding. The upper three bundles are UC fuel in SAP cladding. Three of the five fuel bundles, together with a backup string, were shipped to Chalk River at the end of the period. Planning began for the preparation of fuel elements for irradiation in the WR-1 reactor at Whiteshell. Initial delivery of fuel assemblies is scheduled for January 1966.

Characteristics of a reference UO_2 -fueled, 1000-Mwe plant were established. Reactor physics parameter surveys were conducted over a range of design variables for UO_2 rod, UC rod, and UC annular fuel elements. Heat transfer analyses for these configurations were initiated. A reactor physics test case to compare calculational techniques in use at AI, CE, and ORNL yielded substantially the same results.

Various reactivity control methods were examined as to their applicability to the HWO CR, and two appear promising for further study. Both are variations of a solid-absorber, fluid-actuated type using D_2O as the actuating fluid. Refueling methods to achieve bi-directional refueling were studied, and one was chosen as a reference. Design efforts were initiated on critical components.

A procedure for conducting an optimization study on the main coolant heat transfer system was developed. Design limitations and ground rules were established, together with cost data for the major equipment components. Steam cycle heat balances were calculated. The development of a computer code for use in optimizing the main coolant and steam systems was initiated. Conceptual design of the D_2O moderator system was begun following an extensive literature search and visits to nuclear plants having heavy water systems design and operating experience.

The results of work on the reference 1000-Mwe plant were utilized in providing the initial process systems designs for the 300-Mwe demonstration plant. The first representations of size, configuration, and layout of the 300-Mwe plant were completed.

Design studies of various fuel channel assemblies were made, and a preferred concept utilizing a hot process tube insulated from the calandria tube by a gas gap was selected for initial testing. Reference designs of hot-pressed, rolled, and tandem-extruded transition joints between steel end fittings and the SAP or zirconium alloy process tube were selected for development and evaluation. Design of Zircaloy-4 and SAP experimental process tubes was completed, and material for the SAP tube was ordered. A study of the use of $Zr-2.5\% Nb$ as a process tube material was initiated. Seals for the process tube end closures were ordered for evaluation. Studies of methods for fabricating the fuel channels

resulted in procurement being initiated for SAP rolled-joint specimens. Equipment for producing test joints was designed and is being procured.

Designs of four fuel channel test facilities were initiated. These include: (a) the component test loop (TF-10) which will be used to test various concepts of fuel channel components; (b) two five-channel test loops (TF-20 and -25) which are to be used to conduct process tube life tests under simulated reactor conditions of thermal cycling, stress, wear, and corrosion; and (c) a seal test facility (TF-50) which will be used to test various types of seals required for the HWOCR. Design was completed and construction of TF-10 was initiated. A smaller loop (TF-1), using HB-40 as a coolant, was put into operation to test smaller equipment such as pumps, bearings, and valves in an organic coolant medium.

The program to determine the feasibility of using zirconium-base alloys for the in-core process tube and fuel element cladding was initiated with a review and evaluation of the literature on corrosion, hydriding, mechanical properties before and after hydriding, and irradiation effects. Installation of test equipment is under way, and the test program has been initiated. Test coupons coated with potential hydrogen barrier materials were sent to Chalk River for irradiation as part of the U-305 experiment.

Preliminary evaluations indicate that the extrusion of Zircaloy-4 fuel cladding tubes with integral ribs may be a commercially feasible process and should be pursued with emphasis on extrusion followed by drawing. Until such processes are firmly established, electron beam welding appears to be a useful and reliable method for the attachment of ribs to Zircaloy-4 fuel cladding.

SAP-895 obtained from Canadian stock can be extruded readily as integral ribbed fuel clad tubing. Yields of 2-ft lengths based on eddy current and dye penetrant tests were low.

Work directed toward developing a reliable, low cost process for producing hyperstoichiometric UC fuel elements has produced several process improvements. Surface quality of the slugs was improved by the use of previously outgassed molds, casting into hot molds and casting with the melt at slight superheat. Preliminary results indicate that centerless grinding can provide a satisfactory, economical surface finish. Tests on the carbothermic reduction of UO_2 indicated that hyperstoichiometric UC castings by carbon arc melting can best be made by using slightly hypostoichiometric UC melt stock with the balance of the required carbon made up from the arc melting electrode. A UC fuel production pilot line study was initiated. Scale-up of the skull casting operation from the prior reference pour of 18 kg to a 31-kg pour was accomplished. Process operation batch sizes and yields for 44,000 kg UC per year through-put were calculated, and a preliminary process flow chart and facility layout were prepared.

A program to evaluate billets of SAP alloys procured from existing inventory sources was initiated. This effort includes short and long term mechanical tests, metallographic evaluation, and dimensional measurements. Promising techniques for rapid joining of SAP, viz., flash welding and magnetic force welding, were selected for further development. Ultrasonic techniques for non-destructive testing were shown to be capable of detecting billet and tubing defects and measuring finned tubing wall thickness. An evaluation of improved methods for Al_2O_3 determinations in SAP showed that halogen-organic and hydrogen evolution procedures warrant further investigation. Conventional techniques were found to

be acceptable for C, H, and N analyses. A preliminary evaluation of irradiated SAP cladding from OMRE fuel elements indicated no adverse irradiation effects. A coordinated research program was established with CRNL to develop basic technology for obtaining consistent SAP properties and for SAP fabrication processes.

The conceptual design and cost estimate for two proposed in-pile test loops to be installed in the WR-1 reactor were completed. These loops are to be used for irradiation of advanced fuel element designs and cladding materials. A conceptual design and cost estimate of a large components test loop was also completed. The loop would be used to prove the operating characteristics and reliability of HWOCR primary loop circulating pumps, valves, and other large components.

A compilation of organic coolant physical properties was prepared. Two methods for chlorine analysis were investigated (neutron activation and x-ray fluorescence) with good early results. A new method of measuring film formation by x-ray fluorescence properties was investigated, and preliminary tests look promising. Plans were made and equipment ordered for carrying out experimental work on the oxidation of irradiated coolant and its effect on film formation.

Evaluation of the degree of sophistication required in reactor physics calculation methods in order to adequately predict the nuclear characteristics of the HWOCR-type lattice was initiated. Recently developed digital computer codes were made operational to investigate the use of spatially dependent thermal neutron spectra; the effect of the lattice heterogeneities; and the accuracy of methods for transforming the lattice cell to less complex geometric representations. Some existing nuclear cross-section libraries have been updated to include the latest experimental data.

The flow of technical data of value to the HWO CR Program from Canadian reactor sites began with the assignment of project personnel at Chalk River and WNRE. Technical coordination and assistance in the U-3 loop fuel irradiations were also provided.

A detailed, long-range Program Plan for the development of the HWO CR was completed and submitted to the AEC.

FUEL CHANNEL ASSEMBLY DEVELOPMENT

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HWO CR QUARTERLY REPORT

TITLE: Fuel Channel Design, Development and Procurement SUBTASK NO.: 0211

PROJECT ENGINEER: J. F. Rohlin

REPORTING PERIOD: Inception through
June 1965

I. PROJECT OBJECTIVES

The objective of this task is to develop a fuel channel assembly suitable for service in a Heavy Water Moderated Organic Cooled Reactor. To achieve the task objective, the assembly must be designed to attain the following characteristics:

- 1) Minimum neutron absorption
- 2) Conformance with ASME code allowable design stress
- 3) Compatibility with on-line refueling techniques and equipment
- 4) Short-time removal and replacement potential
- 5) Cost consistent with the economic goals of the plant
- 6) Detection and location of leakage
- 7) Expected life not less than 30 years

The problem areas associated with the fuel channel assembly include:

- 1) Wear due to fuel movement
- 2) Corrosion
- 3) Damage due to vibration
- 4) Excessive heat loss to moderator and gas systems
- 5) Stress problems due to thermal cycling associated with reactor
scram and refueling
- 6) Space limitation due to close lattice pitch
- 7) External loads caused by thermal expansion of fuel channel
assembly and interconnecting piping.

The present pertinent experience with fuel channel type reactors is with pressurized coolant. Useful information is available from this experience on procurement, methods of connecting zirconium alloy tubes to extensions, end-cap seals, and field erection. The HWOCR conditions of higher temperature (750°F) and environment are sufficiently different that the base established from actual operating reactor experience (such as NPD and CANDU, CVTR and PRTR) is somewhat limited.

The fuel channel assembly consists of a zirconium alloy or SAP process tube attached to steel extension pieces by means of rolled joints, tandem extrusions, or other types of appropriate joints. Surrounding the process tube is a calandria tube which will be constructed of either zirconium or aluminum alloys. The annulus between tubes will be filled with a gas to provide a thermal insulation barrier between the hot process tube and the cold calandria tube. Each end of the assembly is equipped with a remotely removable end seal, to permit insertion and removal of fuel elements. Fuel elements are positioned within the process tube by means of internal structural members which can be moved and manipulated by the refueling machines. The upper and lower ends of the assembly will have piping connections to provide access for the organic coolant flow through the fuel channel.

II. PROGRESS DURING REPORT PERIOD

A. DESIGN STUDIES

Work accomplished during this report period consisted of an investigation of the merits, shortcomings, and adaptability of available types of fuel channels, with regard to the HWOCR concept. This investigation included the hot tube, gas balance, and the solid insulation tube concept. The hot tube concept was selected as the reference for further design and investigation as it appears to be the most promising design.

The hot process tube concept is the simplest and most straightforward of all the types studied. It consists of a process tube which is exposed to the full temperature and pressure of the primary system (Figure 0211-1). This concept has been used extensively in pressurized water tube reactors and has had a considerable amount of development work. The HWOCR condition of higher temperature (750 to 800°F) and environment are sufficiently different that the base established from actual operating reactor experience is somewhat limited.

The gas balance concept utilizes a hot process tube in conjunction with a pressurized annulus between the process and calandria tubes. This design approach results in less structural material in the core because a portion of the load is applied to the cooler calandria tube. To accomplish this savings, a number of associated mechanical problems are introduced, causing uncertainties regarding the desirability of this concept when all of the considerations are evaluated. The disadvantages include:

- 1) Seals are required at each tube annulus. This is an undeveloped component that must provide a sliding-type seal in a radiation and high temperature environment (Figure 0211-2).

- 2) Inaccessible gas pressurizing lines to each annulus are required. At this time it does not appear feasible to design an arrangement where maintenance or replacement of these gas lines can be accomplished readily (Figure 0211-3).

- 3) The concept would require larger penetrations through the upper and lower shields in order to install and maintain the seals. This will undoubtedly make the design of the shields more difficult and will require a larger diameter mounting arrangement, which will increase the component integration problems above and below the shields. The gas balance concept

was relegated to a longer range schedule because of the development problems associated.

The solid insulation concept consists of an integral assembly of the process tube, calandria tube, and a solid-packed insulation between tubes. There was considerable Canadian development work in this area before the emphasis was shifted to hot tube designs using Zircaloy. Thusfar in the current HWOCR Program, there has been little design effort with respect to this concept, but it is considered a potential candidate.

An interesting variation of the hot-tube concept is the use of a slip joint (Figure 0211-4). In the hot-tube design where the entire arrangement is an integral assembly over its entire length of 50 to 60 ft, some problems with thermal expansion can be foreseen, particularly where SAP is involved. With a stationary upper end on the assembly, a great deal of movement at the lower end could be troublesome, because of interconnecting piping and refueling machine operations. The slip-joint concept would permit both ends of the fuel channel to be fixed (stationary), thus removing these possible problem areas. The first test section for the components test loop (TF-10) will provide an early and convenient test of this component. A stainless steel simulated process tube is used in this test to provide the expansion movement necessary for joint test and evaluation. This test will give an indication of the slip-joint potential early in the program, prior to the testing of SAP and Zircaloy process tubes.

The initial tests will be followed by SAP-TF-10 test section (Figure 0211-5) on about October 1, 1965 and with the Zircaloy-TF-10 test section (Figure 0211-6) on approximately December 1, 1965.

The selection of assemblies designed for the Five-Channel Loop (TF-20) was based on the following guidelines:

- 1) Include both SAP and Zircaloy-4
- 2) Include the following transition joints:
 - a) Tandem extrusion (Zr/CS) (Figure 0211-7)
 - b) Zircaloy rolled joint (Zr/CS) (Figure 0211-8)
 - c) Hot-pressed joint (SAP/CS) (Figure 0211-9)
 - d) SAP rolled joint (SAP/CS)
- 3) Include both Hot Tube and Gas Balance concepts
- 4) Include both welded and seamless Zircaloy-4 tubing and possibly zirconium 2-1/2 Nb, if procurement considerations will permit.

Zircaloy-4 is the more fully developed material, but Zircaloy 2-1/2 Nb may offer a potentially thinner tube due to its increased short-time strength. Corrosion, wear, and creep properties need to be evaluated fully at HWOCR operating conditions.

As part of this program the properties, cost data, and relative merits of seamless and welded tubing need to be evaluated. Indications are that a more uniform wall thickness can be obtained from welded tubing, thus allowing the use of less structural material in the core. In addition, it appears that welded tubing may offer some cost savings, based on available information on other welded tubular products.

In seamless tubing, the diameter to wall-thickness ratio may present some problems. The Zircaloy-4 wall thickness is 0.75 in.; vendor contacts indicated this ratio appears within present production capability. The zirconium 2-1/2 Nb tubing may be thinner in wall thickness if short-time properties are a relative indication; however, the feasibility of obtaining

significantly thinner wall thickness tubing to our dimensional requirements cannot be ascertained at this time. Further investigation into this problem area will be made.

The design basis for the wall thickness determination followed ASME Code method of establishing nominal sizes and applying a corrosion and wear allowance. In the case of the SAP 895 process tubes, the controlling allowable stress value of 4900 psi at 710°F was found to be the 80% of 30-year stress rupture strength. This combination of temperature and pressure is the maximum coincident design condition and it occurs at the outlet of the first pass of a two-pass system. This resulted in an .090-in. wall requirement, including corrosion and wear allowance. The manufacturing tolerance will make the final size .090 $\begin{smallmatrix} +.020 \\ -.000 \end{smallmatrix}$ in.

The Zircaloy-4 seamless process tube controlling allowable stress of 7900 psi at 710°F is found to be one-third of ultimate tensile strength. This maximum coincident condition also occurs at the first-pass outlet. For reasons previously outlined concerning thin-walled extrusions of this diameter (4.000 in. ID) meeting our dimensional requirements, a heavier walled tube will be used initially. This tube has a .075 $\begin{smallmatrix} +.015 \\ -.000 \end{smallmatrix}$ in. wall thickness. Should results of procurement, assembly, and tests be favorable, a reduction of this thickness will be made.

B. SINGLE-CHANNEL LOOP TEST SECTION

1) Stainless Steel Tube/Slip Joint (Figure 0211-4)

Design and fabrication complete; this section will be used in TF-10 initial test.

- 2) SAP Tube Assembly (Figure 0211-5)
 - a) Design complete
 - b) Tubing on order (delivery 9-15-65)
- 3) Zircaloy Tube Assembly (Figure 0211-6)

Design complete
- 4) Test Procedures

The preliminary test specifications were completed. They will be modified and revised throughout the program.

The SAP tube section for the TF-10 installation will use rolled joint connections to make the SAP-to-carbon steel transition. The Zircaloy tube section will use tandem extrusion joints to accomplish the Zircaloy-to-carbon steel transition. The overall assembly is designed to permit process tube changes without major disruption to the use of the facility. To the present, tandem extrusion technology has been developed with stainless steel and Inconel rather than with carbon steel, but the vendors contacted indicate that a joint using carbon steel would not present any undue problems. It also should be noted that this application for such a joint would exceed the temperatures on any previous application by 100° to 200°F.

C. FIVE-CHANNEL LOOP TEST SECTIONS FOR TF-20

Tentative selection of test assemblies as outlined in Section has been made and design work is proceeding on schedule. The basic philosophy of selecting various type assemblies for the first five-channel assemblies test has been to select different combinations of transition joints in both Zircaloy and SAP pressure tube material for performing a relative evaluation of their performance under operating conditions. This first series of tests will be in the nature of a screening operation and will be narrowed down in later

efforts.

The test plan for these assemblies is in preparation. Almost all of the reactor operating conditions, other than nuclear radiation and initial heat generation, can be obtained. The following areas of concern are outlined in the test plan:

- 1) Process tube vibration
- 2) Hydraulic performance
- 3) Process tube deformation
- 4) Corrosion and hydriding
- 5) Fuel element vibration
- 6) Wear
- 7) Heat transfer
- 8) Thermal cycling
- 9) External loading
- 10) Performance of end seals, packings, bearing, and related items.

D. COMPONENT DEVELOPMENT PROGRAM

A literature search was conducted and several types of seals for the end closure mechanisms were selected for test and evaluation. The selections were made on the basis of commercial availability and having a metal-to-metal seal with low axial sealing forces. Discussions were held with vendors and the seals were ordered. These seals will be installed in a test fixture (Figure 0211-10) and will be evaluated after testing in the seal test facility (TF-50). The tests will be conducted at simulated reactor steady-state and transient conditions. The tests will be conducted to determine leakage, reusability and general overall performance.

A series of preliminary designs including a modified CANDU and a dome seal type closure are currently in progress. The end closure mechanism must be suitable for on-power refueling and have acceptable leakage and performance repeatability at HWO CR conditions. Both rotary and reciprocating motion are being considered for actuating the mechanisms. Reciprocating motion is preferred because it eliminates an indexing operation from the refueling machine.

The dome seal device is a pressure-assisted-type seal which deforms a ring that produces a seal. It was developed by Hanford for a much higher pressure application, and it is manually actuated by a threaded member. To be adaptable to the HWO CR the design must be modified to provide a push-pull actuating motion and have the dome member re-sized for the lower pressure. This type of sealing approach looks attractive for the application and will be subject to further investigation.

A problem of major concern in this area is the functioning of a device having close fitting machined parts in the presence of organic. The requirement that this seal be handled, installed and removed by machine does not lend itself to a simple mechanism. The performance of such a critical component at HWO CR conditions needs to be demonstrated.

The Gas Balance Seal (Figure O211-2), which is essential to the pressurized annulus required for this concept, is in the evaluation stage. Design studies indicate that the available space for these components, while quite limited, may be sufficient. Several seals, (1) floating bushing, (2) Huhn ring clearance seal, (3) expanding and contracting split ring, and (4) circumferential seals, were considered. Discussions with manufacturers indicate the circumferential seal has the best potential for this application. This

recommendation is due primarily to the excessive gas leakage rate of the other seals. Procurement action has been taken for two sets of seals and a test fixture is being designed for installation in the Seal Test Facility (TF-50).

Several mechanical couplings are being investigated for the jumper tube to fuel channel extension connection. This connection will be subjected to steady-state and transient reactor conditions. In addition, it must be capable of resisting external bending and torque loads due to thermal expansion of connecting piping and the fuel channel. Design studies to date indicate that a number of commercially available couplings meet, or can be modified to meet, the space requirements (Figure 0211-11). Procurement steps have been initiated and testing will be done in the Seal Test Facility (TF-50).

In order to determine tooling requirements and develop procedures for making rolled joints for the Zircaloy and SAP tubes scheduled for the CTL (TF-10), double-ended rolled joint specimens for both metals have been designed. These joints will be evaluated by leakage and thermal cycling tests (Figure 0211-12).

A request for quotation on the tooling has been made and manufacture of the specimens will start in the near future. Testing will include:

- 1) Hydrostatic test at 375 psig.
- 2) Helium leak test at 15-16 psig.
- 3) Series of pressure and axial load tests at 500 cycles each,
with helium leak tests at appropriate intervals.

The initial tests will be conducted at room temperature, with provision for the specimens and associated equipment in future testing at elevated temperatures in circulating organic.

E. OTHER ASSOCIATED WORK

1. Tube Rupture Tests

A study of the desirability, feasibility, and necessity for establishing a tube rupture test program for HWOOR process tubes has been under way during the past quarter. It has been established that the state-of-the-art of SAP and zirconium alloys is lacking in the following

- a) Neither the effective stress theory nor the maximum sheer stress theory is adequate for predicting the failure stress for a multi-axial state of stress in a non-isotropic material. All testing to date of both SAP and zirconium alloys has been uniaxial or 2/1 biaxial rupture testing. No data are available on other biaxial stress ratios which will be seen in the HWOOR process tube (and fuel cladding) because of weight and accumulated pressure drop loads.
- b) The actual stress concentration effect of a known-depth notch in a SAP tube has not yet been determined, but it is a necessary input in setting the required thickness of a SAP pressure tube.
- c) The degree of anisotropy is as yet some unknown function of extrusion ratio.
- d) Hydriding effects on the long-time strengths and anisotropic behavior of the zirconium alloys is unknown.

A tube rupture program is being planned which is designed to fill the gaps in the mechanical properties data required for HWOOR design. Investigation of existing facilities in the U.S. in which this program might be carried out is under way. At present it does not appear that any existing facility would be capable of handling the proposed program without modifications to handle inert atmosphere and/or varying biaxial stress ratio requirements. Investigation of existing facilities is continuing.

2. Welding Machine

Fuel channel assemblies, other than those using rolled joints, will require two field welds to be made remotely to accomplish final assembly. Similar types of welds have been made on other programs and this requirement appears well within the capability of existing technology. The CVTR Program, Atomics International, and Combustion Engineering have developed machines for similar applications.

Procurement specifications for a HWO CR remote welding machine are presently in preparation.

3. Mechanical Properties of SAP and Zirconium-based Alloys

Available data on the mechanical properties of SAP and zirconium-based alloys of current interest to HWO CR are summarized in Figures 0211-11 through 0211-22. The curves drawn in Figures 0211-13 through 1200-16 represent a combination of tube and bar data and are estimated mean values. Figure 0211-17 is a modified Larson-Miller correlation of longitudinal test data for SAP-895, and indicates that long time stress-rupture strengths for SAP-895 in the direction transverse to extrusion is approximately $2/3$ to 75% that in the longitudinal direction. Other data indicate that this anisotropy is a function of fabrication history. The $2/3$ factor is presently being used as the basis for tube designs for HWO CR.

The modification to the standard Larson-Miller plot consisted of plotting a set of isothermal lines associated with the time scales on the right hand ordinate of the plot. This allows determination of the rupture stress for any time and temperature combination, directly from the curve. A linear interpolation between isothermals is also possible. Figure 0211-18 illustrates the SAP design stresses presently being used.

Considerably more data are available for Zircaloy-2 than for Zircaloy-4. Analysis of these data indicates that Zircaloy-2 properties can be used as a basis for a design in Zircaloy-4. This expediency will be employed whenever insufficient Zircaloy-4 data are available. Mechanical property data for the zirconium-based alloys are summarized in Figures 0211-19 through 0211-22. The data shown in these figures are for the annealed condition which will be used as a basis for the current HWOCR design.

One factor which is of extreme importance in the design is the variation of failure stress with variation of loading in a multi-axial stress field for an anisotropic material. All available data indicate that testing of SAP and zirconium base alloys has been confined to pure uniaxial longitudinal loads or tube rupture tests with a 2/1 biaxial stress distribution. No data have been reported at other biaxial stress ratios which will be encountered in HWOCR process tubes and fuel element cladding due to weight and pressure loadings.

4. Calandria-tank Tube Design Data

The design of the Zircaloy-4 calandria tube was dictated by the hydrostatic external pressure of the moderator fluid. A tube thickness of .055 in. has been provided to meet the requirements of the ASME code Section III Nuclear Vessels, 1965 edition. The design approaches presently being investigated to reduce this thickness are honeycomb tubes, corrugated tubes, and ring stiffened tubes.

III. EVALUATION OF EFFORT TO DATE

A. DESIGN WORK

The design work to date indicates that a fuel channel assembly suitable

for HWO CR application is feasible. The degree to which all of the objectives can be met can only be evaluated after manufacturing, erection, and test information is available.

B. NEW PROBLEM AREAS

- a) Development of procurement specifications and delivery lead time of the remote welding machine for installation of process tubes may preclude the use of assemblies requiring remote welding in TF-20.
- b) Evaluation of rolling techniques and rolled joint design for HWO CR-sized process tubing indicates the need for further development.
- c) The suitability of devices such as the end closure mechanisms operating after prolonged periods in organic which has been exposed to atmosphere, requires further study.

IV. NEXT REPORT PERIOD ACTIVITIES

Detailed design of TF-20 test assemblies will continue, as will the evaluation and start of tests in the component development program. The TF-10 tests will start in September 1965.

Efforts in the rolled joint development and testing will continue and a program will be developed for process tube structural tests.

Design an assembly to house second-harmonic-displacement sensors and linear transformers for studying flow-induced vibrations in process channels will be initiated.

FIG 0211-1. PRELIMINARY DESIGN OF A HOT PROCESS TUBE

FIG 0211-2. CIRCUMFERENTIAL SEAL ASSEMBLY

FIG 0211-4. TF-10 S.S. PROCESS TUBE-SLIP JOINT CONCEPT

FIG 0211-5. TF-10 SAP PROCESS TUBE

FIG 0211-6. TF-10 ZIRC. PROCESS TUBE

FIG 0211-7. PROCESS TUBE ARRANGEMENT - TANDEM EXTRUSION - ZR-4 TO C.S.

FIG 0211-8. PROCESS TUBE ARRANGEMENT - ROLLED JOINT

FIG 0211-9. PROCESS TUBE ARRANGEMENT - HOT PRESS JOINT - SAP TO C.S.

FIG 0211-10. SEAL TEST ASSEMBLY

FIG 0211-12. ROLLED JOINT TEST SPECIMEN

FIG 0211-13. LONGITUDINAL ULTIMATE STRENGTH, SAP-895

FIG 0211-14. LONGITUDINAL 0.2% YIELD STRENGTH, SAP-895

FIG 0211-15. TRANSVERSE ULTIMATE STRENGTH, SAP - 895

FIG 0211-16. TRANSVERSE 0.2% YIELD STRENGTH, SAP-895

FIG 0211-17. LARSON-MILLER MASTER RUPTURE CURVE, SAP-895
(LONGITUDINAL DIRECTION)

FIG 0211-18. TRANSVERSE DESIGN STRESSES FOR SAP 895

FIG 0211-19. ANNEALED ZIRCALOY 2 LONGITUDINAL PROPERTIES

FIG 0211-20. ANNEALED ZIRCALOY 2 TRANSVERSE PROPERTIES
REF. ORNL - 3281

FIG 0211-21. LARSON MILLER MASTER CREEP CURVE
ANNEALED ZIRCALOY-2, REF. HW-60908

FIG 0211-22. TRANSVERSE DESIGN STRESSES FOR ANNEALED ZIRCALOY-2 & ZIRCALOY-4

HWO CR QUARTERLY REPORT	
TITLE: Process Tube Testing	SUBTASK NO.: 0212
PROJECT ENGINEER: B.J.Selig	REPORTING PERIOD: Inception through June 1965

I. PROJECT OBJECTIVES

The basic objective of this project is the testing of reactor components and process tubes under simulated reactor conditions. This objective will be accomplished by means of one existing test loop and four major test loops to be built. The existing facility is TF-1, the HB-40 coolant pump endurance test loop. The four planned test loops are: TF-10, the Components Test Loop (CTL); TF-20 and TF-25, two Five-Channel Test Loops (FCL); and TF-50, the Seal Test Facility. TF-10 will be used to test various process tube designs. Process tubes will be life-tested in TF-20 and TF-25, and seals will be thermally cycled and leak-tested in TF-50. These test facilities will be located at Combustion Engineering, Windsor, Building No. 2. A layout of the facilities is shown in Figure 0212-1.

II PROGRESS DURING REPORT PERIOD

A FACILITIES SUPPORT EQUIPMENT

The building area within which the testing facilities will be constructed has been cleared of all unnecessary equipment, to eliminate construction interference. The status of equipment required for Building 2 is summarized in Table 0212-1.

Various methods of steam tracing have been investigated for use in the event that Santowax-OMP or other organics not liquid at room temperature are selected

TABLE 0212-1

BLDG 2 EQUIPMENT REQUIREMENTS

FACILITY/EQUIP	SIZE	SPECIFICATIONS	STATUS	REMARKS
STEAM BOILER	200 H. P.	300 psig STEAM PRESSURE	PURCHASED DELIVERY 7/26	STEAM REQ'D FOR STEAM TRACING LOOPS
COOLING TOWER	6×10^6 Btu/hr		PURCHASED	REQ'D FOR HEAT SINK
NITROGEN	GAS BOTTLES	99.997 % PURE < 2.5 ppm O ₂ -90°F DEWPOINT	DESIGN COMPLETE	N ₂ USED AS PURGE & COVER GAS
AIR		47 SCFM AT 100 psig	INSTALLED	
CARBON DIOXIDE	GAS BOTTLES		DESIGNED	CO ₂ REQUIRED AS COOLANT FOR GAS ANNULUS OF PROCESS TUBE
ELECTRICITY	2 MW		TRANSFORMER PADS INSTALLED MTL TO BE DELIVERED IN AUGUST	EACH CELL OF BLDG 2 WILL BE SUPPLIED WITH 1 MW

as the organic coolant. Tests are presently being conducted to determine whether heat transfer cements will be required in addition to tubing tracer lines.

A still will be provided to reclaim organic coolants, rather than bleed and feed the loops, with the bleed being discarded.

B. COMPONENTS TEST LOOP (TF-10)

Design of a components test loop (TF-10) began in January 1965. This loop will be used to test process tube components design concepts. HB-40 will be the coolant during initial loop and component testing, in order to avoid the additional equipment which would be required for Santowax-OMP. Loop design parameters are listed in Table 0212-2 and the loop test objectives are listed in Table 0212-3. Loop design has been completed, and the fabrication drawings prepared.

All major equipment items, with the exception of the primary pump motor and primary filter, are on site. The pump motor, which was received from the HEOCR facility in Idaho, was sent to the Louis-Allis Company for rewinding. The motor was originally wound for 2300 volt service and is being modified for 440 V service. All material necessary to complete loop construction has been ordered and shop fabrication of the primary piping has been completed. Field erection began in the early part of June with the installation of a pump base. Figure 0212-2 is a photograph showing the work area. The TF-10 isometric single-line drawing and engineering flowsheet are included as Figures 0212-3 and 0212-4, respectfully.

C. FIVE-CHANNEL LOOPS (TF-20, TF-25)

Design of the two five-channel loops for life-testing the process tubes was initiated in April. See Tables 0212-2 and -3 for the loop design parameters and test objectives. Figure 0212-5 illustrates the loop process flowsheet.

TABLE 0212-2

TEST FACILITY DESIGN LIMITS

(PER TEST STATION)

DESIGN LIMITS	TF-10 CTL	TF-20, 25 FCL	REFERENCE REACTOR
FLOW (gpm)	600	600	400
PRESSURE (psig)	500	600	250
TEMPERATURE (°F)	750	800	800
TEST SECTION HEIGHT	25	75	65
TEMPERATURE TRANSIENT	150 °F/hr	200 °F/15 sec	200 °F/15 sec

DIRECTION OF FLOW = UP OR DOWN

TABLE 0212-3

ORGANIC FACILITIES TEST OBJECTIVES

COMPONENTS TEST LOOP

DETERMINE PROCESS TUBE VIBRATION
TEST SLIP JOINT DESIGNS
DETERMINE PROCESS TUBE OPERATIONAL STRESSES
DETERMINE FUEL ΔP & VIBRATION

FIVE CHANNEL TEST LOOP

CONDUCT PROCESS TUBE LIFE TESTS
ACCOMPLISH THERMAL CYCLING SIMULATING REFUELING &
SCRAMMING OPERATIONS
DETERMINE PROCESS TUBE WEAR & CORROSION UNDER EXTENDED
OPERATING CONDITIONS
TEST PROCESS TUBE REPLACEMENT METHODS
DETERMINE END SEAL EFFICIENCY & EFFECT OF REFUELING ON
PROCESS TUBE WEAR & STRESSES
DETERMINE FUEL BUNDLE WEAR & CORROSION UNDER EXTENDED
OPERATIONS
DETERMINE THE EFFECTS OF JUMPER TUBE LOADS ON PROCESS TUBES
TEST VARIOUS TUBE SUPPORT METHODS

Piping layouts have been made to assist in calculating the approximate expansion loop sizes required for holding the loads on the test sections to less than 100 lb at all process-fluid temperatures. Further work must be done in this area, utilizing computer codes to determine exact loadings and expansion loop sizes. A staggered channel arrangement (see Figure 0212-6) will be used to ensure optimum utilization of available space and of the channels for test purposes. Refueling machine testing will be limited to cycling fuel up and down within the process tube to obtain wear data.

Other work in progress or completed relative to these loops is listed below:

- 1) Outline specification (written).
- 2) Loop piping arrangement and test section layout (prepared).
- 3) Preparation of equipment location and engineering flowsheet drawings (in progress).
- 4) Two 3000 gpm Byron-Jackson pumps (primary coolant) (ordered in June 1965).
- 5) Air-operated control valves, in-line filters, loop heaters, basic loop process instrumentation, and thermal-cycling bypass pumps (ordered in June)
- 6) Design of loop heat exchanger (in progress).
- 7) Design of purification system (in progress).

An instrumentation system for obtaining fuel bundle vibration data on the five-channel loops is being purchased. Specifically, this equipment will provide simultaneous five-point displacement measurements of the individual fuel elements, relative to the pressure tube. The output will be recorded on strip chart recorders, thus providing frequency, phase, and wave form data, as well as displacement.

Use of this system to record the motion of fuel elements in the five-channel loops required placing a small magnet in the element, during fabrication, at the point from which motion data are required. Adjacent to this point in the test channel, an opening will be provided through the simulated moderator tank into the thermal barrier. A pickup coil will be placed in the opening against the pressure tube. As the fuel element moves away from or toward the pickup coil, the magnetic field in the iron core will vary, thus varying the amount of second harmonic component which in turn will be recorded on a strip chart recorder.

This system will give a resolution of ± 0.001 inch over a range of ± 0.05 inch and will have a frequency response of 150 cps.

D. THERMAL-CYCLING SEAL TEST FACILITY (FT-50)

Design of a facility to test various types of seals that will be required in the HWO CR project began in May 1965. The process flowsheet (Figure 0212-7) indicates the method of subjecting various types of seals to reactor conditions of temperature, pressure, and thermal transients. The facility will consist of two loops in parallel, each sharing a common leg which comprises several test sections in parallel. Each test section will contain a particular seal or coupling design, thus providing leakage data for various concepts simultaneously. The test seals will be subjected to organic coolant at reactor pressure and will be thermally cycled in a manner similar to that which will be experienced in a reactor. Thermal cycling will be accomplished by valving each of the two loops alternately through the test sections. A programmer will be provided to automatically control the thermal cycling. Instrumentation will be installed to record the significant test parameters as the seals are thermally cycled. Valves and long-lead equipment items have been purchased.

E. HB-40 LOOP (TF 1)

A loop was designed and constructed to test small equipment to be used in the organic experimental facilities. Figure 0212-8 shows the loop, which consists of a Model GA-3K Chem-Pump, 3/4 in. and 2 in. pipe and a tank reservoir. The pump and bearings are being life-tested for operation with organic fluids; in this case, HB-40. The loop has been operating for eight-hour periods at 130°F since June 16, 1965.

F. LOOP COOLANT ANALYSIS

In order to perform realistic tests, the loop coolant must be controlled carefully. Control will be realized by frequent coolant analyses for impurities, high boilers and physical properties. Work is in progress to develop the capability to monitor the organic coolants in the loops presently being built.

All of the major equipment and most of the laboratory apparatus for the coolant analysis program has been received. This equipment includes:

- 1) Perkin-Elmer, Model 337, Grating Infrared Spectrometer
- 2) F & M, Model 500, High Temperature Gas Chromatograph
- 3) Beckman, Model DK-1A, Recording UV-Visible-Near IR Spectrometer
- 4) Beckman, Aquameter
- 5) Beckman, Expandamatic pH meter with oxygen analyzer attachment

and a large assortment of distillation and heating equipment.

While the major effort has been devoted to unpacking, checking and setup of incoming equipment, some analytical procedures have been established. Macro- and micro-distillation apparatus have been set up on temporary racks and high boiler determinations have been run on synthetic mixtures. Fabrication of the drying and distillation units associated with the potentiometric titration in

the Karl Fischer analytical method has started. The infrared spectrophotometer has been set up and test runs are now being made.

III. EVALUATION OF EFFORT TO DATE

TF-1 results to date indicate that HB-40 is amenable to normal piping practices; i.e., use of Teflon tape for sealing pipe threads, use of threaded joints and normal process instrumentation. The pump and bearings continue to function normally after 130 hours of operation. Progress in the other areas has been good, design and procurement are proceeding on schedule.

IV. NEXT REPORT PERIOD ACTIVITIES

A. BUILDING NO. 2 FACILITIES

The Steam Boiler and Cooling Tower will be received and installed in August. Routing of the piping for the steam and cooling water, including a distribution system for steam and water in Building 2, will be accomplished during the next three months. Nitrogen and CO₂ lines will be installed in July and August. The transformers and switchgear are to be delivered in early August.

A heat transfer cement was tested on an 80' test pipe (in July) to determine the relative advantages of using the cement with steam tracing.

B. CTL (TF-10)

Piping construction and electrical and instrumentation installations will be completed in August. Loop shakedown will begin in the latter part of August. Actual testing (with HB-40 as the initial coolant) will begin early in September.

C. FCL (TF-20)

Upon completion of the loop design in September, shop fabrication will begin. All equipment will be ordered by the first week in August, and major material item orders will be completed by the end of September. Delivery of the primary pump cannot be made before the end of December. Based on the preceding schedule, the loop start-up date for TF-20 is slated for early February 1966.

D. SEAL TEST LOOP (TF-50)

Loop design will be completed and all equipment will be ordered by August 1. Fabrication of the loop will begin in August and will be completed by mid-September.

E. HB-40 LOOP (TF-1)

The HB-40 loop will continue to operate on an 8 hour/day, 5 day/week schedule. The loop will be insulated to obtain higher operating temperatures.

F. LOOP COOLANT ANALYSIS

Work to set up the organic coolant analysis equipment will continue. Procedures for sampling and analyzing the coolant will be written. Mechanical design of the TF-20 and TF-25 loop purification systems and sampling systems will be completed by the end of August.

FIG 0212-1. LAYOUT OF TEST FACILITIES IN CE BUILDING NO. 2

FIG 0212-2. PHOTOGRAPH OF TF-10 LOOP WORK AREA

FIG 0212-3. ISOMETRIC VIEW OF THE TF-10 LOOP

FIG 0212-4. PROCESS FLOW SHEET FOR THE TF-10 LOOP

FIG 0212-5. PROCESS FLOW SHEET FOR THE TF-20 LOOP

FIG 0212-6. SCHEMATIC OF THE TF-20 LOOP

SYMBOLS

MEASURED VARIABLE \ TYPE OF DEVICE	CONTROL ELEMENT	INDICATION	RECORDER	SWITCH	TRANS	GAGE
FLOW		FE	FR	FT		
PRESSURE		PI				
TEMPERATURE		TE	TI	TR	TS	
HAND(ACTUATED)	HC					
LIQUID LEVEL						LG

NOTES:
 1. RESERVOIRS TO BE INSULATED.
 2. PIPES TO BE STEAM TRACED.

FIG 0212-7. PROCESS FLOW SHEET FOR THE TF-50 LOOP

FIG 0212-8. PHOTOGRAPH OF TF-1 LOOP

FUEL ELEMENT ASSEMBLY DEVELOPMENT

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1) The maximum pellet temperature is reduced, thereby reducing the potential for radial ratcheting and radial ridging and also reducing the amount of differential axial elongation between fuel and cladding. In addition, the lower fuel temperatures will lead to lower total gas release.

2) The fission gas gap at the end of the fuel bundle can be eliminated, thereby reducing power peaking at the end of the bundle.

3) For a given rod length, the heat rating (kw/ft) and heat flux required to produce the total power are reduced, since there is no gap at the end of the rod.

In the U-305 Experiment, the maximum fuel temperature was reduced from 3800° to 3000°F and the peak heat rating and heat flux were reduced 8% by the use of cored pellets. The volume required to accommodate fission gas release was compared for solid and cored pellets. The gas in a cored pellet is stored at a high temperature which tends to increase the required volume. This effect is offset because the cored pellet is cooler and produces less fission gas. The net result for the U-305 Experiment is that the total volume required for fission gas release is approximately the same for the core and solid pellet.

Thus, the ratio of fuel to cladding and coolant is not affected. There appears to be little cost differential between solid and cored pellets. Some quotations from pellet vendors were identical for both types of pellets for the U-305 Experiment.

From calculations on the U-305 Experiment, it was concluded that SAP fins are effective in reducing the surface heat flux and that zirconium fins are not. This results from the difference in conductivity between the two materials. (The conductivity of SAP is ~ 144 Btu/hr-ft°F and ~ 8 Btu/hr-ft°F for zirconium.) Therefore, if fins or wires are used to space zirconium-clad fuel bundles, the

number should be limited to as few as is necessary to adequately space and support the fuel rods.

A digital computer code (unpublished) which does thermal and hydraulic calculations and which can be used for the design of reference fuel assemblies was obtained from AECL. Using the power distribution, geometry, and inlet conditions, it computes the temperature distribution of the coolant and cladding in a single process channel. The code was converted from APEX-4 to FORTRAN so that it could be used on the IBM computer. It was put into operation and used for the U-305 Experiment calculations.

Some preliminary design work has been performed on a 37-rod, SAP-clad, UO_2 fuel rod bundle. In order to reduce peaking at the ends of the fuel bundle, a staggered arrangement, where the ends of the fuel rods are at different elevations, was considered. Staggered rods and end plates require a design which is mechanically complex and requires orientation of the fuel bundle in the process channel. This approach will be investigated only if a substantial neutron economy can be demonstrated, or if there is a serious power peaking problem at the rod ends.

III. EVALUATION OF EFFORT TO DATE

Cored fuel pellets are preferred over solid pellets and will be used in the reference design for UO_2 fuel rod bundles, as discussed above.

IV. NEXT REPORT PERIOD ACTIVITIES

During the next report period, effort will be directed toward developing an initial reference design for a SAP-clad fuel assembly. This design will be based on the list of reactor characteristics presented in Subtask 0910. They

1) The maximum pellet temperature is reduced, thereby reducing the potential for radial ratcheting and radial ridging and also reducing the amount of differential axial elongation between fuel and cladding. In addition, the lower fuel temperatures will lead to lower total gas release.

2) The fission gas gap at the end of the fuel bundle can be eliminated, thereby reducing power peaking at the end of the bundle.

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III. EVALUATION OF EFFORT TO DATE

Cored fuel pellets are preferred over solid pellets and will be used in the reference design for UO_2 fuel rod bundles, as discussed above.

IV. NEXT REPORT PERIOD ACTIVITIES

During the next report period, effort will be directed toward developing an initial reference design for a SAP-clad fuel assembly. This design will be based on the list of reactor characteristics presented in Subtask 0910. They

will be used to establish fuel element fabrication process development requirements and as a reference for future WR-1 in-reactor tests discussed in Subtask 0312. Calculations will also be performed to quantitatively determine end peaking in fuel assemblies.

HWO CR QUARTERLY REPORT	
TITLE: Fuel Element Testing and Fuel Element Process Development	0312 and SUBTASK NO.: 0313
PROJECT ENGINEER: R. H. Young W. P. Chernock	REPORTING PERIOD: Inception through June 1965

I. PROJECT OBJECTIVES

The long range objectives of this task are the design, fabrication development, testing, and evaluation of oxide fuel element assemblies for HWO CR which reflect the following characteristics.

- a) High neutron economy
- b) Low fuel cycle costs, less than 1 mil/kwh
- c) Burnup capability of at least 20,000 Mwd/MTU
- d) Peak cladding (internal surface) temperature of 850°F
- e) Linear power rating in excess of 11 kw/ft
- f) Low peaking factors
- g) Bundling schemes which exhibit
 - 1) Low parasitic neutron absorption
 - 2) Minimal end plate thickness to reduce axial peaking
 - 3) Low fabrication costs
- h) Ease of fuel shuffling
- i) High performance in organic coolant flowing at velocity of at least 30 ft/sec with an outlet temperature of at least 700°F.

Of the possible fuel element configurations which could be considered for HWO CR, rods represent the most developed concept. It is on this basis that initial effort on this task will be devoted to the design, development, testing

and evaluation of rod-type oxide elements. Later efforts will be directed toward improving the performance of the rod-type fuel elements by increasing the heat rating and burnup.

Throughout this entire program maximum utilization will be made of all available technology on oxide elements being utilized in light water technology in this country and as part of the Canadian Heavy Water Program.

Immediate objectives of this task during the current reporting period were:

- a) Evaluation of the U-3 loop* facilities.
- b) Design of the SAP-clad and zirconium-alloy-clad UO_2 fuel assemblies to match existing conditions in U-3 and to provide maximum information for HWO CR.
- c) Purchase of all components necessary for the fabrication of the oxide fuel element assemblies.
- d) Initiation of the fabrication of oxide fuel element assemblies for insertion in the U-3 loop.

A major portion of the effort during this reporting period was directed toward this irradiation test (referred to hereafter as the U-305 experiment**) in the U-3 loop.

* An organic cooled loop in the Canadian NRU reactor at Chalk River. This loop has been made available to this program throughout FY 1966.

** The U-305 experiment contains five fuel element assemblies. Three of these were scheduled as SAP-clad carbide, fabricated by AI. The fourth and fifth elements were scheduled as SAP-clad oxide, and Zircaloy-4 clad oxide, both fabricated by CE.

II. PROGRESS DURING REPORT PERIOD

One of the major objectives of the U-305 experiment was the early establishment of irradiation performance of fuel elements and fuel element assemblies of interest to HWOCR. A detailed literature survey on the performance of oxide elements was completed and was used as a basis for decisions concerning the design of the oxide fuel elements and fuel element assemblies for the U-305 experiment. Although extensive data were available concerning the performance* of UO_2 fuel elements in light water cooled systems, only a limited amount of information was found concerning the performance of oxide elements in organic environments. A summary of the vast amount of water technology pertaining to UO_2 fuel systems has been provided in a recent Geneva paper.¹ A summary of oxide fuel element performance in organic cooled systems is outlined briefly below:

a) OMRE Irradiation Testing of SAP-Clad Oxide Elements²

Ninety-eight SAP-clad oxide rods were irradiated in OMRE to a peak burnup of $\sim 11,000$ Mwd/MTU at a maximum heat flux of $150,000$ BTU/hr-ft². These rods contained an active fuel length of 31 in. and a fuel diameter of 0.300 in. with a fuel-clad gap of 0.003 to 0.005 in. The cladding was 0.030 in. thick and contained eight fins. The maximum cladding design temperature was 850°F and the rods operated at a heat rating in the range of 17 to 20 w/cm. Eutectic bonding was used as the end closure method. Post irradiation examination indicated no failures, no reactions between the UO_2 and SAP,

* It should be noted that light-water-cooled power reactor systems operate at temperatures of 150 to 200°F below those anticipated in HWOCR.

1. Geneva Paper, P. 233 (United States): Oxide Fuel Fabrication and Performance.
2. NAA-SR-9999

no evidence of leakage at the eutectic bonded end closures and fission gas release below 2%.

b) SAP-Clad UO₂ Irradiated in the X-7 Organic Loop^{1,2}

A series of irradiations of SAP-clad UO₂ fuel elements have been performed in the X-7 organic loop. Pertinent data are summarized below:

	<u>704-I</u>	<u>704-II</u>	<u>709-I</u>	<u>709-II</u>	<u>712</u>	<u>713</u>	<u>714-I</u>	<u>714-II</u>	<u>HWOCR (Tentative)</u>
Clad	M583	M583	M583	M583	M257	M257	SAP-930	SAP-930	SAP or XAP
Length (ft)	1	1	2	2	4	4	8	8	3 to 5
Heat Rating w/cm (Jkd [⊖])	38	46	38	45	37	37	38	40	30 to 40
Max. Clad Temp. (°F)	880	890	870	880	925	900	925	975	850
Max. B.U. (Mwd/MTU)	---	1100	2400	---	2100	1000	1500	1500	20,000
Perfor- mance	F*(1)	G*	G	F(2)	G	G	F(3)	F(3)	---

*F - Failed

G - Good

- (1) One element without spacers bowed out to pressure tube, resulting in burnout.
- (2) Used cladding rejected for use in 709-I. Also experienced gross over-power.
- (3) Longitudinal cracks.

As noted in the above Table, all of the irradiations were performed at heat ratings which produced central temperatures well within the grain growth range for UO₂. In addition, all peak cladding temperatures were in excess

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1. Geneva Paper, P/23 (Canada): Development of Fuel and Coolant Tubes for a Reactor Cooled by Organic Liquid
 2. AECL-1782

of those anticipated (850°F maximum) for HWOCR. Of particular interest in these irradiations is the difference in performance between the 714 tests and the other tests which were successful (assuming that the burnout failure in the rod without spacers and the use of rejected cladding bears little significance to the performance of SAP-clad UO₂ fuel elements). A number of factors which must be considered in the analyses of the 714 failures are summarized below:

1) The SAP-930 utilized in the 714 tests had a substantially lower hoop-stress rupture strength than either M-257 and M-583, utilized in the other tests, or SAP-895 which is the cladding selected for some of the WR-1 fuel elements. This lower hoop-stress rupture strength could lead to early failures as a result of either radial-ratcheting or excessive fuel-clad mechanical interaction resulting from a possible high power transient (noted below).

2) Bowing and twisting in longer elements may have provided a sufficient additional stress to exceed the strain limitations for SAP-930. Again, a cladding with a higher stress-rupture strength would probably have exhibited improved performance.

3) Wall thickness variations may have been excessive, resulting in localized weak points. It is significant that all failures were noted at thin regions of the cladding.

4) Radial ratcheting* and fuel wedging may have occurred. Again, this would have been more pronounced in a cladding with lower hoop-

* Use of cored pellets probably would have eliminated the region of grain growth believed necessary for operation of the radial ratchet mechanism.

stress rupture strength.

5) A power transient (possibly in excess of 10% overpower - no firm measurements) occurred at the end of the 714-I test and resulted in simultaneous failure of two of the three fuel rods.

It is interesting to note that no failures were observed in the U-301 experiment (discussed below) in which 8-ft long SAP-clad (SAP 895) oxide rods were irradiated without incident. In addition, no failures were noted in the 712, 713, or 709-I tests in which shorter fuel assemblies were used. This is particularly significant for HWOCR since fuel element assembly lengths in the vicinity of 4 ft will probably be designed.

c) SAP-Clad UO₂ Experience in the U-3 Organic Loop¹

A nineteen-rod (18 fueled, one dummy) fuel assembly which was 8 ft long and utilized SAP 895 cladding was irradiated in the U-3 loop under conditions which were equivalent to the X-714 test for at least 50% of the 2900 Mwd/MTU burnup logged. No failures occurred. These results tend to support the contention that the use of a cladding with higher hoop-stress rupture strength may be desirable although this fact has yet to be demonstrated conclusively.

d) Zirconium Alloy-Clad UO₂ Experience in the X-7 Organic Loop¹

Two 3-element bundles of Zircaloy-2 clad and Zr-2.5 wt % Nb clad UO₂ were irradiated in the X-7 loop at a coolant* temperature of 320 to 350°C, a cladding temperature of 480°C, (895°F, corresponding to 45°F greater than

1. Personal communication with Canadian personnel.

* The coolant contained from 10 to 90 ppm H₂O.

the peak cladding temperature for HWOCR), and a surface heat flux in the range of 263,000 to 327,000 BTU/hr-ft². All elements contained Zircaloy-2 wire wraps as spacers. An interim examination of these elements after an irradiation of 2700 Mwd/MTU indicated the following:

- 1) No clad failures were found
- 2) All elements were successfully handled in the loop, and during transfer to the hot cells. The wire wrap on one Zircaloy-2 clad element cracked during handling in the hot cell (hydrogen content was 9,000 ppm in the wire).
- 3) Hydrogen contents were measured with the following results:
 - a) Zircaloy-clad elements

Wire wrap	9,000 ppm H ₂
Cladding	500 ppm H ₂
 - b) Zr-2.5 wt% Nb clad elements

Wire wrap	1,500 ppm H ₂
Cladding	250 ppm H ₂
- 4) The wire wrap on the Zircaloy-2 clad elements was brittle whereas that on the Zr-2.5 wt% Nb clad elements retained some ductility.
- 5) The inter-element locking wire had a much lower and more uniform hydrogen content than the wire-wraps. These locking wires were not metallurgically bonded to the fuel elements and were thus not capable of acting as hydrogen sinks.

It was concluded that the high hydrogen contents in the wire wrap were probably a result of thermal gradient diffusion from the cladding. It is significant that the hydrogen concentrations in both cladding materials were low and close to solid solubility values at the temperatures involved. These observations lead to the conclusion that the incorporation of

hydrogen sinks on zirconium-base alloy cladding may provide an important solution to the hydriding problem.

A further implication of the X-719 experiment is a reiteration of the possibility of a brittle to ductile transformation in hydrided zirconium base alloys. This is particularly noticeable in the ability to handle the fuel bundle with 9,000 ppm hydrogen through all operations from the in-reactor loop to the hot cell. Cracking of the spacer wire occurred in the hot cell. If this brittle-ductile transformation **exists**, some advantages may be derived from this factor **at operating temperatures of zirconium base alloy clad elements**.

A possible brittle to ductile transformation in hydrided zirconium base alloys was reported by the Canadians¹; the British²; and the Russians³; and these have been summarized⁴ together with other pertinent hydriding data reported at Geneva. The Canadians indicated¹ a transition from brittle to ductile behavior between 150 to 200°C. This observation led to the conclusion⁵ that on-line refueling was safer than other methods because the zirconium alloy was always operating above its brittle-ductile transformation temperature. This observation would be particularly true for HWOCR systems and offers considerable potential for the ultimate use of zirconium base alloys in this system. However, considerable testing is necessary before the full potential of this solution to the hydriding problem can be realized.

In addition, the remaining elements of the X-719 test have been reinserted in the X-7 loop and that burnups of about 4500 Mwd/MTU have been achieved without incident.

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1. Geneva Paper, P.21 (Canada)
 2. Geneva Paper, P.158 (Great Britain)
 3. Oral discussions of Geneva Papers.
 4. Chernock, W. P. and J. C. Tobin, "Non Fuel Materials Technology," Nuclear News, October 1964
 5. Geneva Paper, P/18 (Canada) (Oral discussion)

Results of the above irradiations were used as a guide in the planning of the U-3 irradiation. Furthermore, the design of the oxide-containing fuel elements for the first WR-1 core was evaluated in eight of potential HWO CR requirements in an effort to utilize the maximum amount of available technology in the planning of the U-305 experiment. A comparison of WR-1 and tentative HWO CR fuel element parameters is presented below.

	<u>WR-1</u>	<u>HWO CR</u> (Tentative)
Cladding Material	SAP-895 ZR-2.5 Nb	SAP or XAP Zr-4 or Zr-2.5 Nb
Peak Cladding Temp (°F)	800	850
Linear Power (kw/ft)	15	15
Surface Heat Flux (finned element) (BTU/hr-ft ²)	300,000	250,000 to 350,000
Coolant Velocity (ft/sec)	36	30 to 40
Rod Diameter (in.)	0.606	0.5 to 0.65
Cladding Thickness (in.)	0.025	0.018 to 0.025
Burnup (Mwd/MTU)	5,000	20,000
Fuel	Solid Pellets	Cored Pellets or VIPAK

Except for fuel burnup, the range of parameters being considered for HWO CR is similar to that for WR-1. Results from irradiations of these WR-1 fuel elements will be of substantial interest to HWO CR. However, there were sufficient differences in the anticipated HWO CR fuel element designs and those for WR-1 that warranted the consideration of inclusion of some of these differences in the design of the U-305 fuel. In considering the design of oxide elements for HWO CR, it soon becomes apparent that design lifetimes and performance can become severely limited if

excessive axial peaking occurs. One method of minimizing the extent of axial peaking in fuel strings containing a number of fuel assemblies, is to adopt a design philosophy which minimizes the distance between the top and bottom of fuel columns in adjacent assemblies. In the case of oxide elements, operating at respectable power levels, fission gas release must be accommodated. For this reason, a gas plenum is usually provided at the end of each element. This plenum would induce axial peaking factors which would be undesirable for HWO CR. Therefore, the HWO CR oxide fuel element design will be concentrated on the use of cored pellets or vibratory compacted fuel. Both concepts provide fission gas voids which are uniformly distributed axially throughout the element and both concepts should provide for a minimum in axial peaking based upon the fission gas problem alone. Thus, it was desirable to consider cored pellets or vibratory compacted fuel for HWO CR. Further consideration of these possible fuel concepts for HWO CR indicated that, if radial ratcheting **were** a problem, both concepts should alleviate this problem by removing the grain growth region in the case of cored pellets and providing a configuration not believed to be susceptible to radial ratcheting, in the case of **solid vibratory compacted fuel**. Recent irradiation test results* from the Commission's High Performance UO_2 Program are encouraging with respect to the oxide configurations selected as the basis for HWO CR. Zircaloy-2 clad fuel rods containing VIPAK UO_2 and cored pellet UO_2 were successfully irradiated to peak burnups of 30,000 and 18,000 Mwd/MTU at a peak surface heat flux greater than

* Lyons, M. F., B. Weidenbaum, R. F. Boyle, and T. J. Pashos, "Post-Irradiation Examination of High-Burnup Molten UO_2 Fuel Rods," paper to be presented at the 1965 Annual Meeting of ANS.

1×10^6 BTU/hr-ft². A comparison of these rods with typical rods anticipated for the HWO CR design is presented below:

<u>Cored Pellets</u>	<u>*High Performance UO₂ Program</u>	<u>HWO CR (Tentative)</u>
Enrichment (%)	3.8	Less than 1.5
Fuel OD (in.)	0.494	0.5 to 0.65
Fuel ID (in.)	0.140	0.14 to 0.18
Density (% of theoretical)	92 to 93	94 to 95
Cladding Thickness (in.)	0.030	0.020 to 0.025
Fuel Length (in.)	30	36 to 60
Peak Surface Heat Flux (BTU/hr-ft ²)	1.4×10^6	0.35×10^6
Peak Burnup (Mwd/MTU)	18,000	20,000
<u>VIPAK</u>		
Enrichment (%)	3.8	Less than 1.5
Fuel OD (in.)	0.503 to 0.505	0.450 to 0.600
Density (% of theoretical)	83 to 85	85 to 88
Cladding Thickness (in.)	0.030	0.020 to 0.025
Fuel Length (in.)	30	36 to 60
Peak Surface Heat Flux (BTU/hr-ft ²)	1.3×10^6	0.35×10^6
Peak Burnup (Mwd/MTU)	30,000	20,000
* ANS Transactions Vol. 8, No. 1, 1965, p. 42		

No circumferential ridging was noted for the VIPAK rods. The maximum ridge height observed for the cored pellet rods was about 0.003 in. and was equivalent to that observed for solid pellet rods operating at much lower surface heat fluxes. It was concluded that circumferential ridging was **believed to be related to the heat rating, free standing rods operating at substantially** higher heat ratings with cored pellets produced ridging equivalent to that found for solid pellet rods operated at lower heat ratings. On this basis, the combination of cored pellets and a lower surface heat flux in HWO CR may permit elimination of radial ridging. Thus, the selection of cored pellet rods and VIPAK rods as a basis for HWO CR fuel element development appears to provide the greatest probability of achieving design objectives.

The cored pellet configuration was selected as the basis for the design of the oxide fuel elements for the U-305 experiment on the basis that it removed the region of the fuel most susceptible to grain growth resulting in lower fission gas release and less tendency toward radial ratcheting. Furthermore, it provided the least departure from light water cooled technology when compared with the vibratory compacted alternate.

Details of other design considerations are presented later in the discussion. Included in these later sections is some background material on many of the fabrication efforts which are discussed.

A major portion of the efforts on Tasks 0312 and 0313 were directed toward the planning, design, ex-reactor testing, fabrication development and the initiation of fabrication of fuel elements and assemblies for the U-305 experiment. Work on Task 0313 was aimed at developing the necessary methods for the fabrication of the test assemblies. All design and backup testing as well as the actual fabrication of the U-305 fuel elements and assemblies was performed as part of Task 0312.

A. U-305 EXPERIMENT

The fuel string designed for this experiment consists of five, 18-rod bundles of fuel which are suspended on a central hanger rod, which is attached to the loop breech block by a support tube. The axial arrangement of the U-305 experiment is shown in Figure 0312-1. The three upper bundles, each 16-5/8 in. long, contain UC fuel clad with SAP. The two 25-1/2 in. long lower bundles contain UO_2 fuel with either Zircaloy-4 or SAP-895 cladding.* The central hanger rod, extending the length of the fuel

* It is presently planned to irradiate two Zircaloy-4 clad oxide assemblies in the U-305 experiment with one SAP-clad and three Zircaloy-4 clad oxide assemblies as backup.

string is shrouded by a dummy fuel cladding tube to provide hydraulic similarity to prototype 19-rod assemblies.

The design of the oxide assemblies of the **oxide-carbide fuel** string is described below together with a summary of the initial fabrication procedures already performed for these assemblies. A discussion of the design and fabrication of the carbide portion of the joint fuel string is presented under Task 0351.

A cross section of a typical UO_2 fuel assembly is shown in Figure 0312-2. The individual fuel rods are arranged in a circular array and are fastened to stainless steel end plates in the SAP bundles and Zircaloy-4 end plates in the Zircaloy bundles. The inner ring of rods in the SAP bundles are free to expand in the bottom end plate to allow for the difference in thermal expansion between the inner and outer ring of rods. Because of the small difference in thermal expansion between the rods in the Zircaloy bundles, the ends of the rods are fixed to both end plates.

The assembly of a typical UO_2 bundle is shown schematically in Figure 0312-3. The spacer ring centers the hanger rod and fuel assemblies in the pressure tube and also provides a large bearing surface area to minimize wear and fretting between the fuel string and the pressure tube. The principal parameters and features of the U-305 experiment are given in Table 0312-1. Clad and fuel dimensions are given in Table 0312-2.

1. Fuel Assembly Design

A number of alternate designs were completed and evaluated for the Zircaloy-4 clad and the SAP-clad oxide assemblies for the U-305 experiment. Within the framework of each cladding type, these alternate designs differed with respect to the method of providing spacers between elements, end plate-fuel rod joining methods, and methods for instrumenting fuel elements.

TABLE 0312-1

1) <u>FUEL ENRICHMENT (WT %)</u>	
UC	1.50
UO ₂ - BUNDLE JRA	1.12
UO ₂ - BUNDLE JPA	1.52
2) <u>INITIAL PEAK BURNUP RATE</u> <u>(MWD/MTU-DAY) (191 MEV/FISSION)</u>	
UC	
OUTER RING	46.8
INNER RING	32.8
UO ₂ - BUNDLE JRA	
OUTER RING	39.5
INNER RING	27.9
UO ₂ - BUNDLE JPA	
OUTER RING	39.5
INNER RING	26.1
3) <u>PEAK ROD POWER RATING (KW/FT)</u>	
UC	20.8
UO ₂	13.5
4) <u>PEAK FUEL CENTER TEMPERATURE (°F)</u>	
UC	~ 2000
UO ₂	~ 3000
5) <u>PEAK CLADDING TEMPERATURE (°F)</u>	
UC	~ 850
UO ₂ - BUNDLE JRA, SAP CLAD	~ 790
UO ₂ - BUNDLE JRA, Zr CLAD	~ 880
UO ₂ - BUNDLE JPA, SAP CLAD	~ 770
UO ₂ - BUNDLE JPA, Zr CLAD	~ 860
6) <u>LOOP INLET TEMPERATURE (°F)</u>	620
LOOP ΔT (°F)	82
7) <u>EXPERIMENT POWER OUTPUT (KW)</u>	
UC	920
UO ₂	740
8) <u>FLOW VELOCITY (FT/SEC)</u>	30
9) <u>LOOP FLOW RATE (GPM)</u>	306
10) <u>LOOP INLET PRESSURE (PSIG)</u>	300
11) <u>TEST SECTION ΔP (PSI)</u>	~ 80

TABLE 0312-2

FUEL O.D. (INCHES)	.522
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FUEL I.D. (INCHES)	.184
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SAP CLAD I.D. (INCHES)	.526
<hr/>	
SAP CLAD O.D. (INCHES)	.576
<hr/>	
WALL THICKNESS (INCHES)	.025
<hr/>	
CLEARANCE (INCHES)	.004
<hr/>	
Zr-4 CLAD I.D. (INCHES)	.528
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Zr-4 CLAD O.D. (INCHES)	.578
<hr/>	
WALL THICKNESS (INCHES)	.025
CLEARANCE (INCHES)	.004

Although some development effort was devoted to a number of these designs early in this reporting period, a major portion of the effort was later devoted to the three fuel assembly designs described below, which are currently being fabricated for the U-305 experiment.

a) Rib on rib design

1) Zircaloy-4 clad elements with six fins attached to each tube by electron beam welding. In this design, the tube was twisted after welding to a spiral of 90° /ft. The elements were screwed to Zircaloy-4 end plates.

2) SAP-895 clad elements with six fins extruded integral with each tube. Tubes were twisted to a spiral of 90° /ft. The elements were screwed to stainless steel end plates.

b) Rib on tube design

Zircaloy-4 clad elements with two Zircaloy-2 wires spirally wrapped and spot-welded at intervals along the length. The rods were screwed to Zircaloy-4 end plates. In this design, two wires 180° apart make two spiral turns over the length of the fuel rod. Right and left hand spirals are used so that the wire wrap of one rod bears against the wall of an adjacent rod. Some of the rods facing the pressure tube were made with a partial wire wrap which was subsequently machined to increase the bearing surface area between the fuel element assembly and the pressure tube. This design is similar to that which has been used for other U-3 experiments and CANDU.

The alternate Zircaloy-clad designs which were considered initially and later dropped from consideration included:

a) Zircaloy-4 cladding with integral spacing fins coextruded with the tubing.

b) Zircaloy-4 cladding with brazed spacing pads in lieu of spacing fins.

Since it was believed that Zircaloy-4 cladding with integrally extruded fins would provide the most economical prospect for providing spacers and hydrogen sinks for Zircaloy-clad elements, an effort was initiated for the fabrication development of ribbed tubing by extrusion. It became apparent that this development would not be available in sufficient time for the U-305 experiment and development of electron beam welded ribs was initiated as a backup. It was on this basis, that the fuel element assembly designs finally selected as a basis for the U-305 experiment contained rods with electron beam welded ribs in lieu of rods with integral extruded fins. The brazed spacing pad design was dropped from further consideration on the basis that insufficient prior data existed on its performance capabilities. In particular, some concern was exhibited over the ability of these pads to act as hydrogen sinks. This concern was based upon considerations of hydrogen diffusion paths and the behaviour of the Zr-Be braze layer with respect to hydrogen diffusion.

Three SAP designs, differing only slightly were developed for the U-305 experiment. The reference design assembly was instrumented with twelve thermocouples located in the integral SAP fins. An alternate design, which was later dropped from further consideration on the basis of development activities, provided two fins on each tube which were higher than the standard fins to provide an easier method for embedding thermocouples. The third design was similar to the first and did not provide for instrumentation.

For the specified pressure tube diameter (3.26 ID), design temperature (700°F), and design pressure (300 psig), the number of parameters that could be independently selected was limited. Therefore, it is not unusual that the assembly designs that have evolved look somewhat similar to designs which AECL has been testing in the U-3 loop. One major difference is that the fuel rods for the U-305 were filled with cored pellets of enriched UO_2 . The 0.184 in. diameter hole reduced the maximum fuel temperature from approximately 3800°F to 3000°F and also provided a volume for released fission gases.

At the early stages of fuel irradiation the external pressure, together with the hydraulic lift loads of the coolant, exceed the internal gas pressure resulting in a compressive stress in the sheath. As the fission gas builds up, the internal gas pressure exceeds the external loads, thus reversing the diametral stress in the sheath. During normal operation, the pressure of the organic coolant entering the pressure tube is 300 psig. All cladding was designed to remain free-standing (not collapse) at 300 psi differential and 750°F coolant temperature. Initially, the compression stress due to pressure is 3260 psi. After 10,000 Mwd/MTU, the estimated internal gas pressure for the highest rated UO_2 rod is 407 psia, based on 20% fission gas release. Maximum hoop stress in the SAP cladding under normal loop pressure and 10,000 Mwd/MTU is 1620 psi. For this experiment the allowable maximum working stress (transverse) for SAP-895 at 850°F and 10,000 hr is 4100 psi.

Stress analyses were performed to determine if the axial loads acting on the fuel rods would cause buckling or yielding in compression. Assuming the worst combination of tolerances and loads, it was determined that the

structure was adequate. Loads considered included hydraulic lifting forces, preload of the coil spring, differential thermal growth, and outward bowing of the rods.

The stainless steel end plates which were used on the SAP bundle and the Zircaloy plates on the Zircaloy bundle were designed to be relatively flexible so that they could deflect easily and not overstress the fuel rods under differential thermal expansion conditions. The amount of differential expansion in the Zircaloy assembly between the inner row of rods and the outer row is quite small (~ 0.006 in.) and permits fastening of all rods by cap screws to the end plates at each end. In the SAP bundle, the differential expansion is much greater (~ 0.0016 in.). As an added precaution against overstressing the outer row of tubes, the lower ends of the inner row were not fastened to the end plate. In place of the cap screws, guide screws were used which are free to slide axially in the end plate to maintain alignment without imposing axial loads on the rods.

Sufficient diametral clearance (0.004 in. cold) was provided between the cladding and fuel to prevent plastic strain of the SAP cladding. In the Zircaloy bundles, the pellet was designed to just make contact at the hot spot with the cladding if the tolerance build-up between fuel and cladding resulted in an 0.0045 in. clearance rather than the nominal 0.006 in.

In the Zircaloy-4 assembly, it was necessary to have a minimum gap of $3/16$ in. to allow for the differential thermal growth of the stack. This gap was provided as a conservative measure in the event that required dishing radii were not maintained in pellet fabrication. Furthermore, there was some concern over the matching of land width around the dish and fuel column expansion. Both of these considerations can be resolved in the

ultimate design of a Zircaloy-clad oxide element for HWOCR so that minimal axial gap between adjacent fuel columns is provided. However, to reduce the possibility of damage to the pellets during shipment, it was considered desirable to place a spring in this gap. Wave springs, washer springs, Belleville springs, and serrated springs were investigated but none appeared to be suitable for this application. As a compromise, a short (1/2 in. long) stainless steel coil spring with a spring rate of approximately 29 lb/in. was used. In assembly it was compressed 1/16 in. A thin stainless steel wafer was placed between the spring and the end cap to prevent possible damage to the spring during welding.

An evaluation of the linear power generation versus enrichment for a typical SAP-895 clad UO_2 fuel bundle in the NRU U-3 loop was carried out using the THERMOS Code. Independent calculations performed by AECL using the LPU and LPX Codes, which are based on two group diffusion theory, compared favorably with the results obtained at CE. Comparative data are shown in Figure 0312-4.

Fuel enrichments for the two UO_2 bundles designated JRA and JPA are 1.12 and 1.52 wt%. For each bundle the estimated peak power is 13.5 kw/ft, based on their respective axial location and flux distribution in the U-3 loop when operating at design rated power. Linear power generation was calculated by the following equation:

$$P \text{ (watts/cm)} = c \frac{E}{C} A \phi \sum_f \frac{\phi_f}{\phi_o}$$

where:

E is the energy per fission

C is conversion factor

A is area of fuel

ϕ is thermal flux in the U-3 test section

$\sum_f$ is macroscopic fission cross section for the fuel

$\frac{\phi_f}{\phi_o}$ is flux depression at the fuel relative to the edge of the lattice cell

Fuel temperature distribution and fission gas release calculations were made using AECL post-irradiation measurement data on the release of fission gas versus temperature. The results of this study showed that cored UO_2 pellets having an 0.184 in. diameter central hole and an 0.522 in. OD provided adequate space for fission gas and also reduced the temperature of the fuel. Calculated fission gas release for the highest rated rod operating at a peak power level of 13.5 kw/ft was 20% of the amount of fission gas produced.

Coolant and cladding surface temperatures were determined for the two UO_2 bundles using the AECL flow code. This computer program was specifically developed by AECL for thermal-hydraulic analysis of finned multi-rod clusters. It was converted from APEX-4 to FORTRAN, checked, and used to perform the thermal-hydraulic calculations. The cladding surface temperature as a function of position in the hottest rod is given in Figure 0312-5. Estimated peak cladding temperature on the inside surface was 880°F if the Zircaloy-clad UO_2 bundle was selected for test in the upper position.

Thermal expansion calculations were made for the fuel and cladding.

Because of the inability of the SAP-895 cladding to deform plastically, the fuel rods were designed with sufficient clearance between fuel and cladding to prevent cladding strain. One end of each UO_2 pellet was dished to provide room for axial expansion of the hotter central regions of the fuel. Solid pellets of depleted UO_2 were located at the ends of each fuel column to reduce **power peaking and temperatures of the end plugs**.

The HOC two-dimensional heat transfer code was used to determine the maximum end cap temperature. Based on conservative assumptions of peak power, contact conductance between end cap and depleted fuel, and peaking factors, the calculated maximum temperature of the aluminum end cap was 730°F .

The reference design contained twelve thermocouples which were embedded in selected fins of the instrumented SAP-clad UO_2 bundle JRA. These thermocouples were to indicate hot spot temperatures, bowing, temperature variation across the cladding, and perturbation of flow immediately downstream where fins on adjacent rods contact one another. Each of two maximum rated fuel rods were designed to be monitored continuously by a single pen recorder. Recording of data from all thermocouples would be performed by the REDACE system presently installed at the NRU site.

Bowing of the outer ring of rods is to be determined by temperature variations indicated by six of the thermocouples. It is expected that outward movement of the rods will restrict the flow in the sub-channels that cool the fin being measured. Three rods, 120° apart, were designed to include two thermocouples each to sense sheath temperatures at two elevations.

Three thermocouples located in adjacent fins on the same fuel rod at a given elevation were designed to indicate relative mixing between three sub-channels.

2. Ex-Reactor Tests

A 4-in. cold water test loop was constructed to simulate flow conditions through a U-305 fuel assembly. Stainless steel tubes filled with lead pellets to simulate stiffness and weight of a fuel bundle were assembled into 19-rod bundles and tested.

Vibrational characteristics were difficult to evaluate because of feedback imposed on the test section by the pump and external piping. However, relative motion between individual rods or gross motion of bundle could not be detected when operating at flow velocities up to 30 ft/sec.

Each of 18 fuel rods that make up the Zr-UO_2 bundle is securely fastened at both ends to Zircaloy-4 end plates. Thermal expansion difference between the outer ring of 12 rods and inner ring of 6 rods is resisted by the 0.062 in. thick end plates. Load deflection tests were conducted using geometrically similar plates made of stainless steel. Loads were applied to each plate through tubular supports which matched the fuel rod bolt circles. The average measured spring rate between the inner and outer bolt circles was 3450 lb/in. Adjusting for differences in stiffness between stainless steel and zirconium, the calculated maximum load is less than one pound tension on each rod in the inner ring.

3. Fabrication of Fuel Assemblies

Fabrication of oxide fuel elements and assemblies for the U-305 experiment required the development of the following:

- 1) A method for providing fuel element spacing on Zircaloy clad elements.
- 2) A method for end capping SAP-clad elements.

A decision was made to utilize Zircaloy-4 as the cladding for the Zircaloy-clad elements in the U-305 experiment. This decision was based upon results of hydrogen pickup investigations in organic coolants performed by the Canadians. It was determined that Zircaloy-4 and Zr-2.5 wt % Nb picked up substantially less hydrogen than Zircaloy-2. The latest data on hydrogen pickup in organics indicate little difference between Zircaloy-4 and Zr-2.5 wt % Nb.

The decision to use SAP 895 (10.5% Al_2O_3 , nominal) as the cladding for the SAP-oxide fuel elements was based upon its higher hoop stress rupture strength when compared to SAP products containing lower oxide contents. Furthermore, little difference in creep ductility between the 10 and 7% oxide SAP compositions at 800 and 900°F have been reported. Fortunately, an existing supply of SAP 895 from Canada was made available for use on this program.

On the above bases, the major efforts in fabrication development were devoted to the development of ribbed Zircaloy-4 tubing, the procurement of ribbed SAP 895 tubing, and the extension of developed eutectic bonding techniques to the end-capping of fuel elements containing SAP 895 cladding. Details of these efforts are summarized below:

a. Zircaloy-4 Clad Fuel Elements and Assemblies

The fuel rod assemblies which were designed for the U-305 experiment required ribbed tubing for fuel element cladding. Typical assemblies are shown in Figures 0312-2 and 0312-3. The ribs were designed to provide the following functions:

- 1) To provide the required fuel element spacing within the assembly.
- 2) To promote mixing of the coolant.
- 3) To act as hydrogen sinks.

In evaluating a number of methods of providing ribs on Zircaloy cladding, it was determined that the most favorable approach from a long-range economic standpoint for HWOCR was the extrusion of ribs integral with the tubing. Some success had been attained with this approach on the CVTR project and it was reasonable to expect that extruded ribbed tubing (most probably extruded and drawn) could be fabricated in the future at a cost almost equivalent to that for unfinned Zircaloy cladding. On this basis, a limited* development effort was initiated to produce ribbed tubing from Zircaloy-4 by direct extrusion. This program was initiated with the full realization that the probability of providing cladding fabricated in this manner for the U-305 experiment was marginal. On this basis, two other approaches to the problem of providing ribs on Zircaloy cladding were initiated and carried on in parallel with the extrusion program. The first of these approaches, was the development of electron-beam welded fins. This approach, although it was expensive for the small quantities of cladding which were involved, provided a fuel element configuration very

* This program was directed toward the U-305 experiment and had, as one of its objectives, the fabrication of ribbed tubing for U-305.

close to that of the cladding containing integrally extruded ribs. It is for this reason that this portion of the program was emphasized. The second alternate approach consisted of loading pellets and closing unfinned Zircaloy-4 fuel elements at CE and shipping these to Canadian Westinghouse for wire wrapping and final assembly. A summary of the effort, to date, on all three approaches to this problem is presented below:

(1) Extruded Zircaloy-4 Ribbed Tubing

A best efforts subcontract was awarded to the Torrance Brass Foundry of Torrance, California to provide extruded ribbed Zircaloy-4 tubing of the following description:

Tube inside diameter	0.528 in.
Wall thickness	0.025 in.
Rib height	0.035 in.
Rib width	0.040 in.
Number of ribs	Six, equally spaced
Rib twist	90°/ft

Figure 0312-6 shows the details of this ribbed geometry.

Zircaloy-4 billets for the extrusion program were purchased from the Carborundum Company to the chemistry and mechanical property requirements of Specification NL-100*. The hot forged bar stock was supplied in billets 4-1/16 in. dia, 10 in. long. These were machined to a 4-in. dia, cut to 5-in. lengths and pierced to fit the extrusion chamber. Experimental extrusion runs were made under the conditions shown in Table 0312-3.

* Available upon request.

TABLE 0312-3

EXPERIMENTAL EXTRUSION PARAMETERS

EXTRUSION MANDREL	TOOL STEEL
EXTRUSION TEMPERATURE	800 TO 1200 °F
EXTRUSION REDUCTION RATIO	30:1
EXTRUSION PRESSURE	80 TONS
EXTRUSION RAM SPEED	1 IN. PER MINUTE
EXTRUSION LUBRICANT	PROPRIETARY TO TORRANCE BRASS FOUNDRY
EXTRUDED TUBE OD	0.578 IN. NOMINAL
EXTRUDED TUBE WALL	0.038 IN. (INITIAL RUNS) 0.028 IN. (FINAL RUNS)

Since the extrusions were performed in air, (peak temp 1000°F) the tubes were vapor blasted and pickled in 35 wt%-nitric-4 wt%-hydrofluoric acid solution for 15 sec to remove approximately 0.0005 in. of metal from the tube surface.

Initial extrusions were made using a die that produced a heavy wall tube (0.038 in.) and straight fins. A typical cross section for this type of extrusion is shown in Figure 0312-7. The inside wall had a good surface finish with relatively few longitudinal die marks, while the outside surface was somewhat more irregular and exhibited longitudinal die marks. Considerable rib distortion was also observed.

Previous experience by Torrance indicated that the heavy tube wall would be required to obtain good die fill of the fins. A general "rule of thumb" which was proposed indicated that the rib height and tube wall should have similar dimensions. It was on this basis that the effort on the heavier wall Zircaloy cladding was initiated. It was planned to subsequently reduce the wall thickness to 0.025 in. by honing of the inside diameter.

Several pieces of heavy wall tubing (0.038 in.) were then honed to a target nominal wall thickness of 0.025 in. The wall thickness varied from 0.019 to 0.026 in. after honing and this eccentricity was believed to be the result of bowing in the extruded tube. The honing operation required seven hours to remove 0.015 in. of tube wall over approximately a 2-ft length. The time required for honing (seven hours to remove 0.015 in. over a 2-ft length) and the stringent requirements on tube straightness to obtain concentric honed tubing made this process impractical.

A number of tubes were gun-drilled and then given a final light honing operation to remove only 0.002 to 0.003 in. Tube wall thickness for a typical specimen from this operation varied from 0.020 to 0.026 in. Although the time

required to remove the excess material was substantially reduced, the problem of eccentricity still remained.

To determine if straight fin tubing could be extruded and then twisted several thick walled specimens were cold twisted. Cracks developed in tubes that were not annealed, however, tubes annealed for two hours at 1380°F were twisted 90°/ft without cracking.

In an effort to eliminate the need for twisting the tubes after extrusion, a die to extrude heavy wall spiral finned tubing of the proper pitch was procured. About 30 ft of tubing for evaluation purposes was produced with this die. Since the die fill of the fins appeared to be satisfactory, an attempt was made to modify the die and extrude tubing with a 0.028 in. wall thickness having spiral fins. About 20 ft of the thin wall tubing was submitted for evaluation. It has been determined that this tubing has a wall which varies from 0.0245 to 0.0270 in. in thickness.

Evaluation of the latest tubing produced on this program **had** been delayed until the fabrication of the U-305 fuel element assemblies **had** been completed. This program was relegated to one of low priority after the initial results indicated a low probability of achieving success **during this report period.** On this basis, prime emphasis for a rib-on-rib Zircaloy-clad oxide assembly was placed upon the development of electron beam welded ribs.

(2) Electron Beam Welded Fins

Electron Beam Techniques of Plainville, Connecticut was selected for the fabrication development of electron beam welded fins for Zircaloy-4 cladding. The selection of Electron Beam Techniques was based upon an evaluation and comparison of evaluation samples submitted by three potential vendors. These samples consisted of short lengths of 0.500 in. dia tubing to which 0.040 by

0.060 in. fins were welded. The qualified vendor, Electron Beam Techniques, was then required to fabricate two specimens 2 ft long, with the cross section shown in Figure 0312-8. The wire fins were made to the configuration shown in Figure 0312-9a from 0.062 in. dia Zircaloy-2 wire.

Later specimens and the production runs were fabricated using 0.052 in. dia Zircaloy-4 wire. The initial two specimens were evaluated destructively to determine whether:

- a) Fins whose cross sectional dimensions were of the same magnitude as the tube wall thickness could be properly welded to the tube wall.
- b) A finned tube could be cold twisted to a spiraled fin pitch of 90°/ft without generating cracks or distortions in the fin or tube wall.
- c) Welding parameters could be selected such that the corrosion resistance of the welded fins and tube would meet the requirements of reactor use.
- d) A heat treatment after welding would be required to restore the corrosion resistance of the welded assembly.

The Zircaloy-4 tubing supplied to Electron Beam Techniques was obtained from the Wolverine Tube Company to Specification NL-100*. The tube ID was 0.528 in. and the wall thickness 0.025 in. However, it was necessary to allow the vendor the following deviations:

- a) The "no detectable halogen requirement" would not necessarily be met because the final cleaning operation included nitric-hydrofluoric etching.
- b) The hold time at pressure on the hydrostatic tests was 10 sec instead of one minute.

* Available upon request.

c) Helium leak check* of the tubing was not performed but an ultrasonic test was performed using longitudinal and transverse OD and ID notches 0.002 in. deep and 0.020 in. long as standards.

d) No eddy current testing was performed.

e) The chemical analysis on the finished tubing included all impurities except U²³⁵. The U²³⁵ analysis was performed on ingots only.

The 0.062 in. diameter Zircaloy-2 wire and the 0.052 in. dia Zircaloy-4 wire were procured from the Carborundum Company to the applicable sections of Specifications ASTM B351-60T and NL-100, respectively. The 0.062 in. dia wire was shaved to the configuration shown in Figure 0312-9a and the 0.052 in. wire to the configuration shown in Figure 0312-9b prior to welding. The following requirements were established for electron beam welding of the wire to the tubing.

a) There was to be complete fusion across the width of the rib to the clad interface. (Full weld penetration.)

b) The maximum undercut permitted on the fin or on the tube wall was not to exceed 0.002 in.

c) The heat affected zone was not to extend to a depth greater than one-half the clad thickness.

An evaluation of the 2 ft long electron beam welded specimens obtained from Electron Beam Techniques was made prior to and after cold twisting. A visual examination at 25X magnification prior to twisting showed accurately positioned continuous welds along all fin-to-tube joints. Localized regions of deep undercutting into the fins resulted in shallow weld metal deposits on the tubing surface (Figure 0312-10a). The expelled weld metal formed irregular rippled deposits in the undercut regions of the fins (Figure 0312-10b). There

* All finished fuel elements were helium leak checked.

was no evidence of thinning of the tube wall at any of the weld joints. The tube was subsequently ball sized using a nylon ball. This was necessary because the tube inside diameter under the welded fin areas decreased by approximately 0.010 in. during welding. After ball sizing the tube was cold twisted over a mandrel to a spiral fin pitch of 90°/ft. The post twist evaluation showed shallow marks on the inside diameter of the tube at the welded fin locations.

No evidence of cracks was detected by visual and dye penetrant inspections of exterior surfaces. The inside surface of a longitudinally portion of the tubing was also free of visible cracks or other defects. Metallographic inspections of transverse sections showed full penetration welds (Figures 0312-11 and 0312-12)*. The Zircaloy wire used for the fins on the evaluation samples contained fine voids and impurities which were removed from the fusion zone during welding. It was concluded that cold twisting of a 2 ft length of welded finned Zircaloy-4 tubing to a spiraled fin pitch of 90°/ft did not have any harmful effects on the structure of the tubing.

Following this evaluation the tube was corrosion tested in steam for 3 days at 750°F and 300 psi. A duplicate tube fabricated in the same manner and from the same materials was subjected to a 4-hr, 1200°F heat treatment and also placed in corrosion test. A post corrosion test inspection showed a hard, continuous, lustrous, black film over the entire surface of the heat treated tube. Inspection of the unannealed tube under 25X magnification showed small spots of grayish or nonlustrous black surfaces on the wire and in the weld area. Thus, the 1200°F heat treatment was used for ribbed tubing for the U-305 experiment.

Based on the above evaluation, Electron Beam Techniques was instructed to

*The crack in the wire surface noted in this evaluation sample is not typical of the condition of the wires used for the rods fabricated for irradiation testing.

proceed with the production of 100 ribbed tubes utilizing Zircaloy-4 tubing provided by Wolverine and better quality spacer wire. By the end of this report period, 30 tubes had been delivered. Each of these has been annealed. Sizing and twisting operations are now in progress. Prior to fabrication of fuel elements from these tubes an extensive evaluation of the tubing is planned. Dimensional tolerances will be checked closely as will metallurgical structure and mechanical strength.

(3) Wire Wrapped Fins

To provide backup* fuel bundles for the U-305 experiment, 60 smooth-walled Zircaloy-4 tubes were loaded with pellets and end capped. These were then shipped to Canadian Westinghouse, Port Hope, Ontario for wire wrapping, final assembly, and corrosion testing. The wire wrap consisted of two 0.062 in. dia Zircaloy-2 wires spaced 180° apart on the fuel tube and spiraled to a pitch of 180°/ft. The cross section of this fuel tube is shown in Figure 0312-13. The wire is spot welded to the tube at one-half in. intervals.

The configuration of an assembly using wire wrapped spacers differs from the electron beam welded bundle in that the wire rib bears on the tube wall of the adjacent fuel rod whereas in the electron beam welded configuration the rib bears against the rib of an adjacent fuel tube. Two fuel bundles containing 1.52 wt% U²³⁵ fuel and one bundle containing 1.12 wt% U²³⁵ fuel are to be fabricated in this manner.

The 0.528 in. ID by 0.025 in. wall Zircaloy tubing for these fuel bundles was part of the same lot of tubing procured from the Wolverine Tube Company to Specification NL-100** for the electron beam welded finned tubing. The 0.062

* It has been recently decided to include one of these assemblies as part of the U-305 fuel string.

** Available upon request.

in. dia Zircaloy-2 wire was procured from the Carborundum Company to the chemical and mechanical property requirements of Specification NL-100. Zircaloy-2 wire was used because Zircaloy-4 wire was not available. The performance of the fuel element should not be appreciably affected by the substitution of the Zircaloy-2 in place of the Zircaloy-4 wire. In fact, Zircaloy-2 ribs may act as a more efficient hydrogen sink than those fabricated with Zircaloy-4.

The wire wrapping operation has been initiated at the vendor's plant. Typical specimens of tubing with wire wrapping have been received from the vendor and are being evaluated prior to initiation of work on the 60 tubes. It is anticipated that this effort will be completed early in July with the delivery of three completed fuel assemblies to Chalk River.

b. SAP 895 Clad Fuel Assemblies

The SAP 895 clad fuel assemblies are very similar in geometry to the Zircaloy-4 clad assemblies. Typical details of the assembly are shown in Figures 0312-2 and 0312-3.

Tubing to Specification NL-106* was extruded from 60 mm (2.370 in.) billet material at Nuclear Metals, Inc. and at Torrance Brass Foundry. The billets were sorted for use on the basis of ultrasonic inspection performed at the Magnaflux Corporation in East Hartford, Connecticut. A contact transducer was used in a transverse direction. The instrument was calibrated so that a 3/64 in. dia hole gave a full scale indication. Any indication with a signal-to-noise ratio greater than 2:1 was cause for rejection. The billets were cored and then extruded over a mandrel to a straight six fin configuration. Tubes were subsequently ball sized to the finished inside diameter and twisted to the necessary 90°/ft.

* Available upon request.

Upon receipt, the tubing was dimensionally inspected and eddy current tested. The dimensional inspection indicated that the as-extruded tubing from both vendors was uniform. The inside diameter varied from 0.5253 in. to 0.5270 in. The wall thickness was in general within the specified 0.025 $\pm$ 0.003 in., but five tubes showed variations between 0.0205 to 0.0290 in. This may have resulted from misalignment of the mandrel during the extrusion. The twist angle of the fins ran from 91 to 97°/ft.

Hydrostatic tests were conducted at 1500 psi for five minutes. A total of 42 tubes received from Torrance Brass were tested in 12 ft lengths. Of these, three had leaks. A second vendor's tubing was tested in 6 ft lengths and of the 14 tested, no leaks were found. The leaks took the form of seamlike splits in the longitudinal direction. The wall was generally very thin near the splits.

Eddy current inspection was conducted at the Budd Company in Phoenixville, Pennsylvania. The tubing was passed through a pair of external coils which compared the material inside both coils. A defect on either section would unbalance the circuit and this was recorded on a strip chart. The standard was fabricated from a sample of SAP tubing using dents, slots and holes of various sizes down to a Number 80 drilled hole (0.0135 in. in dia). A study to correlate the type of indication to the type defect has not been completed. Consequently, it was necessary to reject any tubing which showed indications. Those areas which showed indications were cut out, leaving approximately 70% of the original tubing supply for subsequent dye penetrant inspection.

The dye penetrant test revealed many pinhole-like indications. The exact nature of these indications are under evaluation. However, it was determined that these were not superficial indications. Repeated polishing and dye checking

until 0.002 to 0.004 in. of material had been removed did not eliminate these indications. Tubing which showed no indications and tubing where the indications were confined to the fins was accepted for further processing for the U-305 experiment.

The original procedure called for dye penetrant inspections on tubing which was rough cut to fuel tube lengths (approximately 27 in.). Since better than half the tubing sections were being rejected with isolated pitlike indications, the inspection step was changed to precede the cutting operation. This permitted the selection of defect free areas between indications from the 10 and 6 ft lengths.

The tubing which passed eddy current, dye penetrant and hydrostatic testing was cut to length and processed for fuel rods. The first operation was to mill the fins off each end in preparation for end plug bonding. (A summary of eutectic bonding development is presented in the next Section.)

The end plugs were made from 1100 aluminum and were plated with 0.0001 in. thick silver just prior to bonding. The eutectic bonding process described in the next Section was used to bond end plugs. The parameters used were 1170 to 1180°F for one minute with an applied force that produced a minimum of 0.009 in. decrease in the tube diameter. Of the 61 tubes which were processed through the application of the first end plug, six were found to leak helium and were therefore rejected.

The balance of the process to complete the rod fabrication involves:

- 1) Loading the tubes with fuel.
- 2) Bonding the top end plug.
- 3) Machining the end plugs for final rod length and tapping holes for assembly.

- 4) Thermal cycling between 350 and 850°F.
- 5) Inspecting for dimensions, helium leaks, dye penetrant and radiography.

Upon completion of the above steps, 18 fuel rods and one spacer rod will be assembled into a bundle early in the next quarter.

c. Eutectic Bonding, SAP-Clad Elements

In order to fabricate SAP-clad assemblies a method for producing end closures on SAP clad fuel elements was necessary. This procedure is difficult because conventional welding techniques are ineffective for end closures on SAP fuel elements. Fusion of the SAP causes the oxides to segregate and leave gross oxide free areas which subsequently behave as pure aluminum. Other joining methods have been developed for this purpose. The Canadians have used a pressure bonding technique in which a 65S aluminum shim is inserted between the SAP components. They also used magnetic-force-resistance welding* to bond SAP end plugs to SAP cladding. Other methods based on the formation of an eutectic between an aluminum end plug and Ag, Cu, Sn, Zn or Si have been investigated. These eutectics melt at temperatures at least 100°F below the melting point of aluminum. The formation of the eutectic affects bonding without extensive fusion of the part to be bonded. **It has been shown that silver** provides a superior bond because it diffuses rapidly and yields a bond free of second phases. The eutectic bonding operation uses a silver plated aluminum end plug which is compressed within the SAP tube; heated above the Ag-Al eutectic temperature (1036°F), and then cooled. Silver eutectic bonding was selected at the onset of the program because specific information concerning this process

* This approach offers the greatest potential for achieving a design with thin end caps and minimum fuel-to-fuel axial spacing in adjacent fuel assemblies. This method was not used in the U-305 experiment on the basis of long lead times for equipment and the need for subsequent development time. It was believed that eutectic bonding, already used by AI for OMRE experiments, could be adapted in minimal time.

was available **and it was** believed that less time would be required to adopt this process for the SAP-clad oxide elements.

A bonding device, shown in Figure 0312-14, was constructed to permit the selection of the optimum conditions of time, temperature, and deformation to produce closures of SAP 895-clad-UO₂ elements. This device employs a split die with one end fixed and the other end spring loaded against the tube assembly to provide the required compression. Early investigations of bonding parameters were completely unsuccessful. A black deposit was observed on the copper coils inside the glove box and a blackening of the silver plated aluminum end cap was also found. This phenomenon was traced to the presence of sulphur in rubber gloves used in the access ports of the box. New sulphur and chlorine free gloves were installed and no additional blackening of the silver plated aluminum plugs was observed. However, poor bonding was still obtained in spite of a very clean helium atmosphere in the glove box.

The end plugs were silver plated commercially at a local vendor's facility and several hours elapsed before loading into the helium atmosphere of the bonding box. It was found that oxidation of the aluminum surface could proceed at room temperature through the thin layer of silver. The oxidized surface prevented the formation of a metallographic bond or produced a bond with a continuous network of second phase like that shown in Figure 0312-15.

A silver plating process was established **in which the elapse time between** plating and loading into an inert atmosphere was reduced to a few minutes. In addition, the wire brushing technique used to remove the oxide from the inside of the SAP tubing prior to bonding was scheduled such that the amount of air exposure between brushing and loading into the bonding box was limited. As a result of these changes excellent metallurgical bonding was obtained.

A typical bond is shown in Figure 0312-16. No continuous second phase is present in this bond region and the interface is only visible because of the different etching characteristics of the SAP and the 1100 aluminum alloy.

The effect of the amount of silver deposited on the end plugs was also investigated. No effect was observed on the quality of the bond when the coating thickness was varied between 0.00005 and 0.0005 in. A deposit thickness of 0.0001 in. was arbitrarily selected for use during fabrication.

The temperature, time at temperature and amount of compression to produce bonding were evaluated. In order to produce a uniform circumferential closure, the SAP tubing had to be plastically deformed in a uniform manner onto the plug. A minimum deformation of 0.009 in. measured as the change in tubing outside diameter (0.578 in.) was found to be necessary for good bonding. Figure 0312-17 shows a large region of open area between an end plug and tube caused by non-uniform application of pressure and insufficient deformation of the assembly during bonding.

Eutectic bonding of a well-mated surface can theoretically be obtained at a temperature as low as 1036°F, the Al-Ag eutectic temperature. Bonding was attempted at temperature as low as 1100°F, but only partial bonding resulted. At temperature above 1190°F, the softening of the aluminum was excessive. Although good bonding could be obtained, the structural integrity of the plug could not be retained. The optimum temperature was found to be between 1170 and 1180°F. A minimum of one minute holding time was necessary to produce bonding and to allow the silver to diffuse into the aluminum at these temperatures. Longer time up to 15 minutes were investigated but no improvement in bonding resulted.

The eutectic bonding process which is currently being used contains the following steps:

1) The tubes are prepared by milling the fins off each end for a length slightly greater than the 1/2-in. bond length. The inside of the tubes are wire brushed to remove burrs and oxide film.

2) The end plugs (1100 aluminum) are silver plated to a thickness of 0.0001 in.

3) The end plugs and tubes are brought into a vacuum dry box which has been evacuated to 3×10^{-4} mm Hg and backfilled with dry helium (dew point below -70°F).

4) Each tube is again wire brushed inside the dry box just prior to plug insertion.

5) The plug is inserted and a split stainless steel sleeve is placed over the tube in the area of the bond.

6) The assembly is placed between the die halves and a preset pressure is applied by tightening the screw against the belleville spring stack.

7) The induction coils are energized and the temperature of the end plug is brought to 1170°F in approximately five minutes. The temperature is held between 1170 and 1180°F for one minute. The power is turned off and the system permitted to cool.

The evaluation of bond quality involved four types of tests. Metallographic sectioning in both the longitudinal and transverse directions provided information on the metallurgical quality of the bond and the percentage of bond. Complete bonding was a requirement of the product. A "peel test" was used to provide information on the strength of the bonded area. This test involved chiseling the tube away from the plug to separate the members. If the sample components could be separated a weak bond or nonbond was evident. A third

test involved closing both ends of a 10-in. tube section and pressurizing to failure from the inside. A failure in the bond prior to failure in the tube indicated poor bond quality.

The fourth, and most critical, method of bond evaluation was a helium leak test. Bonds were tested with one atmosphere pressure of helium on one side and a vacuum on the other. Any indications of helium above background, using a Veeco Leak Detector calibrated to 2×10^{-10} cc/sec, was taken as an indication of poor bond quality.

At the end of this investigation end plugs could be routinely bonded to the SAP 895 fuel tubes. However, cycle times were such that only one bond could be produced, using a single bonding unit, per hour. Attempts to improve this rate were not successful.

Fabrication of SAP-clad fuel elements for the U-305 experiment is proceeding on the basis of eutectic bonded end caps.

d. Uranium Dioxide Pellets

Uranium dioxide pellets were purchased from Nuclear Fuel Services, Inc. The required configurations for the three types of pellets involved are shown in Figure 0312-18. The enriched pellets were cored to provide a gas void along the central axia of the pellet column. One end of each pellet was dished to provide space for thermal expansion of the pellet. The pellet lengths were different for each enrichment to prevent any possibility of mixing pellets of different enrichment. The depleted pellets are solid and are to be placed at each end of a pellet stack to act as thermal insulators. The pellets were procured to Specification NL-102*, "Ceramic Fuel Compacts." At the request of the vendor the following exceptions to the specification were allowed.

* Available upon request.

<u>Specification</u>	<u>Item</u>	<u>Exception</u>
NL-102	2.0	Vendor permitted to use 1% PVA 1% Carbowax binder.
NL-102	2.1	Vendor permitted a maximum iron impurity level of 150 ppm in place of a 50 ppm maximum.
NL-102	2.1	The definition of a blended batch of UO_2 was left to the definition of the vendor. Standard quality control procedures were accepted.
NL-102	2.3	A drying vacuum of 10 mm Hg instead of 0.1 mm Hg is permitted. The maximum permissible weight loss shall be 40 instead of 30 ppm.
NL-102	2.4	The enrichment of the fuel will be certified by the vendor based on the AEC isotopic analysis of the UF_6 . The vendor will not be required to obtain an isotopic analysis on the finished product.
NM-1	9	All uranium in scrap recovery will be returned to the AEC for credit within 90 days after last shipment of pellets to CEND.

Three lots of pellets were received from Nuclear Fuel Services. The depleted pellet lot conformed to the specification and design tolerances except for the land area width at the dished end of the pellet which measured

approximately 0.0025 in. below the specification minimum limit of 0.045 in. The 1.12 wt% U^{235} and the 1.52 wt% U^{235} pellet lots varied widely in quality. The most common defects were hourglassing, nonuniform land width at the dished end of the pellet, end face surfaces nonparallel by more than 0.002 in. and rough land surface areas. To insure that pellets of a uniform quality were used in the fuel elements for the U-305 experiment, the plant quality control sampling procedure for pellet inspection was abandoned and all of the pellets to be used were subjected to a 100% dimensional and visual inspection. All pellets with significant hourglassing (dimensionally out of specification) and chipping in excess of that permitted by visual standards were rejected. Furthermore, it was decided that the quality of the accepted pellets could be substantially improved with respect to land surface width uniformity, end face roughness, and end face parallelism by end face grinding. Therefore, an end face grinding operation was established at CE and both ends of every pellet were ground.

The finished pellet length was calculated to provide a stack containing an integral number of identical pellets. The nominal lengths for the 1.52 wt% U^{235} and the 1.12 wt% U^{235} pellets were 0.500 and 0.600 in., respectively. The required enriched pellet stack lengths for the Zircaloy clad and the SAP clad fuel rods were 22.125 and 22.375 in. Pellets were ground to three different pellet lengths as shown in Table 0312-4.

A number of pellets were ground to establish the optimum amount of UO_2 removal to retain a practical land width on the dished end, and yet assure grinding of the entire land face area. This required removal of a minimum of 0.0025 in. in pellet length. Because the incoming pellets varied in length over a range of about 0.025 in., it was necessary to grind the non-dished end first to establish a uniform length. The second grinding on the dished end of the pellet then removed 0.0025 in. to attain the final pellet length. After grinding, a chamfering operation was used to break

TABLE 0312-4

FUEL BUNDLE PELLET LENGTH DIMENSIONS

ENRICHMENT	ENRICHED STACK LENGTH (INCHES)	CLAD MATERIAL	ENRICHED NO. OF PELLETS IN STACK	PELLET LENGTH (INCHES)
1.52	22.125	ZIRCALOY-4	46	0.4810
1.12	22.125	ZIRCALOY-4	38	0.5822
1.12	22.375	SAP 895	38	0.5888

the pellet edges at a 45° angle for a distance of 0.005 to 0.010 in.

Special laboratory equipment for grinding sample lots was modified to handle the relatively large quantity of pellets required for this program. The equipment consists of a semi-automatic grinder using a diamond impregnated wheel. The pellet is held in a rotating chuck and the grinding wheel advanced to remove material from the end face of the pellet until a predetermined length is reached. Except for pellet insertion and removal, all other grinding sequence operations were initiated by a single cycle-start pushbutton. A throughput rate of approximately 100 pellets per hour, for the grinding operation, was established. Figure 0312-19 shows the grinding setup.

Chamfering of each end of the pellet was performed on a modified instrument lathe. The pellet was held in a collet mounted in a sliding tailstock. The tailstock assembly was then advanced and rotated against a diamond wheel located within a contamination barrier. Although a minor amount of chipping occurred during the chamfering operation there was no difficulty in holding the length of the chamfer within the 0.005 to 0.010 in. length. Figure 0312-20 shows the pellet chamfering setup.

A total of 9,300 pellets (165 kg of UO_2) were ground, 5,225 of the 1.12 wt% U^{235} enrichment, and 4,075 of the 1.52 wt% U^{235} enrichment. A total of 4.7 kg of UO_2 was removed from the total pellet lot as a very fine UO_2 powder. The flow of cooling water over the pellets during grinding carried the grinding scrap UO_2 powder to a settling tank where the UO_2 settled as a sludge. The cooling water was changed and the UO_2 sludge removed from the settling tank every four to five hours of grinding.

Subsequent to grinding, the pellets were rinsed in demineralized water,

dried in an air oven at 165°F and submitted to a 100% visual inspection. Diametral measurements were made at random on 5% of each day's production.

A complete chemical and metallographic characterization is being made of each pellet enrichment lot. The chemistry of three sets of pellets from each lot are given in Table 0312-5.

Set 1 - As-received pellets

Set 2 - Pellets after grinding and air drying at 150°F

Set 3 - Same as Set 2 with an additional vacuum furnace drying operation at 600°F

In addition to the tests shown, hydrogen analyses, an enrichment determination and spectrographic determination of trace impurities are currently in process, but results are not yet available.

For the components analyzed, the chemistry evaluation confirmed that the UO₂ pellets are in conformance with the chemistry requirements of Specification NL-102*. Comparison of data from Set Nos. 1, 2 and 3 show that the pellets are not contaminated or oxidized during the grinding and drying operations.

Metallographic examination was performed on both the 1.12 wt% U²³⁵ and the 1.52 wt% U²³⁵ enrichment lots of pellets under 25X magnification. A typical pellet cross section is shown in Figure 0312-21. Local areas of very high density and also a dispersion of small voids or pores were noted in some of the 1.12 wt% U²³⁵ pellets. These were tentatively identified as stainless steel chips. However, total iron determined for pellets with relatively large numbers of inclusions was still within the 150 ppm limit as specified.

* Available upon request.

TABLE 0312-5
CHEMISTRY EVALUATION OF UO₂ PELLETS

ANALYSIS	SET NO.	LOT NO. I (1.52 wt% U ²³⁵)	LOT NO. II (1.12 wt% U ²³⁵)
DENSITY	1	10.55 gms/cc	10.53 gm/cc
	2	10.55	---
	3	10.57	10.54
O/U RATIO	1	2.010	2.010
	2	2.010	---
	3	2.011	2.012
FLOURINE	1	5 ppm	5 ppm
	2	5 ppm	5 ppm
	3	5 ppm	5 ppm
NITROGEN	1	12 ppm	12 ppm
	2	23 ppm	---
	3	8 ppm	12 ppm
CARBON	1	50 ppm	125 ppm
	2	35 ppm	95 ppm
	3	40 ppm	100 ppm
TOTAL URANIUM	1	88.13%	88.06%
	2	87.65%	---
	3	87.47%	88.00%
IRON	1	100 ppm	105 ppm
	2	157 ppm	---
	3	108 ppm	95 ppm

B. FLOW TESTING OF FUEL ASSEMBLIES

Design of a 4-in. water test facility was initiated to investigate fuel rod vibration and coolant mixing between the various sub-channels in a typical fuel element. Operating characteristics of the loop are:

Flow Capacity, Total (gpm)	800
Pump ΔP (psi)	200
Maximum Operating Temperature ($^{\circ}F$)	200
Number of Test Sections	2
Test Section Length (ft)	18
Material of Construction	304 S.S.

Operating criteria is based on simulating with heated water the organic cooled reactor flow conditions and Reynolds number with one test section in operation at any given time. The calculated Reynolds number using Santowax OMP with 10% HB at $680^{\circ}F$ is equivalent to that of $150^{\circ}F$ water for equal flow rates and identical test configurations.

It is presently contemplated that coolant mixing will be determined by selectively heating several rods in the cluster and measuring the change in downstream temperature distribution.

III. EVALUATION OF EFFORT TO DATE

A. Several rod assembly designs have been developed which will simulate HWO CR irradiation and environmental conditions in the U-3 loop at NRU. These all contain 18 fuel elements surrounding a central support rod.

B. As a result of fabrication programs completed or nearing completion, the U-305 experiment can include any of the following configurations:

- 1) Zircaloy-4 clad elements with spacer ribs attached to the rods by electron beam welding.
- 2) Zircaloy-4 clad elements with wire wrapped spacers.
- 3) SAP-895 clad elements with extruded ribs.

C. Initial evaluation of thin wall Zircaloy-4 tubing indicates that direct extrusion of tubing with integral ribs may be a commercially feasible process. Additional development with emphasis on extrusion plus drawing is certainly warranted.

D. Electron beam welding appears to be a useful and reliable method for the attachment of ribs to Zircaloy-4 fuel cladding. This method provides a good substitute for integrally extruded ribs until the extrusion process is adequately developed. There is some indication that large cost reductions can be achieved for electron beam welded ribs if production quantities are involved. The exact nature of cost reductions should be evaluated in light of their influence on total fuel cycle costs.

E. SAP-895 obtained from Canada can be extruded readily as integral ribbed fuel clad tubing. Yields of two foot lengths based on eddy current and dye penetrant tests were low.

F. Cored and dished UO_2 pellets of satisfactory chemistry and density can be obtained commercially. As with solid pellets, repair by end grinding may

frequently be required.

G. Eutectic bonding of aluminum end plugs in SAP-895 cladding is practical for the fabrication of elements for irradiation or laboratory tests. The process does not appear to be a practical production technique for full core loadings. Furthermore, long end caps are required. Magnetic force welding permits the use of thin end caps and is a method amenable to production rates. This method for end closures should be pursued.

IV. NEXT REPORT PERIOD ACTIVITIES*

A. Components now on hand for the U-305 fuel assemblies will be completely evaluated and the assemblies will be fabricated and inspected. Completion of these assemblies is anticipated prior to July 19.

B. A decision will be made as to which of the SAP-895 and Zircaloy-4 clad bundles will be part of the U-305 experiment. The experiment will then be inserted into the reactor prior to the startup date during the last week of July.

C. Continued analysis of the performance of the U-305 experiment will be a major effort during the balance of the next quarter.

D. Extruded tubing of Zircaloy-4 with integral ribs will be evaluated extensively. Metallography and micro-hardness as well as corrosion testing will be used to determine the extent of atmosphere contamination occurring during the extrusion of the tubing in air. Burst tests will be used to determine the mechanical properties of the tubing at room temperature and 800°F. Modifications introduced in both structure and strength by heat treatment will also be studied. Based upon the above evaluations the vendor will be directed to produce tubing of the most promising type for further testing and possible use in WR-1 fuel irradiation experiments.

E. Nuclear, thermal-hydraulic, and mechanical design requirements will be established for the WR-1 irradiation tests. Design of hydraulic flow test models to measure pressure drop, rod deflection and vibration will be initiated. The hydraulic tests will be performed using hot water at conditions which dynamically simulate high velocity flow of organic coolant at design reference temperatures.

F. Flow distribution and mixing tests which simulate the reference fuel assembly will be designed. The effect of rod stiffness, flow rate and the most promising spacer plate designs are to be tested and evaluated.

G. Work on UO_2 -SAP reactions, ridging and radial ratcheting, thermal conductance, and defect cladding will be initiated.

* As a result of a recent change in program emphasis, work during the next report period will concentrate on the U-305 experiment. Some additional work on Zircaloy-clad oxide development and new work on carbide development may be initiated as part of this task.

FIG 0312-1. U-305 EXPERIMENT

SAP CLAD UO_2
FUEL ROD
(INTEGRAL FINNS)

ZR-4 CLAD UO_2
FUEL ROD
(WELDED WIRE)

ZR-4 CLAD UO_2
FUEL ROD
(E. B. WELDED WIRE)

FIG 0312-2. CROSS SECTION OF TYPICAL UO_2 BUNDLE

FIG 0312-3 TYPICAL UO_2 FUEL BUNDLE ASSEMBLY

FIG 0312-4. RELATIVE POWER GENERATION VS ENRICHMENT

FIG 0312-5. CLAD SURFACE TEMPERATURE VS RELATIVE AXIAL POSITION FOR UO_2 BUNDLE JRA

FIG 0312-6. UO_2 FUEL ROD ASSEMBLY EXTRUDED FINNED ZIRCALOY-4 CLAD

photo 3460

FIG 0312-7. CROSS SECTION OF EXTRUDED RIBBED ZIRCALOY-4 TUBING

FIG 0312-8. UO₂ FUEL ROD ASSEMBLY ELECTRON BEAM WELDED
FINNED ZIRCALOY-4 CLAD

FIG 0312-9a. ELECTRON BEAM WELDED WIRE
QUALIFICATION CONFIGURATION

FIG 0312-9b. ELECTRON BEAM WELDED WIRE
FABRICATION CONFIGURATION

photo 3264

FIG 0312-10a

photo 3266

FIG 0312-10b

UNDERCUTTING OF RIB BY ELECTRON BEAM WELDING

photo 3272

FIG 0312-11. CROSS SECTION OF TUBE WITH ELECTRON BEAM WELDED RIBS

FIG 0312-12. ELECTRON BEAM WELDED ZIRCALLOY-4
RIB SHOWING WELD ZONE

FIG 0312-13. UO_2 FUEL ROD ASSEMBLY WIRE WRAPPED ZIRCALOY-4 CLAD

FIG 0312 - 14. EUTECTIC BONDING FIXTURE USING SPLIT DIES

MAG 500X

BONDING TEMPERATURE 1175°F
BONDING TIME 15 MINUTES

FIG 0312-15. BOND MADE AFTER PROLONGED EXPOSURE OF
PLATED PLUG TO AIR

MAG 500X

BONDING TEMPERATURE 1170°F
BONDING TIME 15 MIN

FIG 0312-16. BOND MADE WITH PLATED PLUG EXPOSED LESS THAN TEN MINUTES TO AIR PRIOR TO PLATING

MAG 5.5X

FIG 0312-17. POOR BOND RESULTING FROM NON-UNIFORM APPLICATION OF PRESSURE AROUND THE CIRCUMFERENCE OF THE TUBE

DEPLETED

ENRICHED

FIG 0312-18. DEPLETED AND ENRICHED PELLET CONFIGURATIONS

photo 3412

FIG 0312-19. PELLET END FACE GRINDING

photo 3411

FIG 0312-20. SETUP FOR CHAMFERING OF UO_2 PELLETS

FIG 0312-21. ENRICHED PELLET STRUCTURE

HWO CR QUARTERLY REPORT

TITLE: Fabrication and Irradiation of UC Fuel Elements SUBTASK NO.: 0351

PROJECT ENGINEER: S. O. Arneson REPORTING PERIOD: Inception through June 1965

I. PROJECT OBJECTIVES

The objectives of this task are to design, fabricate, and irradiate uranium carbide fuel elements which will operate at the power ratings and achieve the burnup levels required for the HWO CR Program. These tests are aimed at establishing the performance capabilities of HWO CR uranium carbide fuel elements with particular emphasis on irradiation stability, structural integrity of the fuel element configuration, in-pile materials compatibility, and thermal performance of the fuel configuration.

The initial irradiation test is to be conducted in the organic-cooled U-3 loop of the NRU reactor in Canada beginning in July 1965. This experiment is aimed at testing SAP-clad uranium carbide fuel rod clusters at cladding temperatures near 850°F and burnups of at least 10,000 Mwd/MTU.

II. PROGRESS DURING REPORT PERIOD

Since January 1965, the major accomplishment of this project has been the design, analysis, procurement, fabrication, assembly, quality control, and documentation of the first major irradiation experiment of the HWO CR Program. The experiment, designed for irradiation in the organic-cooled U-3 loop of the NRU reactor, is a joint AI-CE assembly designated Exp-NRU-305 (U-305). It consists of an 8-ft string of five 19-rod fuel bundles, 3.25 in. in diameter. The lower two bundles, provided by Combustion Engineering, are each about 25.5 in. long and consist of Zircaloy-4 clad UO_2 . The top three bundles, provided by Atomics International,

are each approximately 16-5/8 in. long and are comprised of SAP-clad UC. Completion of the carbide fuel bundles for this experiment and two alternate or backup strings has been the major work of this project.

Most of the effort during the period was directed toward the construction of the Exp-NRU-305 (U-305) experiment. This effort was begun about mid-January with a parametric study to determine the optimum combinations of fuel rod size, number and spacing, fin number and size, and flow rate to produce a high specific power rating, a total loop power of not more than 2 Mw, peak cladding surface temperatures of about 850°F, and an outlet coolant temperature no greater than 750°F. Other considerations included the desire to approach an HWOCR configuration where (a) parasitic neutron absorption is kept low by reducing coolant, cladding, and other structural materials in the element to a minimum, and (b) rod diameters are large to reduce the total number of rods and thus the total fuel fabrication costs.

The design chosen for this first irradiation experiment is illustrated in Figure 0351-1; the overall performance characteristics are listed in Table 0351-1. The three carbide bundles in the joint UC-UO₂ experiment are comprised of 18 fuel rods in a 19-rod circular array. The central rod position is occupied by a structural rod which supports the stack of fuel bundles. The fuel is slightly hyperstoichiometric UC with a carbon content of 5.0 $\pm$ 0.2%. The enrichment of the uranium in the UC fuel is 1.5% U²³⁵. The fuel slugs are cast and ground right circular cylinders of 0.499 in. diameter, 1 to 6 in. long, stacked as necessary to yield a 13.6-in. fuel column. The loading plan is given in Table 0351-2. The fuel columns are loaded in extruded SAP cladding with a 0.030-in. wall thickness and 14 fins 0.033 in. high by 0.035 in. wide. The fins are spiralled at a twist of 90°/ft to promote interchannel mixing and to

TABLE 0351-1
U-305 EXPERIMENT DATA SHEET

Nuclear Data		Thermal-Hydraulic Data	
1) Fuel Enrichment (wt %)		1) Peak Rod Power Rating [kw/ft (Q/4π w/cm)]	
UC	1.50	UC	20.8 (54.3)
SAP-clad UO ₂	1.52	UO ₂	13.5 (35.3)
Zr-clad UO ₂	1.12	2) Peak Fuel Centerline Temperature [°F (°C)]	
2) Initial Peak Burnup Rate (Mwd/MTU-day) (191 Mev/fission)		UC	~2000 (~1092)
UC		UO ₂	~3000 (~1650)
Outer Ring	46.8	3) Peak Cladding Temperature [°F (°C)]	
Inner Ring	32.8	UC	~850 (~454)
SAP-clad UO ₂		SAP-clad UO ₂	~790 (~421)
Outer Ring	39.5	Zr-clad UO ₂	~860 (~460)
Inner Ring	27.9	4) Peak Cladding Heat Flux [Btu/hr-ft ² (w/cm ²)	
Zr-clad UO ₂		UC	326,000 (103)
Outer Ring	39.5	SAP-clad UO ₂	246,000 (77)
Inner Ring	26.1	Zr-clad UO ₂	304,000 (96)
3) Peak Fission Rate (fissions/cc-sec)		5) Burnout Safety Factor	
UC		UC	
Outer Ring	1.94 x 10 ¹³	Normal Operating Conditions	3.96
Inner Ring	1.24 x 10 ¹³	Incipient Reactor Trip	2.94
SAP-clad UO ₂		UO ₂	
Outer Ring	1.31 x 10 ¹³	Normal Operating Conditions	4.25
Inner Ring	0.92 x 10 ¹³	Incipient Reactor Trip	3.72
Zr-clad UO ₂		6) Loop Inlet Temperature [°F (°C)]	620 (326)
Outer Ring	1.31 x 10 ¹³	7) Loop ΔT [°F (°C)]	82 (45)
Inner Ring	0.87 x 10 ¹³	8) Loop Outlet Temperature [°F (°C)]	702 (372)
4) Thermal Neutron Flux (neutrons/cm ² -sec)		9) Experiment Power Output (Mw)	
Unperturbed, Peak	24.0 x 10 ¹³	Total	1.66
UC		UC	0.92
Outer Ring, Average	9.9 x 10 ¹³	UO ₂	0.74
Inner Ring, Average	6.8 x 10 ¹³	10) Flow Velocity [fps (m/sec)]	30 (9.14)
SAP-clad UO ₂		11) Heat Transfer Coefficient [Btu/hr-ft ² -°F (w/cm ² -°C)]	2030 (1.15)
Outer Ring, Average	9.9 x 10 ¹³	12) Coolant	Santowax OM + 25% HB
Inner Ring, Average	7.0 x 10 ¹³	13) Loop Flow Rate	
Zr-clad UO ₂		(US gpm)	306
Outer Ring, Average	7.3 x 10 ¹³	(lb/hr)	1.343 x 10 ⁵
Inner Ring, Average	4.8 x 10 ¹³	(kg/sec)	16.9
5) Fuel Weights (g)		14) Loop Inlet Pressure [psig (kgf/cm ²)]	300 (21.1)
UC		15) Test Section ΔP [psi (kgf/cm ²)]	80 (5.6)
Total UC	31,600	16) Loop Outlet Pressure [psig (kgf/cm ²)]	220 (15.1)
Total U	30,200		
U ²³⁵	454		
UO ₂			
Total UO ₂	28,000		
Total U	24,700		
U ²³⁵	321		
6) Peak Specific Power (kw/kg U) (191 Mev/fission)			
UC	46.8		
UO ₂	39.5		

TABLE 0351-2
U-305 EXPERIMENT - URANIUM CARBIDE FUEL LOADING PLAN

Bundle Identification	Element Cladding Material	Fuel Column Length (in.)	No. of Elements Per Fuel Compositional Range	Element Identification	Fuel Compositional Range (wt % C)
LRA	SAP	13.6	3	B, C, D	4.82 to 4.92
			3	E, F, H	4.92 to 5.0
			3	J, K, L	5.0 to 5.1
			3	M, N, P	5.1 to 5.2
			6	R, S, T, W, X, Y	4.92 to 5.2
LSA	SAP	13.6	3	B, C, D	4.82 to 4.92
			3	E, F, H	4.92 to 5.0
			3	J, K, L	5.0 to 5.1
			3	M, N, P	5.1 to 5.2
			6	R, S, T, W, X, Y	4.92 to 5.2
LTA	SAP	13.6	3	B, C, D	4.82 to 4.92
			3	E, F, H	4.92 to 5.0
			3	J, K, L	5.0 to 5.1
			3	M, N, P	5.1 to 5.2
			6	R, S, T, W, X, Y	4.92 to 5.2

space the rods. End plugs are silver-eutectic bonded to the tubes, and the individual rods are fastened between two spider-like end plates to form a 16-5/8 in. bundle, 3.25 in. in diameter. The bundle locations are shown in Table 0351-3.

Setting the clearance between the fuel and cladding is dictated principally by two considerations: minimizing the cladding strain induced by fuel swelling, and minimizing the fuel temperature buildup which results from the relatively poor heat conductance of the gas gap. The maximum design strain for the SAP cladding has been specified as 0.5% of the cladding OD, and the maximum fuel temperature of the highest rated element is not expected to exceed 2000°F (1092°C). A minimum radial gas gap of 0.0025 in. (0.00635 cm), with helium as the filling gas, yields acceptable temperatures in the fuel.

On the basis of an extensive survey of data on the swelling of UC fuel, the most probable fuel diameter change of the most highly rated fuel at the target burnup is expected to be 0.75%, assuming isotropic swelling. Figure 0351-2 shows the results of this study expressed in volumetric changes. Based on calculated hot operating clearances, a fuel diameter change of 1.5% would be required to strain the SAP cladding to the 0.5% maximum design value (see Figure 0351-3). The increase in gap temperature drop, which is caused by dilution of the helium with poorer conducting fission gas, is expected to be offset by the reduction in the gap thickness caused by fuel growth. A conservatively sized gas space of 1.4 in. (3.56 cm) is provided at the top of each fuel rod for the accommodation of fission gas. Fission gas release in the highest rated element is expected to lie between 0.075 and 0.15%, resulting in a maximum internal pressure of 38.8 psia. The experimental data on which this prediction is based are shown in Figure 0351-4. Thus, pressure stresses are not expected to be induced in the

TABLE 0351-3

U-305 EXPERIMENT BUNDLE DESIGNATION

LENGTH		ELEVATION (COLD)	BUNDLE DESIGNATION	CLADDING	FUEL	ENRICHMENT (wt%)
16.625 in. (42.23 cm)		465 ft - 1.125 in.				
		463 ft - 8.500 in.	LKA	SAP	UC	1.5
16.625 in. (42.23 cm)
		462 ft - 3.875 in.	LLA	SAP	UC	1.5
16.625 in. (42.23 cm)
		460 ft - 11.250 in.	LWA	SAP	UC	1.5
16.625 in. (42.23 cm)
		459 ft - 6.625 in.	LMA	SAP	UC	1.5
16.625 in. (42.23 cm)
		458 ft - 2.000 in.	LNA	SAP	UC	1.5
16.625 in. (42.23 cm)
BOTTOM PROJECTION		456 ft - 3.1875 in.	LPA	SAP	UC	1.5

0351-6

cladding from fission gas release.

The fuel dimensions for the SAP-clad UC fuel bundles are listed below.

U-305 EXPERIMENT - URANIUM CARBIDE DIMENSIONS AND CLEARANCES

Fuel Diameter	0.499 ± 0.0005 in. (1.27 ± 0.00127 cm)
Cold Diametral Clearance	0.005 ± 0.0015 in. (0.0127 ± 0.0038 cm)
Fuel Length	13.6 in. (34.54 cm)
Cold Axial Clearance	1.4 in. (3.56 cm)

There were a total of 216 fuel rods fabricated, including 54 for the three bundles in the U-305, an equal number for the U-305 Alternate I string, and 108 for the six-bundle, all-carbide U-305 Alternate II string. More than 1100 half-inch slugs (~ 3300 in.) were machined and chemically analyzed for the experiments. Of these, approximately 400 were subjected to metallographic characterization and density determination. A complete report will be issued next quarter describing the experiment in detail, including fuel preparation, characterization, fabrication, cladding testing and inspection, and all analyses.

A large number of related activities have been conducted this period, some in direct support of the U-305 experiment effort and some in support of the longer range uranium carbide development. The more important among these activities are listed below.

A. HYDRAULIC TESTING

A hydraulic mockup comprised of two full-sized U-305 type bundles has been built, instrumented, and flow-tested in a warm water loop at Canoga Park. The mockup consisted of two U-305 UC bundles enclosed within a plexiglass tube sized to simulate the U-3 loop process tube. The bundles were constructed with the same size finned SAP tubing and the same type spacers used in the U-305 with simulated "fuel" of stainless steel. Six pressure taps were spaced along the length of the plexiglass tube to measure the desired pressure drops and absolute

pressures (see Figure 0351-5). Two thermocouples were positioned at the inlet and outlet to the test section to monitor the water temperature. Conductivity cells were positioned along the plexiglass tube and the center dummy rod to monitor rod vibration. Similar conductivity cells were positioned along the plexiglass tube opposite the bundle spacers to monitor bundle vibration. (Figure 0351-5)

The conductivity cell consisted of a nickel plated wire positioned flush with the inside of the plexiglass process tube or the outside of the dummy center rod (insulated from the rod by epoxy). This wire served as an anode to the grounded bundle. Using a salt solution as coolant in the loop, the resistance between the cell and the bundle was measured. This resistance, a function of the distance between the cell and the bundle, was monitored as the coolant flowed past the bundle. Any vibration of the bundle or rods will change this distance and be recorded as a variation in resistance. By the use of dynamic analysis techniques in the processing of the resistance data, the characteristic frequency and magnitude of vibration for the rods and bundles can be determined.

The tests were run in a water loop with water heated to 122°F in order to match the expected Reynolds number of the Santowax OM+25% HB in the U-3 loop. A number of tests on the conductivity cells were made prior to testing to calibrate and check out the system. Vibration data were taken at a number of flow rates in an attempt to produce a curve of vibration frequency and amplitude as a function of flow rate (gpm and velocity), however no vibration was detected. Pressure drops across the spacers and the bundles were also taken at a number of flow rates. Results indicate that the pressure drop with the U-305 experiment should be approximately 90 psi compared with an analytically predicted 80 psi. This difference should not result in difficulties in the conduct of the experiment. The results will be compared with calculated vibration and pressure drop and will be reported next period.

B. CONTACT CONDUCTANCE

A first-order analysis indicates that the HWO CR reference design lies near a region where central fuel temperature is a sensitive function of the conductance of the fuel-cladding interface. It is desirable, therefore, to measure the conductance of the UC-SAP interface under conditions of a contact pressure between 0 to 300 psi; cladding temperature of 550 to 1000°F; with surface finishes between as-cast and those representative of centerless grinding capability; gas atmosphere of helium, argon, xenon; and a UC composition range of 4.8 to 5.0 wt% C or higher.

A program has been initiated to determine the conductance of the SAP-UC interface under conditions of interest for the present HWO CR reference design and will also provide a sound background of information on the surface properties of UC and SAP so that existing conductance theory can be used in engineering calculations for these materials.

Apparatus to study fuel-cladding contact conductance was designed, fabricated, and checked out. The UC fuel specimens were fabricated and characterized preparatory to testing.

C. COMPONENT COMPATIBILITY

The determination of fuel-bonding media-cladding compatibility for the several materials of interest is of prime importance in establishment of a reference reactor design. A program has been initiated to ascertain the extent of the compatibility in UC-SAP couples, and in variations of these materials. The influence of time, temperature, and pressure will be examined, and the extent of any reaction and its consequence will be evaluated.

More than 50 capsules for fuel system component compatibility studies were fabricated. Ten scoping capsules were assembled with UC₂, SAP (UC + UC₂), (UC + U), Al₂O₃ and heat treated at 900, 1000 and 1100°F for 500 hours. Twenty

more capsules were assembled with similar components for testing at lower temperatures (700 to 900°F) for much longer times ($>10,000$ hours).

D. UC FUEL MECHANICAL PROPERTIES

Out-of-pile physical and mechanical property determinations have been initiated to provide guidance in the design of fuel elements. Preliminary data have been obtained on hyperstoichiometric UC (5.3 wt% C) utilizing a sonic pulse-echo technique to determine material elastic properties. These data are:

1) Poisson's ratio	0.28
2) Shear modulus	7.6×10^6 psi
3) Young's modulus	19.5×10^6 psi

The moduli values were lower than expected (stoichiometric UC single crystals show 12 and 30×10^6 psi for Shear and Young's modulus, respectively). The difficulty appears to be large grain size in the samples. Arc casting was tried in order to obtain a finer grain size, but was not successful. Supplementary equipment has been ordered. This will lower the available frequency range by a factor of five, which in theory, would enable the use of a sample with a grain size five times larger than required with the present equipment. A factor of two or three is estimated to be sufficient.

E. DATA ACQUISITION AND PROCESSING

Two efforts have been under way in the area of improved utilization of experimental data. A computer code is being written for the conversion of raw operating data from the U-305 experiment into useful instantaneous and integrated performance information. The experimental parameters of coolant flow rate, reactor power, coolant inlet and outlet temperatures, neutron flux profile, and pressure drop serve as input data for correction of the cladding temperature data to normal operating conditions. The uncorrected cladding temperature data are compiled in

histograms to yield temperature-time profiles. In addition, the time dependence of burnup, pressure drop, and cladding temperatures are monitored weekly and monthly.

The second effort involves the utilization of fast magnetic tape recorders and analog-to-digital conversion equipment to evaluate the high frequency variations in temperature, power, and flow signals obtained from fuel elements under test. Dynamic analyses of these signals permit the measurement of vibration, mixing, film formation, and heat transfer coefficients as characteristic transfer functions of the experiment. Basically, dynamic analyses methods employ electronic filters and analog computers to yield power spectral density and cross spectral density functions. The resultant system response can be examined with the aid of analysis to yield the important parameters. The tape recordings of the raw data are three orders of magnitude more sensitive than chart recordings, and thus system response to known input driving functions can be accurately measured. The added sensitivity of the magnetic tape contains the capability for inherently more accurate data, and the filtering and rescaling allows perception of variations otherwise swamped out by noise.

III. EVALUATION OF EFFORT TO DATE

The primary objective of this task during this period has been accomplished. The uranium carbide fuel bundles for the U-305 irradiation experiment have been designed, fabricated and shipped as planned.

IV. NEXT REPORT PERIOD ACTIVITIES

For the next report period and the rest of **FY 1966**, the activities of this project will be divided into two projects: **viz., Fuel System Development and Fuel Irradiation Testing.**

During the next quarter, the principal activities under Fuel System Development include:

a) Hydraulic and mechanical tests will be initiated on fuel bundles representative of SAP-UC designs for HWOCR to determine the effects of fuel rod spacing and length on vibration and coolant mixing using a water loop.

b) The results of the USAEC-AECL cooperative irradiation experiment, EXP-NRX-721 Phase II containing UC clad in SAP-930, will be evaluated in light of the Phase I experiment and other uranium carbide tests to improve the current understanding of SAP-UC irradiation behavior at high power and high burnup conditions.

c) A detailed plan of action for the post-irradiation examination of the U-305 experiment will be prepared.

d) Extension and verification of SAP-UC compatibility relationships will continue with fabrication and testing of couples of UC-SAP, UC-Pb-SAP, UC-Al₂O₃, and UC-Al.

e) Contact conductance tests will begin to determine interfacial pressure, surface finish, and filling gas effects for the SAP-UC system.

f) UC mechanical properties measurements will be performed as required to fill in necessary areas of information not otherwise available.

During the next quarter, the principal activities under the Fuel Irradiation Testing subtask include:

a) Final mating of the UC half of the U-305 string with the UO₂ half and insertion of the completed assembly into the NRU reactor. This insertion is scheduled for the week of July 26. Prior to this date, AECL will circulate the loop coolant through the purification system for at least a week to establish the necessary level of coolant purity. Also, the experiment thermocouples will be checked out as will the external portions of the data acquisition system.

b) The startup and approach to full power of the U-305 experiment will be monitored, and adjustments in temperatures and/or flow rates will be made as necessary to achieve desired operating conditions.

c) The temperature, flow rate, power level, pressure drop, and neutron flux data will be monitored continuously on the U-305 experiment during the irradiation, and its performance will be evaluated in comparison with predicted behavior. Weekly results will be reported in the form of CRT plots of key parameters.

d) Design and analysis of the first two UC irradiation experiments planned for WR-1 core positions will be initiated.

e) Procurement action will be initiated on long-lead-time equipment and materials required for the WR-1 irradiation tests.

f) Assembly effort will be completed on the U-305 Alternate I experiment which is intended as a backup string to be inserted in the U-3 loop in case difficulties arise with the initial irradiation.

g) Design drawings on the U-305 and the alternate strings will be modified as necessary to reflect the as-built details.

h) The pre-irradiation data package, including fuel and cladding characterizations, hydraulic testing results, results of cladding tests, and the finalized data reduction code writeup will be issued.

FIG 0351-1. U-305 EXPERIMENT

FIG 0351-2. URANIUM CARBIDE FUEL GROWTH AND SWELLING CHARACTERISTIC AS A FUNCTION OF TEMPERATURE

FIG 0351-3. CLADDING STRAIN vs FUEL BURNUP FOR A 0.005 - IN. FUEL - CLADDING DIAMETRAL GAP AT POWER AND VARIOUS FUEL GROWTH RATES

FIG 0351-4. URANIUM CARBIDE FISSION GAS RELEASE AS A FUNCTION OF TEMPERATURE

BUNDLE CROSS SECTION

FIG 0351-5. INSTRUMENTATION LAYOUT - U-305 HYDRAULIC MOCK-UP

HWO CR QUARTERLY REPORT

TITLE: UC Fuel Element Process Development

SUBTASK NO.: 0352

PROJECT ENGINEER: A. L. Kohl

REPORTING PERIOD:

Inception through
June 1965

I. PROJECT OBJECTIVES

The objective of this project is to develop a reliable low-cost process for producing hyperstoichiometric UC fuel elements within HWO CR specifications required for high performance.

II. PROGRESS DURING REPORT PERIOD

Melting and casting studies showed that UC fuel slug surfaces are improved by: (a) the use of previously-outgassed molds ($\sim 2000^{\circ}\text{C}$), (b) casting into hot molds ($900 - 1000^{\circ}\text{C}$), and (c) casting with the melt at slight superheat. Preliminary tests indicate that centerless grinding can be employed for the economical surface finishing of hyperstoichiometric cast UC to less than 200 micro-inches.

Tests on the carbothermic reduction of UO_2 indicated that the objectives of producing hyperstoichiometric UC castings by carbon arc melting can probably best be met by using slightly hypostoichiometric UC melt stock in the 4.70 to 4.80% carbon range (with the balance of the required castings to be made up from arc-melting electrode). Hypostoichiometric UC melt stock was found to be preferable to hyperstoichiometric melt stock because of the susceptibility of the latter to hydrolysis under normal handling conditions. A UC fuel production pilot line study was initiated during the report period to determine technical and process cost information required

for the preparation of a process specification. Most of the work is still of a preliminary nature. The scale-up of the skull casting operation to a 31-kg pour from the prior reference 18-kg pour was accomplished successfully. Process operation batch sizes and yields for 44,000-kg UC/year throughput were calculated. A preliminary process flow chart and facility layout were prepared.

A small exploratory casting program was initiated to determine the feasibility and/or problem areas involved in casting hyperstoichiometric UC ring-type fuel for the HWOCR Program. A unique mold gating system was developed which resulted in the successful single-cavity arc casting of UC fuel rings in outside diameters varying from nominal $3/4$ to $3\frac{1}{2}$ in., all having nominal $\frac{1}{4}$ -in. wall thickness.

A. UC SURFACE FINISH IMPROVEMENT

The current reference UC fuel rod design utilizes cast UC slugs helium-bonded in SAP cladding. While fuel-to-cladding clearances exist in the early stages of fuel rod life, some UC growth occurring with irradiation will reduce significantly or entirely eliminate this clearance by the end of the fuel life. At this stage of fuel rod life, surface irregularities commonly found on UC slugs in the as-cast condition may stress the SAP cladding beyond its local ductility limit and cause fuel cladding failure. For this reason, a means must be developed for improving the surface quality of UC cast slugs.

The principal deterrents to the use of slugs in the as-cast condition include surface protuberances, excessive "cold-shuts" (surface irregularities), and excessive "bow" (non-straightness). Techniques were investigated to

improve the surface of cast slugs either through modifications in the UC melting and casting operation, or by subsequent surface finishing operations. Investigations of acceptable slug straightness have not yet been undertaken.

A small arc-melting furnace was modified for hot-mold casting using a helical, graphite-resistance mold-heater. Significant improvements were noted in the cast surface finish due to: (1) the use of outgassed graphite molds ($\sim 2000^{\circ}\text{C}$), (2) hot-mold casting, with 900 to 1000°C mold temperature appearing optimum, and (3) the use of a slight superheat in the melt. The use of outgassed preheated molds reduced the amount of diametric "clean-up" required on $\frac{1}{2}$ -in.-dia castings from ~ 32 mils (with ambient temperature, non-outgassed molds) to ~ 19 mils. Metallographic examination also indicated a more uniform grain size is obtained using a hot mold heated to 1000°C , as shown below in the data from grain sizes observed in 6-in.-long UC castings (4.9%C):

<u>Mold Temp ($^{\circ}\text{C}$)</u> <u>Slug Portion</u>	<u>Ambient</u>		<u>1000</u>		<u>1200</u>	
	<u>Edge</u>	<u>Mid Radius</u>	<u>Edge</u>	<u>Mid Radius</u>	<u>Edge</u>	<u>Mid Radius</u>
Top (in.)	0.0100	0.0120	0.0085	0.0085	0.0100	0.0070
Mid (in.)	0.0085	0.0085	0.0100	0.0085	0.0100	0.0070
Bottom (in.)	0.0070	0.0070	0.0085	0.0085	0.0070	0.0070

Melting and casting tests made at 2000, 2500 and 3000 amp, indicating superior surfaces from a 2500-amp input (compared with the previous reference 2200 amp), suggest advantages in slight melt superheat; excessive UC_2 formation by reaction of UC with the graphite mold was observed at 3000-amp operation.

Preliminary tests with a centerless grinding device indicated this to be a low-cost method for improving the surface finish of cast UC fuel slugs.

Though only 3-in. length slugs ($\frac{1}{2}$ -in.-dia) could be handled on this unit, the data indicated that slugs with initial straightness of ~ 0.010 in. for a 6-in. length could be: (1) "cleaned-up" with less than 0.030-in. diametral removal; (2) reduced in diameter by stock-removals of 0.003 to 0.004 in./pass after some "skinning" cuts had been made for moderate "true-ups"; and, (3) controlled to ± 0.0005 -in. on OD and T.I.R. With this setup (10-in. dia x $\frac{1}{2}$ -in., 60 grit, metal-bonded diamond wheel), surface finishes of 120 to 180 micro-inches were readily attained, with ~ 18 sec required for 0.003 in. stock-removal per pass on a 3-in. length. The unit was used successfully in a small production effort involving the grinding of ~ 300 ft of $\frac{1}{2}$ -in.-dia slugs, in the 4.82 to 5.4% carbon range.

Based on these promising results, a small (5 hp) commercial centerless-grinding machine was purchased and has been installed. Preliminary tests on this unit, with a 10-in. dia x 4-in. wide, vitreous SiC wheel have been even more encouraging; this unit can easily handle 6-in. lengths. Diameter tolerances and straightness of slugs appear to be controllable to 0.0001 to 0.0002 in., and surface finishes of ~ 75 micro-inches were attained. Diametral stock removals of 0.005 to 0.010-inch/pass may be possible, and a 6-in. slug was ground in only 12 sec at approximately one-half the rated speed of the machine.

B. UC FEED STOCK PROCESSING

UC feed stock synthesis tests were carried out at the 2 to $2\frac{1}{2}$ -kg level to evaluate: (1) optimum reaction temperatures and UC carbon compositions for the conversion of UO_2 to low oxygen content (< 500 ppm) uranium carbide, and (2) the potential composition control benefits to be derived from (a) ball-milling the raw materials to achieve a more uniform and

intimate blend, or (b) the use of nickel additives in the blended powders to promote removal of excess oxygen via liquid phase flux mechanism at reaction temperatures $\leq 1900^{\circ}\text{C}$.

A graphite induction furnace was used with a multi-crucible arrangement permitting the simultaneous reaction of four oxide-graphite batches. Individual batches (10 kg) were partitioned for comparative vacuum treatments at different temperatures; results are shown in Tables 0352-1 and 0352-2. These data indicated: (1) reactions were incomplete resulting in high oxygen levels at temperatures below 1950°C , and the full 4-3/4-hour hold time (reference production method) was required for reaction completion, (2) results were improved by either ball-milling or adding nickel powder (even without optimization of heat cycle or nickel content), (3) greater amounts of carbon "pickup" occurred with increasing temperature, (4) UC_{1+x} could be prepared with low oxygen content, but hydrolyzed readily during normal handling to pick up additional oxygen, and (5) good "in-batch" homogeneity and reproducibility of data ($\pm 0.05\%$ C) could be obtained from a given mix, under similar treatment. Overall, ball-milling provided the best results but the extra costs involved in charge preparation cannot yet be justified by these marginal improvements.

Since as-reacted UC_{1-x} appears less susceptible to hydrolysis action and can apparently be prepared with oxygen contents similar to those of UC_{1+x} , and since there are advantages for UC feed stock of lower carbon content in arc-melting operations (lesser U-metal additions required to compensate for carbon pickup by electrode erosion), it appears that slightly hypostoichiometric UC will prove to be the preferred carbide feed stock preparation. Additional experiments, indicating a trend for combined $\text{C}+\text{O}_2$

TABLE 0352-1

UC FEED STOCK PREPARATION (2 1/2-kg LEVEL)

% C ADDED	11.42		11.42		11.57		11.72	
UC INTENDED (% C)	4.70		4.70 W/Ni *		4.90		5.1	
C AND O ₂ ANALYSES † (Wt % & ppm) FINAL TEMP (°C)	C	O ₂	C	O ₂	C	O ₂	C	O ₂
1650	4.71±0.05	3020±630	4.75±0.03	980±65	4.83±0.02	2140±1250	4.96±0.07	2190±810
1800 **	4.75±0.03	3060±630	4.71±0.04	690±290	4.94±0.04	1380±190	5.07±0.12	1303±820
1800	4.72±0.07	2390±390	4.71±0.02	925±170	4.90±0.04	2150±65	5.02±0.02	860±90
1950	4.67±0.03	480±165	4.74±0.02	353±173	4.90±0.05	625±230	4.98±0.05	480±35

NOTES: * POWDER BLEND CONTAINED 9.7% NICKEL POWDER (-200 MESH, 99.9% PURITY)

† FROM FIVE ANALYSES FOR CARBON AND THREE ANALYSES FOR OXYGEN

** SPECIAL HEAT CYCLE INTENDED TO BENEFIT BATCH WITH NICKEL, i.e., 3 1/2 hrs AT 1550 °C AND 1 1/2 hrs AT 1800 °C; REMAINING RUNS WERE 4 3/4 hrs AT TEMPERATURE

TABLE 0352-2

UC FEED STOCK PREPARATION (2-kg LEVEL)

% C ADDED	11.574		11.644		11.530		11.802	
UC INTENDED (% C)	4.90		4.90 B-M *		4.90 W/Ni †		5.10	
C AND O ₂ ANALYSES (Wt % & ppm) FINAL TEMP (°C)	C	O ₂	C	O ₂	C	O ₂	C	O ₂
1950	4.90±0.03	335±55	5.02±0.02	335±60	4.87±0.03	205±45	5.17±0.05	690±35
1950 **	4.83±0.05	1190±70	4.98±0.03	415±70	4.80±0.02	385±30	5.12±0.05	865±45
1900	4.83±0.04	1085±80	4.99±0.03	345±15	4.87±0.04	555±50	5.18±0.04	657±40
2050	4.93±0.03	620±35	5.08±0.03	235±60	4.90±0.05	425±75	5.19±0.09	365±20
1950	4.97±0.04	455±45	5.07±0.03	315±12	4.84±0.04	385±65	5.32±0.03	415±40

NOTES: * BALL-MILLED BATCH; OXIDE AND GRAPHITE WET BALL-MILLED 24 hrs, WITH ISOPROPYL ALCOHOL IN PORCELAIN JAR WITH STEEL BALLS

† NICKEL ADDED AS 0.8 Wt %, i.e., 80 gm (-200 MESH) PER NOMINAL 10-kg BATCH

** SPECIAL HEAT CYCLE, 3 1/2 hrs AT 1550 °C AND 1 1/2 hrs AT 1950 °C; REMAINING RUNS WERE 4 3/4 hrs HOLDTIME AT TEMPERATURE

content (as in UC_xO_y) to total $\sim 4.75\%$ when UC_{1-x} is prepared at $1950^\circ C$, have suggested the feasibility of preparing hypostoichiometric UC of low oxygen content, if the carbon content is 4.70 to 4.75%. Typical confirmatory data obtained at the 10-kg level are shown in Table 0352-3, with the 4.75% (C+O₂) total evident even with batches exposed to furnace leaks or weighing errors. These preliminary conclusions require confirmation at production-level batch size.

TABLE 0352-3
CARBOTHERMIC-REDUCTION OF UO_2 TO UC_{1-x} (10-kg LEVEL)
PREPARED AT $1950^\circ C$

GRAPHITE INPUT (Wt %)	ANALYTICAL RESULTS		COMBINED	COMMENTS
	CARBON (Wt %)	O ₂ (Wt %) (ppm)	C + O ₂ (Wt %)	
11.433	4.67 ± 0.01	527 ± 114	4.72	SLIGHTLY LOW CARBON
11.459	4.74 ± 0.02	140 ± 66	4.75	
11.459	4.73 ± 0.02	270 ± 35	4.76	
11.432	4.53 ± 0.02	1983 ± 331	4.73	KNOWN FURNACE LEAK
11.451	4.63 ± 0.01	973 ± 136	4.73	SUSPECTED WEIGHING ERROR
11.530*	4.73 ± 0.04	312 ± 114	4.76	

* 20-kg RUN; HIGHER GRAPHITE INPUT REQUIRED DUE TO LESSER CARBON PICKUP WITH LARGER BATCH SIZE.

C. UC FUEL PRODUCTION PILOT LINE STUDY

1. Melt Stock Production

A preliminary analysis of existing UC manufacturing technology indicated that scaleup of UO_2 -C agglomeration for UC melt stock production to 100-kg batches (from the present maximum batch size of 48 kg) should be feasible, considering both nuclear safety criteria and efficiency of the blending and agglomerating operations. Carbothermic reduction of the 100-kg agglomeration batches to UC similarly appeared to be feasible with minor equipment modifications

The required equipment modifications (including procurement and installation of a 3-cu-ft blender and modification of a vacuum sintering furnace by installation of a larger induction coil and crucible assembly) are presently in progress so that process scaleup feasibility can be established.

2. Skull Casting

The developed UC skull casting technology, which was utilized for the fabrication of the Hallam Nuclear Power Facility Core II, was based on a 60-kg maximum skull which yielded an average pour size of 18-kg and an average of 12.7 kg of fuel slugs. Batch sizes in this fabrication effort were limited by nuclear safety criteria. A cursory analysis of nuclear safety criteria indicated that scaleup of skull size to approximately 120 kg is feasible in the enrichment range of 1.0 to 1.5% U^{235} under consideration for the HWOCR demonstration plant. Skull furnace vessel and power supply limitations similarly indicated a maximum scaleup to this level with the objective of pouring approximately 40 kg from the melting crucible. A 120-kg skull melting crucible was designed for insertion in the existing furnaces. However, prior to initiating fabrication of the 120-kg capacity crucible, a smaller capacity crucible remaining from the HNPF production effort was re-machined to increase its diameter and depth, thereby increasing the maximum skull size to 85-kg. Five melts were made using normal uranium metal and graphite as a charge in an 80-kg skull. Maximum melting power was increased to 12,500 amp from the 10,000 used for HNPF Core II, and pour size for the five melts averaged 31.6 kg, with a range of 29.5 to 34.0 kg. The largest UC pour made prior to this was 29.5 kg from a 70 kg skull.

Mold optimization studies were made which indicated that ~30 kg of slugs could be obtained from a 38 in 40-kg pour, using a 37-slug mold configuration with each slug 0.645-in. dia. by 12-in. long. This slug diameter is 0.020 in. larger than the largest diameter presently under consideration for a HWOCR reference SAP-clad UC fuel element. Some skull furnace vessel modification will be required to achieve the 12-in. cast length. The same 37-slug configuration was used for the trial melts, but with a 9-in. slug length and 0.520-in. diameter, to provide fuel slugs for dry centerless grinding experiments.

Casting of an end dish in the bottom end of the fuel slug was incorporated in the mold design to reduce the amount of end dish machining. The elimination of bowing in long cast slugs is a further objective of the mold optimization studies.

3. Slug Machining

The elimination of oil coolants in UC centerless grinding operations is a prime objective of the slug machining phase. One of the UC machining operations involves end dishing of slugs; the purpose of end dishing is to provide space in the fuel rod to accommodate fuel growth and axial fuel motion, and thereby to minimize stresses on the fuel rod cladding. End dishing of UC slugs to date has been accomplished by electric discharge machining (ELOX) which also requires an oil coolant. The elimination of coolants in UC machining operations reduces fire hazards and eliminates the requirement for vapor degreasing and vacuum outgassing of the fuel following machining. Procurement of tooling for dry diamond grinding of end dishes was initiated. Testing of this equipment is expected during the next report period.

A cursory evaluation of the feasibility of recovering UC grinding dust from a scrubber as U_3O_8 and recycling through carbothermic reduction is being made. Similar material collected in HNPF Core II manufacturing operations is being analyzed for impurity levels to gain preliminary information. If recycling of this material proves feasible, both the quantity of raw material required for a core and scrap recovery costs will be reduced significantly.

D. MANUFACTURING ENGINEERING ANALYSIS

The manufacturing engineering analysis includes: (1) the investigation of maximum batch sizes from an equipment and criticality standpoint; (2) establishing a balanced UC fuel element production line; (3) evaluating elemental costs of each processing step as related to the operation in the process; and (4) determining tooling requirements, tooling life, and tooling costs.

Based on an assumed throughput of 44,000 kg UC/year, process operation batch sizes and yields were calculated. This process throughput was selected as that approximately required to fabricate the fuel loading for the prototype HWOCR core in an 18-month period. Tooling requirements were established, and tool life is being investigated. Process machine rate calculations per operation are being correlated to equipment requirements and manpower needs. An overall process cost estimate is being prepared.

E. UC RING-TYPE FUEL ELEMENT CASTING DEVELOPMENT

An exploratory casting program was initiated to determine the feasibility and/or problem areas involved in casting hyperstoichiometric

UC ring-type fuel, and a survey of the state-of-the-art technology was completed. In previous efforts by other investigators, the major problem areas encountered were the inherent brittleness of UC and the large differential expansion between the UC and graphite molds. The thermal expansion differential caused the castings to crack during solidification from the molten state and subsequent cooling to room temperature.

The initial series of arc casting experiments used thin-walled (0.010-in.) graphite cores on the theory that the core should be strong enough to restrain the molten UC during solidification, but should have sufficiently low strength so that it would crush with the contraction of the UC during cooling. This approach was unsuccessful, because the molten UC (poured at $\sim 2500^{\circ}\text{C}$) partially dissolved the thin-walled cores before they could be crushed. The use of increasingly thicker core walls in a second series of experiments eventually eliminated the dissolution problem, but re-introduced the problem of core inflexibility and cracking of the UC rings.

In a third series, a slip-fit solid-graphite core was inserted into a 0.010-in. wall hollow graphite core. The close tolerance fit between the solid and hollow graphite cores again led to cracking of the UC casting, but prevented dissolution of the thin-wall graphite core. Therefore, the use of solid graphite cores as heat sinks was continued in subsequent experiments.

In a fourth series of casting heats, sufficient clearance was allowed between the hollow and solid cores to permit the simultaneous action of almost unrestrained contraction of the UC, crushing of the hollow core, and removal of excess heat to prevent reaction between the thin-wall core

and molten UC. This resulted initially in the successful casting of 3/4-in. OD ($\frac{1}{4}$ -in. wall) x 3-in.-long hollow UC cylinders, and, subsequently, the successful casting of 2-in.-OD ($\frac{1}{4}$ -in.-wall) x 3-in.-long and 3- $\frac{1}{2}$ -in.-OD ($\frac{1}{4}$ -in.-wall) x 1- $\frac{1}{2}$ -in.-long cylinders. Representative castings of each dimension are shown in Figure 0352-1.

A radiographic analysis of the castings made by this newly developed technique revealed a sound interior, free of casting defects. A metallographic analysis indicated a uniform microstructure transversely across the wall thickness as shown in Figure 0352-2 for stoichiometric UC and Figure 0352-3 for hyperstoichiometric UC. The carbon content which was varied during the program was analyzed to range from 4.72 to 4.81 wt% for cylinders which were intended to be of stoichiometric composition, and 4.92 to 5.02 wt% for hyperstoichiometric cylinders. The oxygen content was found to range from 50 ppm to 310 ppm, with the average being 138 ppm. Maximum permissible oxygen content is currently specified to be 300 ppm.

All of the initial development work described above was carried out in an experimental-size arc-melting furnace, capable of pouring a maximum of 1.2 kg of molten UC in single-cavity molds. The design and fabrication of multiple-cavity molds for large-scale tilt-pour casting of cylinders 3- $\frac{1}{2}$ -in.-OD ($\frac{1}{4}$ -in.-wall) up to 6-in. long, in multiples of five, were completed and scheduled for use in the next series of development experiments.

III. EVALUATION OF EFFORT TO DATE

Surface quality of cast UC has been improved by the use of: (1) previously outgassed molds, (2) hot mold casting, and (3) slight superheat in the UC melt. Centerless grinding appears to be a satisfactory technique for the economical surface finishing of cast, hyperstoichiometric

UC to HWOGR requirements. Further tests are required in each of these areas to confirm the preliminary conclusions.

For the present tentative UC fuel rod design, UC slug surface grinding costs have been estimated roughly to represent 5 to 10% of finished fuel element costs, when the fabrication process is based on the current state-of-the-art. The magnitude of this cost element indicates that a determination of slug surface requirements as related to fuel element performance must be made, because grinding costs are strongly influenced by both product specification requirements and as-cast surface quality.

Slightly hypostoichiometric low-oxygen-content UC has been synthesized from UO_2 , and appears to be a preferred feed stock for carbon-arc casting of hyperstoichiometric UC. Process changes leading to improved UC feed stock homogeneity include: (1) use of 1950°C temperature (in lieu of 2050°C), and (2) synthesis of UC from 4.70 to 4.80% C (in lieu of the present 4.65% C target composition).

The successful feasibility demonstration of the proposed scale ups of UC melt stock production and skull casting will provide a sound base for fabrication cost estimates and will eventually result in significant cost reduction. To date, results on the intermediate scaleup of skull casting strongly indicate that scaleup to the proposed 120-kg level is feasible, and suggest that procurement of a higher capacity arc-melting power supply could result in even larger pours.

If the assumed process scaleups can be demonstrated, a 44,000 kg UC/year production rate appears feasible with AI's existing major process equipment, except in the area of melt stock synthesis. The fabrication of a UC prototype core at costs competitive with other fuels appears very

feasible.

The feasibility of fabricating ring-shaped UC fuel by arc casting has been demonstrated at least six months ahead of the originally anticipated milestone achievement date. The successful casting of UC ring-type fuel represents a major contribution to the UC casting technology and may thereby have a significant effect on future fuel element designs for the HWOCR Program.

A major problem area in the current UC fuel-ring casting process is the introduction of residual stresses and potential cracking sites in the casting due to restrained solidification and cooling. Further development and testing is required to resolve this problem and to adapt the single-cavity casting method to multiple-cavity casting.

IV. NEXT REPORT PERIOD ACTIVITIES

Casting experiments to improve the surface quality and straightness of the cast slugs will be continued. Extensive evaluation of grinding techniques will be conducted with the newly-installed commercial, center-less-grinding machine.

Efforts will be continued on the carbothermic-reduction of UO_2 to UC at the 20 to 30-kg level, with emphasis in the 4.70 to 4.80% C range, at $\sim 1950^\circ\text{C}$ finishing temperatures, attempting to verify the promising results indicated from tests at the 2 and 10-kg level.

Equipment modifications required for scaleup feasibility experiments will be completed. Based on pilot scale runs, a reference UC fuel fabrication process will be defined and base-line process specifications will be established. The manufacturing engineering analysis will be completed and a summary report prepared.

Due to redirection of the program effort, further work on development of a fabrication process for UC ring-type fuel will be deferred until FY 67.

(a) ARC CAST UC HOLLOW CYLINDERS AS-CAST CONDITION - NOMINAL DIAMETER SIZES SHOWN ARE 3 1/2 in. OD, 2 in. OD AND 3/4 in. OD

(b) ARC CAST UC HOLLOW CYLINDERS AS-CAST CONDITION - POSITIONED TO SHOW A POTENTIAL REACTOR FUEL CONFIGURATION

FIG 0352-1. ARC CAST UC HOLLOW CYLINDERS

(a)

(b)

(c)

250 X

FIG 0352-2. TYPICAL TRANSVERSE MICROSTRUCTURE OF AS-CAST HOLLOW UC CYLINDER
2.00 in. OD (1/4 in. WALL) HAVING STOICHIOMETRIC CARBON COMPOSITION,
(a) STRUCTURE ADJACENT TO ID, (b) CENTER, (c) STRUCTURE ADJACENT TO OD

(a)

(b)

(c)

250 X

FIG 0352-3. TYPICAL TRANSVERSE MICROSTRUCTURE OF AS-CAST HOLLOW UC CYLINDER 3 1/2 in. OD (1/4 in. WALL) HAVING A HYPERSTOICHIOMETRIC CARBON (4.92 - 5.02 Wt % C) COMPOSITION, (a) STRUCTURE ADJACENT TO ID, (b) CENTER, (c) STRUCTURE ADJACENT TO OD

MATERIALS DEVELOPMENT

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HWO CR QUARTERLY REPORT	
TITLE: SAP Development	SUBTASK NO.: 0451
PROJECT ENGINEER: S. O. Arneson	REPORTING PERIOD: Inception through June 1965

I. PROJECT OBJECTIVES

The objectives of this task are to: (a) develop and evaluate fabrication processes, end closure techniques, product specifications, acceptance standards, and techniques for producing semifinished and finished SAP shapes; (b) develop physical and mechanical property data on the SAP alloys in configurations required for HWO CR application; and (c) develop reliable sources of SAP alloys and establish the quality and uniformity of obtainable material.

II. PROGRESS DURING REPORT PERIOD

During the report period, the project was organized, plans for immediate and long-range development were made, and work was initiated in the areas of:

- a) SAP fabrication development
- b) SAP joining development
- c) Nondestructive testing
- d) Chemical analysis development
- e) Irradiated SAP cladding evaluation

The following SAP materials were procured for evaluation and fabrication development:

<u>Alloy</u>	<u>Quantity (lb)</u>	<u>Inventory Source</u>	<u>Remarks</u>
SAP-930	400	AECL	Made by Swiss Aluminum
SAP-895	200	AECL	Made by Swiss Aluminum

(Continued)

<u>Alloy</u>	<u>Quantity (lb)</u>	<u>Inventory Source</u>	<u>Remarks (cont.)</u>
XAP-001	450	AECL	Made by Alcoa
XAP-001	100	Alcoa	Current Production
XAP-005	100	Alcoa	Current Production

Approximately 1300 ft of 14-fin tubing was extruded from AI's supply of SAP-930 and XAP-001. An additional 1000 ft of 14-fin tubing was extruded from Alcoa's most recent XAP-001 and XAP-005. Preliminary metallographic examination indicates that the XAP-001 products were significantly improved over the four-year-old XAP-001 material, and that the XAP-005 is much improved as compared to the most recent XAP-001. The improvements were: (a) smaller and few inclusions, and (b) greater macro- and microhomogeneity. A report summarizing these evaluations is being prepared.

Short-term burst tests were run on XAP-001 tubing to determine the effects of macrosegregation on strength. Fourteen tests conducted at 850°F indicated a stress level at failure of 10,000 ± 1000 psi, which is in good agreement with previous data, and showed no significant effect of macrosegregation. These data are only an initial part of the mechanical testing program that was initiated, which includes tensile, burst, and biaxial creep.

Because of the need for development of rapid, highly reliable joining techniques for SAP rod-type and advanced ring-type fuel elements, potential joining processes were evaluated on the basis of AI experience, published information, and discussions with various companies. The techniques considered were (a) eutectic bonding, (b) flash welding, (c) magnetic force welding, (d) ultrasonic welding, (e) friction welding, (f) cold welding, and (g) electron beam welding. The criteria for evaluation included (a) bond quality, (b) reliability,

(c) speed, (d) degree of additional development required, and (e) cost. Data from AI experience and published literature were supplemented with personal discussions during visits made to Canadian Westinghouse, Canadian General Electric, Precision Welder and Flexipress Corp., and Aero projects, Inc. Based on this information, magnetic force welding appeared most feasible for SAP rod-type fuel elements, with eutectic bonding and flash welding as promising backup processes. The joining development program was planned accordingly.

To initiate development of the magnetic force welding process, a purchase order was issued for a 150-kva dual-transformer magnetic-force welder to be delivered during the second quarter of FY 66. In order to accelerate this program during the waiting period for delivery and installation, negotiations were initiated with Battelle Northwest Laboratories for a contract to provide magnetic force welding services for bonding HWOCR fuel element end closures.

All efforts for developing ring-type end closures were postponed until FY 67. However, in the initial evaluation mentioned above, it was concluded that ultrasonic welding is probably the most promising technique for this fuel element configuration.

A six-week development contract was completed with Automation Industries to develop nondestructive test techniques for inspecting billets, finned tubing, and bonded end closures. Ultrasonic inspection of billets was performed by three techniques: (a) longitudinal propagation normal to the billet axis, (b) shear propagation transverse to the billet axis, and (c) shear propagation parallel to the billet axis. All techniques showed promise but the first mentioned proved to be most informative and was used to inspect SAP-895 billets received from AECL. Ultrasonic techniques for measuring finned tubing wall

thickness and for detecting defects have also shown promise. Wall thickness measurements with a precision of ± 0.001 in. are expected. Artificial flaws within tubing fins as shallow as 0.002 in. were detected, and also known defects of 1/16 in. diameter were found in end closure bonds. Evaluation of these processes and recommendations for necessary equipment items will be summarized in a formal report to be issued by Automation Industries during the next report period.

For some time, various sites have indicated confusion regarding accuracy and correlation of Al_2O_3 analysis in SAP materials. An effort was begun to investigate possible reasons for discrepancies and to ascertain which technique(s) offer greatest accuracy. Several techniques were attempted with inconclusive results. The methods included (a) Fast Neutron Activation Analysis, (b) Anhydrous HCl Volatilization, (c) Hydrogen Evolution, (d) Amalgamation, (e) Anhydrous Bromine, (f) Anhydrous Iodine, (g) Acid Dissolution, (h) Caustic Dissolution, (i) Vacuum Evaporation of Aluminum, (j) Direct Oxidation of Aluminum, and (k) Direct Reduction in Aqueous Solution. Initial data indicate halogen-organic, hydrogen evolution, and fast neutron activation analyses warrant further investigation. The effort is still in progress and the results will be reported during the next period.

A program was performed to evaluate SAP cladding irradiated in the OMRE test elements. These elements contained finned, nominal 7% Al_2O_3 SAP irradiated at temperatures up to 850°F with a UO_2 fuel burnup of up to 11,200 Mwd/MTU. The estimated cladding irradiation dosage was 2.4×10^{20} nvt thermal. A metallographic examination was performed to determine whether any noticeable changes in microstructure, such as microcracking or void formation, occurred as a result of irradiation. No such changes were observed. In addition, a

small mechanical testing program was performed which included (a) room temperature fatigue tests, (b) room temperature, 800, and 900°F tensile tests, and (c) 800, 900, and 950°F burst tests. The mechanical tests are in progress, and the results will be reported during the next period.

III. EVALUATION OF EFFORT TO DATE

The fabrication development effort has begun to produce significant data regarding inter-relationships of SAP Al_2O_3 content, fabrication history, microstructure, and mechanical properties which will be extremely important in ultimately defining process, product, and material specifications. Selection of a qualified SAP producer and selection of a reference alloy (i.e., 7% Al_2O_3 vs 10% Al_2O_3) remain significant problem areas which must be resolved at the earliest possible date. Techniques for determining the Al_2O_3 content of SAP alloys require further evaluation.

Based on available information, magnetic-force welding appears to be the best production method for making end closures in SAP-clad fuel elements. Magnetic-force welding equipment is being procured, and an interim contract for magnetic-force welding services is being negotiated.

The hot cell examination of the irradiated SAP cladding from OMRE test elements has shown no observable change in microstructure. The short-term elevated temperature mechanical tests to be reported during the next period will serve as a supplement to the limited irradiation data available on SAP materials.

IV. NEXT REPORT PERIOD ACTIVITIES

Evaluation of finned tubing representative of the most recent XAP-001 and XAP-005 alloys will continue. Biaxial creep tests will be initiated to assess the significance of microstructure, alloy composition, fabrication history, and defects. Attempts will be made to develop additional vendors for extrusion finished and semi-finished shapes. SAP billets and shapes produced by European suppliers will be evaluated for comparison with Alcoa material as an aid to development of material sources.

Coordination of procurement for the 150-kva magnetic-force welder will continue with delivery expected in the second quarter of FY 66. Magnetic-force welding work will be initiated at Battelle Northwest Laboratories to expedite the overall joining effort.

Efforts will continue to correlate microstructure with nondestructive test results, and to develop and refine meaningful testing methods.

The results of the chemical analysis evaluation will be reported.

Short-term mechanical test data on OMRE irradiated SAP cladding will be evaluated and reported.

HWO CR QUARTERLY REPORT	
TITLE: Transition Joint Development	SUBTASK NO.: 0452
PROJECT ENGINEER: S.O. Arneson	REPORTING PERIOD: Inception through June 1965

I. PROJECT OBJECTIVES

The objective of this task is to develop a tubular transition joint between the out-of-core steel piping and the dissimilar material flow tubes which carry the organic coolant through the reactor core. These joints are required to maintain strength and leak-tight integrity under all anticipated conditions of reactor operation. The candidate flow-tube materials for the HWO CR are SAP and zirconium alloys; transition joints are to be developed for both of these materials.

The operating environment of the flow tube includes internal pressure and flow of organic coolant at temperatures near 750°F, thermal cycling, pressure cycling, and vibration. The flow tube will also be subjected to limited piping reactions and other mechanical loads. The principal potential problem areas associated with transition joints are: (a) stresses due to differences in thermal expansion of the dissimilar metal joint components, (b) formation of brittle intermetallics at the joint interface by diffusion, and (c) reactions with the organic coolant such as hydriding while under irradiation.

Parallel joint development efforts are being pursued for both SAP and zirconium alloys. The techniques of hot pressing and roll bonding are being pursued, and the transition joints are being evaluated in both of these efforts. In addition, zirconium-steel joints fabricated by the tandem coextrusion technique are being evaluated. An intensive test and evaluation program for joints

which appear most promising will be conducted, including mechanical and metallurgical evaluation. In the final phases of development, the most promising joint types will be tested as parts of flow-tube assemblies, both out-of-pile and in-pile. The development of a suitable diffusion barrier between the steel and SAP or zirconium flow-tube material is an integral part of this project.

II. PROGRESS DURING REPORT PERIOD

Reference transition joint designs have been selected for development and evaluation. A review of the literature was conducted to determine the nature and extent of difficulties which may be encountered in using rolled transition joints in HWOCR flow tubes. Although rolled joints have been successfully used in water-cooled reactors, the leak-tight integrity of rolled transition joints at $\sim 750^{\circ}\text{F}$, with respect to organics, is questionable and must be demonstrated to be resolved. The short-term mechanical strength of this type of joint usually exceeds the strength of the flow tube and is not considered a problem.

The possible problem of leak-tightness is particularly exemplified by work performed under the ORGEL Program* with SAP-steel rolled joints, where it was shown that this type of joint did not give the degree of leak-tightness desired and that a metallurgically bonded joint is preferred. The experimental work was based on leak-rate measurements of gas-pressurized flow-tube assemblies. These gas-leak measurements were subsequently converted to organic-leak rates. The results indicate a 30 mg/hr organic leak-rate at each transition joint under a pressure differential of 285 psi across the wall of the flow tube. The maximum service temperature requirement of the ORGEL flow-tube is about 420°C (788°F).

* EUR 1830.e, Part II, Chapter 6, p. 74 (1964)

The prime factors affecting leak-tightness appear to be:

- a) Degree of working and wall reduction during rolling
- b) Relaxation of the residual stresses in the tube at elevated temperature operating conditions
- c) "Tube walking" during rolling (rotation of the tube with respect to the outer sleeve or tube sheet)
- d) Cleanliness and dimensional uniformity of joint surfaces prior to joining

The degree of working for small wall reductions also determines the strength of the rolled joint.

A literature survey in the area of tandem extruded Zircaloy-steel transition joints disclosed that no major problems exist from the fabrication viewpoint. This type joint has been qualified by experimental work for reactor service up to 600°F with no measurable formation of Fe-Zr intermetallics at the bond interface. However, the long-term higher operating temperature conditions of the HWOGR tubes (~750°F) will require an investigation into the formation rate of Fe-Zr intermetallics and the subsequent effects on joint strength.

The design of the necessary hot-pressing equipment for fabrication of SAP-steel and Zr-steel joints is complete and procurement action has been initiated. An induction-heated die assembly will be utilized in a vacuum enclosure wherein the flow-tube material is back-extruded over knife-edge ends of the steel tube component. It is expected that hot-pressing temperatures approaching 1600°F may be required for the Zr-steel joints, as compared to about 1050°F for the SAP-steel joints.

Three thermal-pressure cycling test rigs utilizing an organic environment

have been designed, and the components are being fabricated and assembled. These test rigs will permit thermal cycling of short flow-tube test assemblies between organic and argon gas media at temperatures of about 300° to 750°F combined with internal pressure cycling of the tube between atmospheric pressure and about 400 psig.

The diffusion barrier study has begun with the establishment of the diffusion couple bonding technique. Development efforts on other programs* have shown that Nb, Ta, V, and V-Si show promise as diffusion barrier materials for SAP-steel joints. Accordingly, these potential barrier materials are being procured as vapor-deposited coatings on stainless steel components for diffusion studies with SAP. Other coatings and coating methods are being considered for evaluation of diffusion barriers.

A study of the bonding parameters for SAP-steel hot-pressed joints has been initiated.

III. EVALUATION OF EFFORT TO DATE

A considerable portion of the effort during the report period was devoted to procurement of materials, equipment, and tooling. Most of these items are now on hand, and fabrication development in the specific HWOCR geometry is under way. Of major assistance was the determination that information developed in work on SAP-steel transition joints performed in support of the Spanish DON reactor is available to the HWOCR Program.

*AI-65-114, Atomics International, Canoga Park, Calif. (May 1965)

"New Nuclear Materials Including Non-Metallic Fuels," Vol. II, International Atomic Energy Agency, Vienna, Austria, pp. 511-34 (1963)

IV. NEXT REPORT PERIOD ACTIVITIES

Activities during the next report period include the following:

a) Bonding parameter studies (optimization of time, temperature, and pressure) for SAP-steel hot-pressed joints will continue. This effort will include a study of application methods of potential diffusion barrier materials to steel joint components and the effect of varying the joint geometry.

b) Bonding parameter studies will be initiated for hot-pressed zircaloy-steel joints to establish the time, temperature, and pressure requirements for optimum bond quality.

c) Diffusion couple fabrication for both the SAP-steel and the zircaloy-steel material systems will be initiated. A diffusion study over the temperature range of 750 to 1050°F for times up to 5000 hr is planned for the SAP-steel diffusion barrier development and for the Zr-Fe intermetallic formation rate determinations.

d) Procurement action for rolled SAP-steel joints and for both rolled and tandem extruded zircaloy-steel joints will be initiated. These joints will be subjected to testing and comparison with hot-pressed joints.

HWO CR QUARTERLY REPORT

TITLE: Zirconium Alloy Development

SUBTASK NO.: 0453

PROJECT ENGINEER: S. O. Arneson

REPORTING PERIOD: Inception
through June 1965

I. PROJECT OBJECTIVES

The primary objective of this project is to establish the feasibility of utilizing zirconium base alloys for the in-core flow tube and fuel element cladding in the HWO CR. The principal factor limiting the immediate use of zirconium as a core material in HWO CR is its high affinity for hydrogen at HWO CR temperatures and the potential loss of material integrity if hydriding occurs to excessively high levels. The acceptable level of hydrogen content is not well defined, and therefore, establishment of such a limit is a part of the primary objective for this project. A second objective is to specify the optimum conditions for the use of zirconium base alloys under the reference HWO CR environmental and operating conditions.

The immediate objective of this project is to develop sufficient information on the conditions under which zirconium base alloys can be reliably used in order to enable these alloys to be specified for the 300-Mwe prototype plant flow tube and fuel element cladding. These objectives will be accomplished by out-of-pile and in-pile testing where an evaluation of performance will permit:

- a) Selection of an alloy composition(s) with optimum corrosion and hydriding characteristics
- b) Selection of suitable surface treatments or coatings for

inhibiting hydrogen absorption

c) Determination of the recommended level of additives and/or the acceptable limits of impurities for the HWOCR reference coolant in contact with the zirconium base alloys.

II. PROGRESS DURING REPORT PERIOD

The literature was reviewed and evaluated on the corrosion and hydriding characteristics of zirconium base alloys, the mechanical properties of zirconium base alloys prior and subsequent to hydriding, and the irradiation effects on the performance of these materials.

Installation of autoclaves for static hydriding tests was completed, and the autoclaves were placed in operating condition. Engineering and design of a dynamic test facility was initiated and is nearly complete. Area preparation and procurement of material and equipment for the dynamic test facility were initiated.

Effort on hydrogen barrier development was initiated. Oxide coatings were applied by exposure of zirconium specimens to air, oxygen, ozone, and steam. Nitride coatings were applied by exposure of zirconium specimens in a nitrogen retort. Aluminum coatings were applied by a diffusion process, by hot dipping, and by plasma spray. Screening tests to evaluate the efficiency of the hydrogen barriers were initiated in a pressure-controlled hydrogen chamber.

A. LITERATURE SURVEY

The literature was reviewed and evaluated in the following areas:

1) Corrosion characteristics of zirconium and zirconium-base alloys exposed to the reference environment organic coolants with

water additions; the nature and properties of oxide films; and the effects of alloying elements and impurities on the corrosion process

2) Permeation of hydrogen through zirconium corrosion films; the nature of the Zr-hydrides; and the factors controlling the precipitation of the Zr-hydrides

3) Mechanical properties of corroded and hydrided zirconium base alloys.

4) Radiation effects on the corrosion and hydriding behavior of zirconium-base alloys exposed to organic and water coolants.

The literature review shows that while the corrosion and hydriding of zirconium-base alloys in water and steam have been extensively investigated, this information cannot be fully applied to organic coolants where hydrogen is produced by both the pyrolytic and radiolytic decomposition of the organic coolant and where the operating temperature may be 200 to 250°F higher than in water-cooled reactors. The hydriding rates of zirconium alloys not protected by a suitable coating are probably unacceptably high when exposed to an organic environment. Other work indicates that the hydriding characteristics of zirconium-base alloys may be markedly reduced by the application of oxide and aluminum coatings, where these function as hydrogen barriers. In the experiments where oxide coatings were investigated, it was noted that control of the water level in the organic at a level of at least 50 ppm was required to maintain the protective oxide barrier and to heal defects which occur as a result of cracking or spalling. Aluminum was shown to be an effective hydrogen barrier where reductions in hydriding rates by a factor of 200 were observed. In these experiments, the thickness of the aluminum layer was 0.030 in. Subsequent work, however, indicates that the principle hydrogen

barrier is the surface oxide on the aluminum layer rather than the actual coating thickness.

Based on the evaluation of the available information, a development program was prepared. A first phase of this program covers the exposure of promising and commercially available zirconium alloys, coated with potential hydrogen barriers, Zr-oxide and aluminum, to static hydrogen and organic environments. These tests will permit screening and the selection of the more promising combinations for further experimentation under dynamic conditions in out-of-pile loops. The final determinations of reliable base alloy/coating combinations will be carried out in irradiation experiments under HWO CR environmental conditions.

B. PROCUREMENT OF MATERIAL

Zircaloy-2, Zircaloy-4, and $\text{Zr-2}\frac{1}{2}\text{Nb}$ alloys were procured as tubing, sheet, and/or bar stock in accordance with the applicable ASTM specifications for Zr-base alloys.

C. EXPERIMENTAL FACILITIES

The following available facilities were adapted and made operational to the HWO CR Zirconium Development Program:

- 1) One 5-gal stainless steel autoclave with electric heating
- 2) One 1500-cc stainless steel autoclave
- 3) One 1000-cc stainless steel autoclave
- 4) Eight 125-cc stainless steel autoclaves
- 5) A combination chamber and thermal balance with continuous gravimetric recording system
- 6) An induction furnace for the W vapor deposition by $\text{WF}_6\text{-H}_2$ reaction

7) Electron beam equipment for the W vapor deposition by vacuum evaporation

8) Plasma spray equipment

9) A hydrogen retort for tensile testing

Two small material testing loops for environmental testing under dynamic conditions are being designed. Engineering design of the test system is ~90% complete. Each loop includes two test sections for coolant velocities of 15 and 30 fps at bulk organic coolant temperatures up to 750°F. Area preparation and procurement of material and equipment are under way.

D. HYDROGEN BARRIER DEVELOPMENT

In the development of suitable barriers for inhibiting corrosion and hydriding, the following surface treatments and coatings of Zr specimens have been performed:

1. Oxidizing Treatments

Oxidizing treatments of Zr specimens were carried out using dry air, oxygen, ozone, and steam. The exposure times required to develop a 1- μ oxide film onto etched Zr-2 and Zr-4 coupons treated with air and oxygen are shown below.

Time (hr) Required to Produce a 1-Micron Oxide Layer on Zirconium Alloys

Temp.	Dry Air		Oxygen	
	Zr-2	Zr-4	Zr-2	Zr-4
550°C	1.2	--	2.1	--
500°C	5.5	4.3	11.3	3.8
450°C	14.5	19.5	28.1	19.5
400°C	~ 213	114	179	113

Ozone treatments were performed with the Wesbach Ozonator. No weight gain

was observed at temperatures below 100°C . Treatments performed at 400°C indicate weight gain rates similar to those obtained with pure oxygen.

A lustrous, tenacious oxide coating of dark blue-black color was obtained in these oxidizing treatments.

A series of Zr-2 and Zr-4 specimens were oxidized in steam in autoclave treatments at 800°F , 1500 psi, for 70 hr. The specimens were exposed to the steam in three different surface conditions: etched, sand blasted, and abraded. Eighteen of twenty-five specimens developed the continuous and tightly adherent layer of black-lustrous oxide; however, seven coupons of Zr-4 in either the abraded or sandblasted condition yielded the loosely adherent, white oxide. Additional autoclaving trials are in progress.

Representative samples of each oxidation treatment are under metallographic inspection. Microprobe scanning and X-ray diffraction analysis are under way for the comparative evaluation of the treatments.

Additional samples containing the oxide layer will be used for hydriding testing in retorts, autoclaves, and in the circulating loops currently being designed. In addition, some specimens were attached to the hanger rod assembly of the U-305 irradiation experiment.

2. Nitriding Treatments

Based on the high temperature characteristics of the Zr-N reaction kinetics and on the favorable Zr and N to ZrN molar volume ratio, nitrogen has been considered as a potentially efficient interstitial barrier.

The samples of Zr-2 initially nitrided in a chamber and thermal balance attachment yielded poor results. Air leaked in at a partial vacuum and gave nonuniform coatings. Nitriding was subsequently done in the hydriding retort. Temperatures over 800°C were required to reach appreciable Zr-N₂ reaction rates. Weight gains obtained on 0.030-in thick, 1-in x 4-in of

Zr-2 and Zr-4 are given below:

<u>Material</u>	<u>Temperature(°C)</u>	<u>Time(hr)</u>	<u>Weight Gain(gm)</u>
Zr-2	800	10	0.0036
Zr-2	800	23	0.0031
Zr-2	850	7	0.0039
Zr-2	850	18	0.0065
Zr-2	900	5	0.0079
Zr-4	800	5	0.0036
Zr-4	850	7	0.0067

The golden-color ZrN coatings were obtained on all of the retort treated specimens. The appearance and uniformity of these coatings was excellent. Control specimens are currently under metallographic and microprobe analysis. Samples were attached to the U-305 irradiation experiment hanger rod.

3. Aluminum Coating

Since experimental work performed by other investigators show that aluminum or the subsequently developed oxide is a promising hydrogen barrier, a variety of coating experiments were performed.

Two approaches are being investigated as means of applying aluminum as a hydrogen barrier:

a) Direct application of the aluminum onto the pickled or abraded surface of Zr specimens, with and without a Zr-Al solid-state bonding between phases

b) Surface protection of the Zr specimens by a Zr-Al diffusion barrier prior to the Al deposition.

The following progress was made on the first approach:

a) A total of 34 Zr-2 and Zr-4 specimens were coated with aluminum using a diffusion process. The nature and penetration of the Al-Zr intermetallics developed by this treatment are under evaluation.

b) Six specimens of Zr-2 were aluminized by a hot dipping technique (1400°F - 2 min). Four specimens received a subsequent heat treatment under vacuum at 1000°F for 24, 48, 120, and 144 hr. Metallographic examination is under way to evaluate the kinetics of the Zr-Al diffusion reaction.

c) A series of Zr-2 and Zr-4 specimens were coated by plasma arc spraying. A 40-kw Plasmadyne argon sprayer was used to apply a 3 to 5 mil coating of Al to a set of specimens and a similar coating of Al_2O_3 to another set; both sets had been previously grit blasted. Visual examination indicated excellent appearance and continuity; adherence and compactness of the coating (expected to have a density greater than 90% of theoretical, with no open porosity) are under evaluation. Heat treatments of the specimens are planned for improving coating characteristics, if necessary.

4. Tungsten Coating

Tungsten was briefly evaluated as a potential excellent hydrogen barrier, even though it has a relatively high neutron absorption cross section, because insufficient information is available to evaluate both the amount of tungsten required for hydrogen barrier effectiveness and the effect of this quantity on overall reactor neutron economy.

Tungsten coatings have been applied on Zr specimens by two different deposition techniques:

a) Electron beam vacuum evaporation

b) Gas-phase chemical decomposition

Electron beam vapor deposition was performed on etched specimens using the following operating parameters:

a) Accelerating voltage (kv)	140
b) Beam current (ma)	15
c) Focusing current (ADC)	0.5
d) Distance from W target (in.)	3
e) Integrated coating time (min.)	15
f) Vacuum (mm Hg)	1×10^{-4}

The tungsten layers, several microns thick, were of excellent appearance and uniformity. Metallographic examinations of some of these specimens have shown that no effective bond was obtained between the tungsten and the zirconium base material. In some areas, a continuous void between the two metallic surfaces was found. Since it is apparent from these examinations that a significant effort will probably be required to develop a process for applying high integrity tungsten coatings on a zirconium base material, no further work with tungsten coatings will be performed.

5. Preliminary Testing of Hydrogen Barrier Efficiency

A series of screening tests were performed to obtain a preliminary evaluation of the efficiency of some of the proposed surface treatments as hydriding suppressors. The results can be used only on a comparative basis since the testing conditions do not reproduce actual hydriding conditions anticipated in the HWOCR core. To perform these tests, the pretreated specimen is placed in a hydrogen chamber and the system is evacuated and backfilled three times to give a pure H_2 atmosphere at normal atmospheric pressure and room temperature. The volume of the system is 350 cc. The

retort containing the test coupon is then heated to 500°C and manometer readings are taken for 6 to 8 hr. Pressure decrease during the test permits the calculation of the quantity of H₂ reacted. To illustrate typical reaction periods for various coatings, some of the results are given below:

<u>Specimen</u>	<u>Time to absorb 100 ppmH₂ (hr)</u>
Pickled Zirc-2	0.1
Zirc-2 oxidized in dry air at 400°C	4.68
Zirc-2 oxidized in dry air at 500°C	1.75
Zirc-2 oxidized in dry air at 550°C	2.55
Zirc-2 oxidized in O ₂ at 450°C	5.30
Zirc-2 oxidized in O ₂ at 500°C	2.68
Zirc-2 nitrided in N ₂ at 850°C	2.90

These results indicate that this method can be used as a simple, inexpensive means for gross screening of potential hydrogen barrier materials.

III. EVALUATION OF EFFORT TO DATE

The majority of the effort on this project has been directed at providing a firm technological basis for the experimental program, consistent with HWOCR program objectives, and for preparing test facilities and test specimens.

Progress has been made in the deposition of coatings on zirconium-base alloys; however, reliable test results are not yet available for evaluation. Because the results obtained for those specimens coated with tungsten indicate that a process must initially be developed and the overall acceptability of tungsten in HWOCR is not favorable, this effort has been discontinued.

IV. NEXT REPORT PERIOD ACTIVITIES

Environmental testing, in a static system, will be performed to determine the inhibiting characteristics of a variety of hydrogen barrier materials. In these tests, coated zirconium-base alloy specimens will be exposed to hydrogen and to organic coolant in the temperature range of 400 to 500°C. In addition, the coating of various barrier materials will continue. Representative samples will be subsequently evaluated for coating integrity, metallurgical characteristics, and the effects on the base material. Two small dynamic loops, currently in the design phase, will be constructed. These loops should become operational during the second quarter of FY 66.

HWO CR QUARTERLY REPORT	
TITLE: Lubricity and Wear of Organics	SUBTASK NO.: 0454
PROJECT ENGINEER: S. O. Arneson	REPORTING PERIOD: Inception through June 1965

I. PROJECT OBJECTIVES

The objective of this task is the determination of materials which are capable of long-life performance when subjected to various degrees of sliding and rolling friction wear in organic coolants under HWO CR operating conditions of temperature and pressure. The information from this task is the primary basis for design and development of HWO CR components (such as the refueling machine) which are expected to utilize an organic working fluid.

II. PROGRESS DURING REPORT PERIOD

A. LITERATURE REVIEW

A literature review was conducted to establish the extent of the state-of-the-art technology on the lubricity of polyphenyls and on wear characteristics of bearing materials in organic media.

Experimental work performed at AI for the POPR Program indicates that terphenyls, at 700°F and bearing pressures to 637 psi, have lubricating qualities approximately comparable with those of water and not significantly better than no lubricant at all. There are also data which indicate that decomposition products accumulate between bearing surfaces and form carbonaceous deposits which may effect a reduction in the coefficient of friction.

The selection of potential materials and combinations thereof for the fuel handling machine and the control rod drive mechanism for the POPR Program

was reviewed for applicability in the HWO CR Program.

In a series of sliding friction tests run on 29 different combinations of materials (with bearing pressures of 537 psi and with 700°F Santowax R as the lubricant and also without lubricant), the following combinations were found to be most promising:

<u>Top</u> <u>(Oscillating)</u>	<u>Bottom</u> <u>(Stationary)</u>
Titanium Carbide*	M-3 Tool Steel
Titanium Carbide*	1040 Chromized
Titanium Carbide*	Nitrided 347 SS
Titanium Carbide*	Nitralloy-135
T-5 Tool Steel	H-12 Tool Steel
T-5 Tool Steel	1040 Chromized
52100 Tool Steel	M-3 Tool Steel
M-3 Tool Steel	1040 Chromized
<hr/>	
* Ferre-Tic "C"	

The above combinations selected as acceptable for the POPR Program appear to fit into the following groups of materials:

- 1) High alloy tool steel
- 2) Low alloy tool steel
- 3) Surface treated alloys (chromizing, nitriding)
- 4) Carbide cermets

Since materials from these groups were found to be significantly superior to others, they will serve as a guide for the selection of materials to be evaluated under HWO CR conditions.

The friction and wear characteristics in terphenyls are qualitatively equivalent to the friction and wear characteristics in water; therefore, the data from friction tests in water should be usable as guides in this test program. In a series of reciprocating sliding friction tests designed to evaluate materials for the AMF on power refueling machine for the Douglas Point Plant, split sleeve specimens were evaluated in demineralized water with a pH of 9.5 to 10.5 at 550°F under a bearing pressure of 8 to 16 psi. The water was pressurized to 1500 psi in an autoclave. Combinations of 316 SS (hard-chromed or honed) in contact with hardened 17-4 pH SS withstood 30,000 cycles, and this combination was considered satisfactory for system application. Other combinations, such as honed 316L SS in contact with hard-chrome plated 304L SS, were found to be unsatisfactory due to severe galling after six cycles.

In research on wear on reactor core materials conducted for the ORGEL program, the effect of a simulated "hammering" wear between fuel element and pressure tube was evaluated. In the ORGEL reactor concept, SAP was selected as flow tube material, in direct contact with organic coolant at 400°C (752°F). The fuel elements are located inside the SAP tube. The hammering between fuel element and flow tube was a simulation of possible component vibration at the high flow rate of the organic coolant. Material combinations evaluated include SAP/SAP, SAP/321 SS, and 321 SS/321 SS.

Although the results are not applicable for bearings, they indicate that the wear mechanism of the SAP/SAP, SAP/steel, and steel/steel combinations in terphenyls is basically both cutting (abrasive) and adhesive wear. The wear rate of SAP, whether in contact with SAP or with steel,

was found to be unacceptable. The wear rate of SAP/SAP was 500 to 1000 times the wear rate of steel/steel. Protection of SAP by surface plating did not improve wear resistance. The steel/steel combination showed the least wear.

B. LABORATORY WEAR TEST FACILITY

A laboratory wear-test facility is being set up to investigate the wear properties of materials in polyphenyls. The laboratory now includes one 5-gal autoclave in operating condition with all necessary controlling equipment, utilities, and a ventilation system for 750°F operation. One 10-gal autoclave has been purchased for future large-component testing, and three 1-gal autoclaves are on order for accelerated basic wear tests in polyphenyls at the reference design conditions of temperature and pressure expected in the HWO CR.

Fixtures for both rotary sliding and rolling friction tests have been designed. Installation of the fixtures for rotary sliding friction tests has been completed.

III. EVALUATION OF EFFORT TO DATE

The initial milestones of this project scheduled for FY 65, including a review of pertinent literature, installation of wear-testing equipment and fixtures, and preliminary planning of a friction and wear-test program, have all been achieved.

IV. NEXT REPORT PERIOD ACTIVITIES

A literature survey report will be issued. Rotary sliding friction tests will be conducted under simulated HWO CR operating temperatures and loads. Initial tests will be run in the 5-gal autoclave at a temperature and fluid

pressure of 750⁰F and 500 psi, respectively.

Test couples will be selected from combinations of materials including 440 C SS, 17-4 pH, Stellite 21 and 1, Inconel X, Nitralloy 135, 316 SS, 321 SS, and 347 SS. The test lubricant will be the reference organic coolant which at present is Santowax OMP.

Installation of the 10-gal and 1-gal autoclaves will be completed.

COOLANT TECHNOLOGY

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ATOMICS INTERNATIONAL

COMBUSTION ENGINEERING, INC.

HWO CR QUARTERLY REPORT	
TITLE: Coolant Development	SUBTASK NO.: 0550
PROJECT ENGINEER: R.T.Keen	REPORTING PERIOD: Inception through June 1965

I. PROJECT OBJECTIVES

The objective of this task is to investigate new aspects of organic coolant technology arising from the HWO CR Program. These include physical property studies and coolant selection, coolant composition control, film formation studies, and coolant decomposition and makeup rates. In FY 1966, this task will be divided into two major areas: Coolant Development and Parameters Affecting Film Formation.

II. PROGRESS DURING REPORT PERIOD

A compilation of coolant physical properties was prepared and will be issued as a topical report. Two methods for chlorine determination were investigated and showed good early results. Equipment was ordered and plans made for experimental work on coolant oxidation. The coolant technology section of the HWO CR ten-year development plan was written. Good progress was made on the design and component specifications for a Film Formation Test Loop. A new method of measuring film formation by X-ray fluorescence was proposed, and a preliminary test looked promising.

A. LOW MELTING COOLANTS

A draft of a report was prepared and will be issued as a topical report, summarizing the best values of physical properties of possible coolants as well as their probable cost and availability. A differential scanning calorimeter

was ordered for measurements of melting ranges of irradiated coolants whose dark color makes visual observation difficult. This instrument should permit an estimate of the amount of unmelted coolant present as a function of temperature.

B. CONTINUOUS WATER ANALYZER

Instruments were procured for water vapor determination by P_2O_5 electrolysis and by sorption on a vibrating crystal. A stripping apparatus was ordered for transferring water from a coolant stream to a gas stream. Some preliminary experiments were carried out with the P_2O_5 instrument on samples of PNPf coolant.

C. HALOGEN CHEMISTRY

1. Neutron Activation Analysis for Chlorine

Apparatus was constructed and techniques developed for chlorine determination by neutron activation and coincidence counting of the 1.65 and 2.15-Mev gammas from 37-min. chlorine-38. Samples of coolant and standards were irradiated in the Shield Test Irradiation Reactor (STIR) at Santa Susana for about 10^3 seconds to an integrated flux of 1.2×10^{16} . All samples were irradiated in one-ounce polyethylene bottles, then transferred to unirradiated bottles for counting. The zone of uniform flux was found to be deep enough to irradiate five samples at once; for example, two unknowns and three standards.

Samples are counted by two sodium iodide crystals whose amplified output is fed to two single-channel analyzers set for the desired energy intervals. Pulses which pass these analyzers are fed to a coincidence unit with a resolving time of 100 nanoseconds. The output is the number of gammas in one energy interval which were in coincidence with gammas in the other energy interval. A multichannel analyzer is used to check the single-channel analyzer settings

and the shape of the peaks. The coincidence background is uniformly zero.

A standard solution was prepared by dissolving dichlorofluorescein in deionized distilled water to give an added chlorine concentration of 0.909 ppm. The chlorine-38 activity from this solution was sufficient to follow it through six half-lives. Because of impurity activation, the single-channel analyzers gave complex decay curves from which it would be difficult to estimate chlorine concentration, but coincidence counting eliminated all interferences except a small amount of sodium-24, which was easily distinguished by its longer half-life (15.4 hr). Dilutions of the standard solution were made with the same water, and gave the results shown in Table 0550-1. The first column gives the volume of standard solution in 30 ml of diluted solution, the second column the number of coincidence counts in 4 min. (corrected to an arbitrary time ~ 1 hr after irradiation), and the third column the concentration of chlorine added as dichlorofluorescein. The total concentration in the fourth column is obtained by adding 0.048 ppm, the apparent chlorine concentration in the water. The last column is calculated by assuming the value for the standard solution to be correct (since contamination may occur in dilution). The 15-ml dilution was apparently contaminated, but the others agree within 0.01 ppm.

One run was made on samples from two different lots of Santowax OMP, along with the standard solution and water. The results are shown in Table 0550-2. More data will be needed to estimate the accuracy of the method, but it appears that samples below 1 ppm can be analyzed, and 0.05 ppm or less can be detected with certainty. Considerable refinement of the technique is possible; in particular, larger scintillator crystals will be used to increase the sensitivity and resolution. This will permit using smaller samples or making accurate measurements at very low chlorine levels which may be encountered in coolant purification studies.

TABLE 0550-1

CHLORINE NEUTRON ACTIVATION DATA ON AQUEOUS SOLUTIONS

M1 STD	COUNTS	CHLORINE (ppm)		
		ADDED	TOTAL	FOUND
30	1300	0.909	0.957	0.957*
20	890	0.606	0.654	0.655
15	800	0.454	0.502	0.589
10	490	0.303	0.351	0.361
0	65	-	0.048	0.048

*STANDARD

TABLE 0550-2

CHLORINE IN SANTOWAX OMP BY NEUTRON ACTIVATION

SAMPLE	COUNTS	CHLORINE (ppm)
STANDARD	1070	0.96*
WATER	60	0.05
AB 129**	700	0.63
AB 99**	1020	0.92

*STANDARD

**PNPF FEED COOLANT LOT NUMBER

2. X-Ray Fluorescence Analysis for Chlorine

A germanium crystal detector was purchased. Some measurements were made on samples prepared by adding terphenyl tin chloride to Santowax OMP (PNPF Feed Coolant). The chlorine content of the pure terphenyl, determined at BMI by neutron activation, was 1.2 ppm. The initial results are shown in Table O550-3 as counts per second above a background of over 200 counts/sec. The statistical error shown is largely due to the background. The results indicate a sensitivity (about 6 counts/sec-ppm) sufficient to detect 1 ppm of chlorine. With further improvement of the technique, this method should be useful for routine analysis and acceptance testing at the 1 to 5-ppm level now encountered with PNPF coolant.

D. EFFECT OF OXYGEN ON IRRADIATED COOLANT

A Bendix time-of-flight mass spectrometer, already on hand, was modified to modernize its circuitry and increase its resolution and sensitivity. This instrument will be used to measure higher molecular weight compounds in the oxidation experiments and in other coolant studies. It has unit resolution to 400 and usable resolution to 2000 amu.

Plans were made to measure the effect of radiation on the reaction of coolant with oxygen by measuring the amount of oxygen consumed in a small closed system under irradiation and at the same temperature without irradiation. Products will be analyzed by gas chromatography and mass spectrometry.

E. MIT REACTOR ORGANIC LOOP

Contact was maintained with the MIT loop operation by telephone, correspondence, and visits. It was recommended that the present irradiation at a fast neutron fraction of 0.1 should be completed as soon as possible in order to return to a high fast neutron fraction and high-temperature operation from which estimates for HWO CR conditions can be made with some confidence.

TABLE 0550-3

XRF DATA ON CHLORINE IN SANTOWAX OMP

PPM ADDED	CHLORINE TOTAL	COUNTS/SEC	COUNTS/SEC - PPM
0	1	7 ± 6	7 ± 6
1	2	21 ± 5	10 ± 3
2	3	18 ± 5	6 ± 2
4	5	30 ± 5	6 ± 1
10	11	42 ± 5	3.8 ± 0.5
20	21	89 ± 5	4.2 ± 0.2
100	101	610 ± 6	6.0 ± 0.1

F. CANADIAN PROGRAM

Cooperation with Chalk River and Whiteshell activities was maintained with AECL personnel. Two technicians were loaned to Chalk River for one month to assist in coolant analysis. Recommendations were made for routine coolant quality control tests for the U-3 loop irradiations. These recommendations were based on experience obtained in evaluating coolant quality in the PNPF, and cover such tests as:

- 1) Membrane stain test
- 2) Pulse height analysis of gamma spectroscopy
- 3) Content of iron, water, chlorine, and carbonyl-bond oxygen
- 4) Coolant electrical conductivity
- 5) Gross beta-gamma count
- 6) High boilers
- 7) Flowing pyrolytic fouling tests
- 8) Ash content
- 9) Terphenyl isomeric composition
- 10) Physics properties (density, viscosity, and vapor pressure)
- 11) Flash and flame points

Section 2.6 of AI-CE-1 on the NRU-305 experiment describes in detail the recommended frequencies and sampling procedures for these tests.

G. TEN-YEAR DEVELOPMENT PLAN

The coolant technology section of the HWOCR ten-year development program plan was written.

H. FILM FORMATION TEST LOOP

Design, specification, and procurement activities were initiated on a forced-flow loop to measure film formation rate as a function of coolant composition and of important physical parameters such as temperature, radiation

intensity and type, and flow rate. An earlier design proposed in 1962 was reviewed, and a new design was started based on somewhat expanded design parameters. It was decided that the loop would be constructed and tested at Atomics International, then transported to a large electron accelerator facility for irradiation experiments, which initially will employ a pure gamma flux (bremsstrahlung). Suitable sites were surveyed, and a final choice will be made early in FY 1966.

The heater section (including the target area) was designed and shop prints were prepared. Requirements were determined and purchasing specifications written for the heater section power supply, temperature measurement system, pressure control system, and flow control valves. Pump requirements were established and vendors were contacted regarding its construction. Design calculations were started on orifice meters for flow measurements.

I. FILM THICKNESS MEASUREMENT TECHNIQUES

The film formation studies will require accurate measurements of thin films. One method which is being considered is proton spectroscopy, which would indicate not only the amount but also the composition of the film. The closest facility for this purpose is at Washington State University. A visit was made to this facility, and a request for a proposal to investigate the method was issued.

It would be desirable to have a method of measuring the film-forming potential of the coolant continuously — a film formation rate meter. Two methods which have been considered depend on the electrical conductivity of the film and its iron content, respectively. Some preliminary experiments were carried out on the electrical conductivity method, but measuring the resistance between two electrodes while forming a pyrolytic film on the surface separating

them. While not conclusive, the results suggest that the effect is not large enough to be useful. The other method would use x-ray radiation to form the film on a beryllium window, while measuring the concentration of iron by its x-ray fluorescence. Measurements on a graphite-iron mixture, equivalent to a one-micron film containing 5% iron, indicated that this amount could easily be detected by scanning. Increased sensitivity would be obtained by counting at the iron peak. This latter method will be tested further.

III. EVALUATION OF EFFORT TO DATE

Supplemental funding permitted the program to proceed somewhat ahead of the original schedule, especially in equipment procurement. The work on chlorine analysis is proceeding well, and the results look promising. Other experimental programs are in their initial stages. Design of the new Film Formation Test Loop has progressed; the 1962 design was a very useful starting point.

IV. NEXT REPORT PERIOD ACTIVITIES

A. COOLANT DEVELOPMENT

Experimental work will continue on the water and chlorine analytical methods. Experiments on coolant oxidation rates will be initiated. Coolant technology results from PNPF, MIT and Canadian and European sties will be followed closely and applied to the HWOGR Program.

B. PARAMETERS AFFECTING FILM FORMATION

The heater section of the test loop will be fabricated and flow-tested. The heater power supply, the temperature measurement system, pressure control system, and flow control valves will be purchased. Selection of other components will be completed. The design will be completed except for the by-pass loop for purification and sampling. The irradiation site will be chosen. Experiments on methods of measuring film thickness will continue.

REACTOR PHYSICS

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HWO CR QUARTERLY REPORT		
TITLE: Reactor Physics	SUBTASK NO.: 0610	
PROJECT ENGINEER: S. Visner	REPORTING PERIOD:	Inception through June 1965

I. PROJECT OBJECTIVES

The objectives of the reactor physics task are to provide nuclear design methods incorporating digital computer codes, nuclear cross-section data, and analysis techniques which will predict the nuclear characteristics of HWO CR plants with sufficient accuracy. The uncertainties in such nuclear characteristics as fuel enrichment, lifetime power distribution, control requirements, plutonium production, and reactivity coefficients must be small enough so as not to limit the achievement of the lowest possible energy costs.

The work started during this report period is concerned with the problem of what physics models, codes, and cross sections are to be used in survey, optimization, and design calculations, and how to test the validity of these methods. A rapid survey method is now in operation and with some cross sections, replacements will be checked against critical experiment data during the coming quarters.

II. PROGRESS DURING REPORT PERIOD

A comparison was made of reactivity and effective thermal cross sections, using a 10-region, integral-transport code, THERMOS,¹ with a spatially-dependent thermal spectrum and a homogeneous THERMOS which has an infinite medium spectrum representative of BAM. Three UO₂, 37-rod, HWO CR lattices containing fresh fuel

1. H. C. Honeck, "THERMOS - A Thermalization Transport Theory Code for Reactor Lattice Calculations," BNL-5826 (September 1961).

of 1.3 wt% U^{235} enrichment were considered: A, B, and C, having a D_2O/UO_2 volume ratio of 11.9, 13.6, and 10.0, respectively, and volume fractions of organic inside the pressure tube of 0.42, 0.34, and 0.34, respectively. Using the THERMOS cross sections, 10-region, isotropic scattering CEPTR² cases were run to maintain the same transport and geometrical treatments. These thermal effective cross sections, together with the fast cross sections from FORM,^{3,4} were used in the INGEN-CRUMB^{5,6} burnup studies. The values of k_{eff} and infinitely dilute Pu^{239} cross sections which are sensitive spectrum discriminators, calculated with fresh fuel, as well as k_{eff} and concentrations of Pu^{239} and U^{235} at 20,000 Mwd/T are presented in Table 0610-1. It is seen that the eigenvalue differential decreases slightly during burnup, and that the Pu^{239} content at 20,000 Mwd/T is increased in the infinite medium cases by essentially the same percentages that the Pu^{239} thermal cross sections are reduced. This is because of the near saturation of Pu^{239} in which the macroscopic Pu^{239} cross section tends to be preserved as the microscopic cross section is varied.

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2. P. B. Daitch, J. B. McGovern, and M. Schilder, "CEPTR - An IBM-704 Code to Solve the P-3 Approximation to the One-Velocity Transport Equation in Cylindrical Geometry," MPC-20, Combustion Engineering, Inc. (September 1959).
 3. D. J. McGoff, "FORM - A Fourier Transform Fast Spectrum Code for the IBM-709," USAEC Report NAA-SR-Memo 5766, Atomics International (September 1960).
 4. R. D. Blaine, "NAA Program Description - FORM," AMTD-117, Atomics International (December 1961).
 5. D. Kreidler, "Draft - INGEN-2 - Input Description," Combustion Engineering Internal Memo dated May 13, 1965.
 6. C. J. Hansen, "CRUMB-2 - A One-Point, Four-Group Reactor Burnup Code for the IBM-7070," CEND-130 (May 1961).

TABLE 0610-1

EFFECT OF SPATIALLY-DEPENDENT THERMAL CROSS SECTIONS

	SPATIALLY DEPENDENT (1)	INFINITE MEDIUM (2)	PERCENT DIFFERENCE (3)
$k_{\text{eff}}^{(4)}$ (FRESH FUEL)			
A	1.242	1.253	+ 0.9
B	1.281	1.292	+ 0.9
C	1.246	1.255	+ 0.7
k_{eff} (20,000 MWD/T)			
A	0.8300	0.8353	+ 0.6
B	0.8397	0.8457	+ 0.7
C	0.8567	0.8603	+ 0.4
$\sigma_a (\text{Pu}^{239}) / \sigma_a (\text{U}^{238})$ (5)			
A	693.2000	617.0000	- 11.0
B	661.4000	599.6000	- 9.4
C	687.4000	640.9000	- 6.8
$\alpha_{\text{Pu}^{239}}$			
A	0.5341	0.5095	- 4.6
B	0.5256	0.5035	- 4.2
C	0.5314	0.5157	- 3.0
Pu^{239} CONTENT AT 20,000 MWD/T IN gm/kg ORIGINAL U			
A	2.264	2.520	+ 11.3
B	2.250	2.465	+ 9.6
C	2.501	2.661	+ 6.4
U^{235} CONTENT AT 20,000 MWD/T IN gm/kg ORIGINAL U			
A	1.240	1.153	- 7.0
B	1.130	1.057	- 6.4
C	1.377	1.313	- 4.7

- (1) Spatially-dependent cross sections are from 10-region THERMOS calculations.
 (2) Infinite medium cross sections are from infinite medium homogeneous THERMOS calculations.
 (3) Percent difference between the infinite medium and spatially dependent results.
 (4) k effectives are obtained from 10-region CEPT plus FORM results.
 (5) The Pu cross sections are for infinite dilution in the fresh fuel cases.

A simplified model using two-region THERMOS cross sections was tested in 10-region CEPTR's. Compared with the results from 10-region THERMOS cross sections, the η_f values for case A were 0.8% higher, and cases B and C came within 0.1%.

A test was made of an iteration scheme for obtaining effective thermal homogenized cross sections, such as used in BAM.⁷ In general, this iteration is between infinite medium cross-section codes (such as TEMPEST)⁸ and transport codes (such as CEPTR) to obtain self-shielding factors for modifying the number densities to be used again in the cross-section code. Two-region THERMOS-CEPTR^{1,2} calculations were used as reference, and iterations were made between infinite medium THERMOS cross sections and two-region CEPTR fluxes. The results show that the discrepancy in reactivity between the spatially-dependent thermal spectrum calculation and the infinite medium homogeneous spectrum calculation is increased by 0.2%. More important are the greater discrepancies in Pu²³⁹ to U²³⁸ absorption cross section ratio, where the iterated self-shielded cases now give values about 20% too low. The decreases in α for Pu²³⁹ are 8.7%, 9.7%, and 8.0%, respectively, for cases A, B, and C. The reason the iteration scheme fails is that the fuel in an infinite medium calculation is in too low an energy thermal spectrum, and the additional effect of reducing the fuel number densities through self-shielding tends to soften the thermal spectrum still further.

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7. Blaine, Anderson, and Maskowitz, "BAM, A Code to Compute Thermal Group Constants," AI-64-Memo 185 (August 1964).
 8. R. H. Shudde and J. Dyer, "TEMPEST II - A Neutron Thermalization Code," USAEC Report AMTD-111, Atomics International (June 1962).

Because the KINDLE⁹ burnup code has only two energy groups, two- and four-group comparison calculations were run, using INGEN-CRUMB for the three typical 37-rod HWOCR lattices. Criticality was maintained by thermal poison searches in which buckling was set equal to zero so that differences between two- and four-group leakages would not complicate the comparison. Table 0610-2 shows the relatively small changes in eigenvalue, fissionable plutonium, and amount of U²³⁵ destroyed per Mwd at 20,000 Mwd/T fuel exposure. The differences between two and four groups for percentages of total plutonium at end of life are less than 1%.

The INGEN-CRUMB point burnup chain does not at present extend to Pu²⁴²; calculations show that inclusion of Pu²⁴² (which is assumed to be non-burnable, since daughter products also have high cross sections) would reduce k_{eff} at 20,000 Mwd/T by $\sim .75\%$.

A heterogeneous treatment of fast fission was made by using 10-region, P-3 cylindrical, CEPTR calculations of flux weighting factors above 0.8 Mev. For the three cores considered previously, the increase in k_{eff} above the homogeneous FORM treatment ranged from 1.46 to 1.50%. The contribution to k_{eff} from the fast fissions is about 3.5%, but is partially compensated by the reduced fast-removal cross sections and increased absorption cross sections. Two-region CEPTR calculations gave the same effect within 5%, but a simpler method based on escape probabilities overestimated the heterogeneous fast effect by about 20%.

An evaluation of cross sections used at present indicates that the thermal fuel isotopes are in good agreement with BNL-325 second edition, except for U²³⁵ where the older values of α and γ were used. A comparison of fast-cross-

9. R. Sevy, "A Simplified Two-Group Method for Calculating Reactor Lifetime Behavior" (June 1960).

TABLE 0610-2

COMPARISON OF TWO-AND FOUR-GROUP BURNUP CALCULATIONS AT 20,000 MWD/T

	CASE	FOUR-GROUP	TWO-GROUP	PERCENTAGE ERROR OF TWO-GROUP
k_{∞}	A	0.8385	0.8352	-0.40
	B	0.8499	0.8514	+0.18
	C	0.8624	0.8644	+0.23
FISSIONABLE Pu LEFT PER kg INITIAL U	A	2.941	2.914	-0.91
	B	2.917	2.907	-0.35
	C	3.258	3.241	-0.55
GRAMS U235 DESTROYED/MWD	A	0.5867	0.5874	+0.12
	B	0.5921	0.5923	+0.04
	C	0.5792	0.5796	+0.07

section effects with a recent ORNL-GAM¹⁰ library indicates: (1) reasonable agreement for resonance integrals in U²³⁸ oxide for the same effective rod size, (2) about a 0.4% reduction in reactivity by neglect of the n - 2n cross section in D₂O, and (3) about a 0.6% reduction in reactivity because of CE's higher oxygen absorption cross sections above 4 Mev. Figure 0610-1 shows widely different oxygen cross sections, depending upon the cross section compilation source.

As a step in assessing the validity of physics methods, a literature search of heavy water moderated physics experiments was made. A compilation of those of interest is given in Table 0610-3, including UO₂, UC, and U metal fuel rod clusters. Some simple single-rod and annular fuel cases are included, to give a better check on cross sections, since the less well-understood complications of the cluster geometries are not present.

Certain nuclear codes were made operational, and in some cases additional subroutines were incorporated. A modified form of GAKER¹ was put into operation to obtain bound deuterium, thermal scattering matrices based on Nelkin's model and including a transport approximation for scattering without energy change. TONG-GAM^{11,10} was operated and comparisons made with FORM calculations. TONG-THERMOS^{11,1} was put into operation and LIBP (Library Preparation Routine for THERMOS)¹ was modified to permit: (1) editing or dumping a library tape, (2) updating and writing a new library tape in a more reliable manner. A cross section writer routine for generating a TONG depletion library tape was compiled and employed to generate a four-group library tape for use in comparing

10. ORNL Tape 6010 M-GAM-1 Cross-Section Library.

11. D. R. Vondy, T. B. Fowler, "Computer Code TONG for Zero-Dimensional Reactor Depletion Calculations," ORNL-TM-DRAFT 10-7-64.

TABLE 0610-3

HEAVY WATER EXPERIMENTS

REF NO.	DATE	ENRICHMENT	INSTALLATION	MATERIAL	NO. OF RODS	DIA (in.)	COOLANT	MOD TEMP °C	PITCH (in.)	METHOD
11	1963	NAT	SACLAY	METAL	1	1.73 Al	D ₂ O	RM	SQ 5.1-11.8	CRIT
12	1962	NAT	AB ATOM ENERGI, SWEDEN	METAL	1	1.20 Al	D ₂ O	20-50	SQ 5.1-8.3	CRIT
13	1961	NAT	AB ATOM ENERGI, SWEDEN	OXIDE	1	.53 Al	D ₂ O	20-89	SQ 2.1-2.5	CRIT SUBSTIT
14	1962	3.0%	IFA-NORWAY	OXIDE	1	.28 SS	D ₂ O	RM	SQ 1.3-3.9	CRIT
15	1962	NAT	PLATR	CARBIDE	7-19-31	.50 Al	AIR-DOWTH	RM	TRI 7.5-13.2	Δk_{∞}
16	1963	NAT	SAV RIVER	OXIDE	37	.50 Al	AIR-H ₂ O DOWTH	RM	TRI 9.3	SUCCESSIVE SUBSTIT
17	1962	NAT	CHALK RIVER ZED-II	OXIDE	19	.56 Zr	AIR-D ₂ O ORG	RM	TRI 7.1-14.2	CRIT δ_{28} , CONV RATIO
18	1962	1.1	WALTZ MILLS	OXIDE	19	.43 Al	D ₂ O	RM	SQ 6.5-7.2	CRIT ρ_{25} , δ_{28} , δ_{25}
19	1959	NAT	HANFORD-PCTR	UO ₂ +Pu	19	.50 Al	AIR-D ₂ O	RM	TRI 8.0	PCTR- k_{∞} , f
20	1962	NAT	SAV RIVER-PSE	OXIDE	19	.67 Zr	D ₂ O	25-218	TRI 9.4-11.8	EXPONENTIAL
21	1962	NAT	SAV RIVER-PDP	METAL	2-4 ANNULI		AIR-D ₂ O H ₂ O-ORG	RM	9.3	SUCCESSIVE SUBSTIT
22	1962	NAT	PLATR	OXIDE	2 ANNULI		AIR-D ₂ O H ₂ O-DOWTH	RM	TRI 8.5-13.2	Δk_{∞}
23	1962	.91-1.8%	SGHW WINFRITH	OXIDE	36-37	.31-.55	H ₂ O-D ₂ O POLY	RM	TRI 9.5-10.2	SUBCRIT & CRIT THERMAL FLUX
24	1963	NAT	JUNTA de ENERGIA NUCLEAR, SPAIN	CARBIDE	19	.50 Al	D ₂ O	RM	TRI 8.5-10.5	EXPONENTIAL

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TONG depletion calculations with INGEN-CRUMB. Also made operational was a two-dimensional diffusion code EXTERMINATOR;¹² its running time was found to be comparable to PDQ-2.¹³ EXTERMINATOR offers the possible advantage of including more than one thermal group so that the effects of neutron leakage between fuel regions of different compositions can be evaluated. Another possible advantage is the option of running this code in R- θ geometry, to investigate local or azimuthal power peaking effects.

It was decided to adapt the FORM, THERMOS, INGEN-CRUMB code scheme with a two-region CEPTR to calculate fast group flux weights for present survey calculations. With newer cross sections, this program will be used to analyze many of the experiments listed. Further improvements in these methods, such as more sophisticated burnup and substitution of GAM (with resonance self-shielding for other fuel isotopes) for FORM will be evaluated.

IV. EVALUATION OF EFFORT TO DATE

This work has pointed out the importance of continually evaluating new cross section experiments and theoretical calculations in order to keep code libraries up to date. The importance to reactivity of using a heterogeneous fast fission treatment has been made evident, and the advisability of using more energy groups should be investigated. The necessity of using a spatially-dependent thermal spectrum has been demonstrated, but the importance of aniso-

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12. T. B. Fowler, M. L. Tobias, D. R. Vondy, "EXTERMINATOR - A Multigroup Code for Solving Neutron Diffusion Equations in One and Two Dimensions," ORNL-TM-842 (February 1965).
 13. R. D. Whitesell, "PDQ-2-90, An IBM-709-7090 Code to Solve the Two-Dimensional Few-Group Neutron-Diffusion Equations," International Business Machines Corporation (June 14, 1961).

tropic scattering remains to be investigated. In fuel burnup it was found that two-energy groups give nearly the same results as four groups, but work remains in introducing Pu^{242} and determining the best type of fission product chains to utilize.

V. NEXT REPORT PERIOD ACTIVITIES

Heavy-water-moderated physics experiments will be analyzed using CE's present physics methods with some updated cross sections. A more flexible burnup scheme, such as is possible in the TONG depletion code and its one-and-two-dimensional counterpart ASSAULT,¹⁴ will be evaluated by varying the fuel and fission product chains. Proper physics representation of adjacent fuel bundles having different fuel depletion, and the effects on reactivity when they are fed bi-directionally in steps and burned up, will be initiated during the coming quarter.

FIG 0610-1. OXYGEN MICROSCOPIC ABSORPTION CROSS SECTION vs NEUTRON ENERGY

HWOCR QUARTERLY REPORT	
TITLE: Reactor Physics	SUBTASK NO.: 0650
PROJECT ENGINEER: J. Jacobson	REPORTING PERIOD: Inception through June 1965

I. PROJECT OBJECTIVES

The objective of this task is to establish and verify the calculational techniques required for the physics analyses needed in the design of a HWOCR. Methods of predicting analytically the nuclear characteristics of UC (and UO₂) fueled HWOCR systems will be developed as necessary. Existing codes will be modified as appropriate. These advances are expected to permit a reduction in contingency allowance in HWOCR core design as a result of increased calculational reliability. This reduction is directly translatable into a lower core capital cost and fuel cycle cost.

II. PROGRESS DURING REPORT PERIOD

A. RBU CALCULATIONS*

The RBU Monte Carlo Code ^{1,2} has been used to calculate several HWOCR lattice cells. The basic case considered is the 37-rod UO₂ fuel assembly used as a test case for the HWOCR Physics Methods Comparison. The cell has been calculated using the actual geometry, and also with the fuel assembly homogenized as in the design calculations, in order to study directly the error arising from the homogenization. Analysis of the results is still in progress.

During this work, the limitations implied by the neutron thermalization

* References appear at the end of this subtask section.

model used by the RBU code were more fully revealed. The selection routines used in calculating collisions imply a scattering cross section independent of the relative velocity of the neutron and the target atom. To avoid violating the principle of detailed balance, the cross section used must be consistent with this assumption, instead of being taken directly from experiment. The necessity of avoiding violation of the principle of detailed balance arises because this is the principle which leads to a Maxwellian neutron spectrum in the absence of absorption. However, a constant cross section gives the wrong dependence of mean-free-path on neutron energy, and thus misrepresents the effect on the spectrum of leakage from the moderator into the fuel. These difficulties tend to reduce the value of RBU calculations for thermal neutrons except in evaluating homogenization recipes.

B. CODE DEVELOPMENT AND CODE ASSIMILATION

With a view to the ultimate replacement of the RBU Monte Carlo Code, work was done to extend the applicability of the Monte Carlo resonance-capture code SARA and the Monte Carlo criticality-hazards code SAFE³. SARA was translated into FORTRAN 4, and its input was rewritten for convenience in use in the computing machines. The monitoring capability of SAFE, which was originally restricted to k_{eff} because that is sufficient for criticality-hazards analysis, was extended to provide additional neutron-balance information as desired in a more general nuclear code; also, the convergence was investigated in a series of comparisons with critical experiments, since the convergence requirements are less severe for a criticality-hazards code than for a reactor code. Repetition with SAFE of some of the cases calculated with RBU has been initiated.

For the purpose of improving the speed and convenience with which parameter studies can be carried out, the codes most relied upon in survey work

have been assembled into the comprehensive code THREDES. The thermal-neutron capability of this system is provided by BAM⁴, the fast-neutron capability by FORM^{5,6}. The lattice parameters resulting from BAM and FORM can be used for reactor calculation by the 4-group diffusion code FOG⁷, or for burnup calculations by the two-group point-burnup code KINDLE, both of which codes are also incorporated in THREDES. This system has been made operative at AI and is in production, but work is continuing in polishing, documenting, and extending it.

A similar design system, TONG, recently written at Oak Ridge, has been acquired. Its elementary parts are more elaborate but slower-running than those of THREDES, and include THERMOS⁸ for thermal neutrons and GAM-1⁹ for fast neutrons, as well as burnup and economic capabilities. Significant effort was required to adapt this code package to the AI computing system because of differences from the Oak Ridge computing system. It has now been run successfully both as separate parts and as a complete package. The documentation of this code, both at Oak Ridge and AI, is not yet complete, and this reduces its availability.

The code HERESY I¹⁰ has been acquired from the Argonne Codes Center and made operable at AI. This code is based on the treatment of a reactor lattice as an assembly of neutron sources and sinks representing fuel rods, rather than as an assembly of cells. Its availability will permit later studies to evaluate this alternative method.

C. NEUTRON CROSS SECTIONS

A new FORM multigroup library has been prepared in which the element (or isotopes) H,D,C,O,Al, Cr, Fe, Ni, and Zr cross sections were revised in

accordance with the current information in the AI Evaluated Nuclear Data File¹¹. The Data File itself was modified or supplemented in the following ways:

1) Cr was added to the file.

2) Recent improvements¹² in the data on the (n,α) cross section of O¹⁶ for the higher energies have been included in the new FORM library. The change is indicated below:

<u>The Reaction O¹⁶ (n,α)C¹³</u>			
<u>FORM Group Number</u>	<u>Lower Energy of Group (Mev)</u>	<u>Former Group Cross Section (barns)</u>	<u>Present Group Cross Section (barns)</u>
1	7.78	0.2562	0.0957
2	6.06	0.1540	0.0798
3	4.72	0.0578	0.0390
4	3.68	0.0507	0.0524
5	2.86	0.0000	0.0019

A recent thermal cross-section evaluation for heavy water, acquired from AWRE, Aldermaston,¹³ was the basis of the treatment of thermal neutron scattering by deuterium in the RBU Monte Carlo calculations. As noted in Section A, it was necessary to extract from this information the constant center-of-mass deuterium cross section that, when transformed to the laboratory system, would best represent the data.

The literature was surveyed and references collected in preparation for revising the uranium and plutonium information in the AI Evaluated Nuclear Data File, and for adding xenon and samarium isotopes to this file.

An investigation was initiated of methods for supplying Doppler-broadened resonance cross sections to Monte Carlo codes. The possibilities of using the

Monte Carlo method itself to Doppler-broaden a Breit-Wigner resonance, or of simplifying the analytical calculation and tabulation of the broadened line-shape, as well as the necessity for stochastic selection of the resonance parameters themselves in the unresolved-resonance energy-range, all tend to complicate the choices and decisions in this area.

The age of fission neutrons to the indium resonance energy in D_2O (with and without hydrogen contamination) has been calculated using the Monte Carlo Code TYCHE¹⁴ which also supplies the fourth and sixth spatial moments of the neutron distribution. The results are being analyzed.

D. EVALUATION OF METHODS

The literature reporting reactivity and other measurements on D_2O -moderated systems has been reviewed in preparation for selecting cases against which to test calculation methods. As a result, certain experiments performed by the Chalk River Nuclear Laboratory on the ZED-2 critical facility¹⁵ were chosen for the first run.

In this test, three cases corresponding to three lattice spacings were calculated using the same methods (i.e., the same codes and libraries, and the same treatments of the fast effect, resonance-integral self-shielding, and fuel homogenization for thermal neutrons) as in the HWOCR Physics Methods Comparison and the design-survey calculation that have been performed. The fuel element consisted of a 19-rod bundle of UO_2 fuel rods, of natural enrichment, clad in Zircaloy-2. The fuel pellet diameter was 0.559 in., resulting in a fuel area of 4.66 sq. in. The fuel bundle was inserted in a 0.105-in. thick AI pressure tube. The pressure tube was contained with a 0.056-in. thick AI calandria tube. The coolant material was HB-40 and the moderator was D_2O .

having an average purity of 99.61% D₂O. Three lattice pitches, corresponding to D/U atom ratios of 28, 39, and 57 were analyzed. The results are as follows:

Case	1	2	3
D/U	28	39	57
B ² (calc)	0.000154	0.000219	0.000224
B ² (exp)	0.000121	0.000157	0.000153
K _∞ (calc)	1.031	1.053	1.069
K _∞ (exp)	1.024	1.038	1.047
% error in K _∞	0.7	1.4	2.1

III. EVALUATION OF EFFORT TO DATE

The activities described above represent a favorable start on the program. It is expected that the Monte Carlo calculations with RBU, SAFE and TYCHE will provide more exact models of selected cases and thus be of value in understanding the limitations of other calculation methods; and it is hoped that SAFE can ultimately be made sufficiently convenient and accurate to have continuing value in reactor design calculation. The comprehensive code THREDES has been brought to a point where it can be tested by practice, and by comparison with TONG, as regarding convenience and accuracy. The new FORM library represents a first step in making up-to-date cross sections available for HWOCR calculations; the remaining steps are revision of the TEMPEST library (used by BAM), updating of the U, Pu, Xe, and Sm isotope cross sections in the Data File, and incorporation of these updated data in the several libraries.

IV. NEXT REPORT PERIOD ACTIVITIES

Results of the RBU and SAFE Monte Carlo calculations will be analyzed and compared with other calculations. Documentation of THREDES will be completed.

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REACTIVITY CONTROL

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HWO CR QUARTERLY REPORT

TITLE: Reactivity Control

SUBTASK NO.: 0710

PROJECT ENGINEER: R. L. Furguson

REPORTING PERIOD:

Inception through
June 1965

I. PROJECT OBJECTIVES

The objective of this task is to develop the systems required for reactivity control and adjusting the power distribution in the core. In the HWO CR, excess reactivity must be provided for the following purposes: (1) cold to hot swing (2) 0 to full power (3) equilibrium xenon and (4) the small amount of burnup associated with the refueling cycle. Four categories of control devices may be used to compensate for reactivity changes, adjust power distribution, and regulate power; on-power refueling, absorber columns, booster rods, and moderator poison. On-power refueling is discussed under Task 0810.

The HWO CR Program goal of high fuel utilization and economic power costs presents several problems since the attainment of this goal requires that the design of the reactivity control devices minimize:

- 1) The excess reactivity needed for control purposes
- 2) The loss of neutrons to control devices and associated structural materials while the reactor is operating at power.
- 3) The distortion to the spatial power distribution in the core due to control devices.

In addition, the initial capital costs and replacement costs of the control system must not be excessive.

The HWO CR's size, configuration and temperature distribution present

several mechanical design problems because:

- 1) The space available for vertical control devices above and below the core is quite limited
- 2) The access to actuators and control elements in this space, for maintenance and replacement, is very limited
- 3) The temperature in this space is moderately high
- 4) Although space does not seem as limited at the side of the core, the in-core structure associated with horizontal control devices will be greater than for vertical control devices
- 5) The large size of the reactor can give rise to power oscillations due to changes in xenon concentrations following slight disturbances
- 6) The use of on-power refueling machines at the top and bottom of the reactor precludes the use of control actuators which extend beyond the face of the fuel channel extensions
- 7) The control element travel must equal the core height.

II. PROGRESS DURING REPORT PERIOD

To date, the following types of control devices have been studied for possible use in the HWOCR:

1) Solid-Poison Fluid-Actuated Absorber Columns

This system consists of poison elements, either balls or cylinders, contained in tubes and actuated by the forced circulation of the heavy water moderator.

2) Solution-Poison Fluid-Actuated Absorber Columns

This system consists of a neutron poison dissolved in heavy

water and circulated through control columns within the reactor core. Control can be achieved by varying both the poison concentration and the number of tubes which are filled.

3) Solid-Poison Absorber Columns with External Actuators

This system consists of solid poison elements positioned by mechanical, electrical, or hydraulic actuators.

4) Booster Rods

These consist of enriched fuel elements inserted into the reactor core by mechanical, electrical or hydraulic actuators. An associated cooling system is required.

5) Moderator Poison System

This system consists of a neutron poison dissolved in the moderator. Control is achieved by varying the poison concentration.

Moderator level variation, which is used as a startup and shutdown system in the NPD and CANDU reactor, is not being considered as a means of control, due to the high capital cost of a moderator dump system and operating cost penalties due to the lower power output of the shortened core length. Large changes in moderator height would be required to effect control.

The radial power distribution of the core can be measured by the temperature and flow sensors on each or on groups of fuel channels. This instrumentation system will be supplemented by an in-core power distribution measurement system to determine axial power distributions.

The control methods investigated during this report period were based on the reactivity requirements established in CEND-175, "Large Heavy Water Moderated Reactor Power Plant Conceptual Design and Evaluation",

CE (November 1964). The excess reactivity of the core can be adjusted by the insertion or removal of neutron absorbers distributed throughout the core. The first system studied consists of absorbers installed between lattice positions; the absorbers are vertical so that gravity will assist their insertion into the core.

Based on past studies of heavy water moderated reactors^{1,2} the first control element configuration studied consisted of a column of balls made of 2 wt% natural boron stainless steel. The balls are contained in a Zircaloy tube to establish their position in the core, and are actuated and cooled by a forced circulation of heavy water. Removal of the balls from the core is accomplished by increasing the flow of heavy water until the pressure drop across the column is sufficient to overcome the weight of the element. The column is lifted into a vertical tube above the core. This tube passes through the shield tank and pigtail area. The column is held in the upper position and cooled by the flow of heavy water. To insert the column into the core, the flow of heavy water is decreased until the weight of the column overcomes the pressure drop across the column. These concepts are discussed in detail and shown in the following pages. If direction of fluid flow is reversed the column is forced into the core. Since this type of control element behaves as a fluid with two concentrations, it has been identified as a solid-poison fluid-actuated absorber column.

1. CEND-175, "Organic Cooled, Heavy Water Moderated Reactor Power Plant Conceptual Design and Evaluation", vol I & II, Combustion Engineering (March 1963).
2. Heavy Water Moderated Reactors Evaluation Study, East Central Nuclear Group Inc. (October 1962).

The engineering studies of a similar type control element for light water moderated power reactors, conducted as a two-year program under the joint sponsorship of the European Atomic Energy Community (Euratom) and the United States Atomic Energy Commission (AEC) have been reviewed. This program, completed in June of 1964, had as its objective the design development, analysis and testing of a reactor control system known as the hydraulic ball (Hy-Ball) Control System.³

The basic concept studied during this program utilized small neutron-absorbing columns distributed throughout the reactor core. Each column contains spheres or cylinders of neutron-absorbing material which can be ejected hydraulically from the core or inserted into the core by a combination of gravitational and hydraulic forces. Reactor water is the actuating fluid which maintains a given column of absorbers in either the full out-of-core or the full in-core position. Light water reactor environmental tests simulating all conditions except radiation for many reactor lifetimes (greater than 100,000 cycles), have demonstrated the reliability of the system and the simplicity of its component parts. During these studies circular cross-section tubing was determined to be the best shape for the columns. Both ball and cylinder-loaded columns were tested and found to provide satisfactory performance. The cylindrical shape was found to have the greater nuclear worth for the same size column, and does not restrict the selection of the neutron absorbing material, since clad slugs of any poison material could easily be made into a cylindrical shape. Zircaloy-4

3. SROO-207-30 "Hydraulic Ball (Hy-Ball) Control System Development Program Final Report", General Nuclear Engineering Corporation (June 1964).

was found to give adequate service for the column tubing, as there was no evidence of fretting corrosion after considerable testing. Several types of control valves and hydraulic shock absorbers were analyzed and tested. These components were found to be adequate for the intended purpose.

Using the design parameters described in the plant engineering tasks, a screening study of seven conceptual designs of solid-poison fluid absorber columns for use as two-position devices, and their external systems, has been completed. These concepts include columns which are either "open" or "closed" to the moderator tank environment and have piping systems for the actuating fluid which may be external or internal to the moderator tank. A discussion of each scheme and its advantages and disadvantages follows.

A. ABSORBER COLUMN CONCEPTUAL DESIGN STUDY

1. Method No. 1, Open System Bottom Feed (Figure 0710-1)

Withdrawal of the poison element from the core is accomplished by increasing the flow of D_2O into the bottom of the column. Insertion is accomplished by dumping the flow into a low pressure sink and allowing the poison element to fall into the core. A minimum flow for cooling is maintained.

The bottom feed pipe is located in the space between the moderator tank and lower shield. Therefore, it is difficult to install and maintain. The adjustment of flow for cooling, lift and shutdown is quite critical.

2. Method No. 2, Open System Top Feed (Figure 0710-2)

Both insertion of the poison element into the core and cooling are accomplished by a flow of D_2O into the top of the column. Withdrawal

of the poison element is effected by venting the top of the column to a low pressure source and allowing the moderator pressure to provide the lift flow.

This method requires only one external feed line for each column. The external system is accessible for installation and maintenance; however, it is very dependent on conditions in the moderator tank. If an external system failure occurs, the moderator tank pressure could cause all the columns to be ejected from the core. Such a hazard is inherent in the open system, top feed scheme.

3. Method No. 3, Closed System External Piping (Figure 0710-3)

Heavy water is circulated through the column from top to bottom for insertion and cooling of the poison element. Withdrawal of the poison element is achieved by reversing the direction of circulation. In the event of low flow rate the control elements will drop into the core. An external feed line is required for both ends of each column.

This system and all other closed systems have an advantage over the open systems, in that they are not dependent on conditions in the moderator tank. The system is completely accessible for maintenance. However, it requires a penetration through the lower plate of the moderator tank and the lower shield tank. This adds to the sources of potential leaks in the moderator tank and adds to the complication and cost of the lower shield tank. Repair of the lower piping would require draining the moderator tank. This concept is the most expensive, since it requires a 20-ft. column extension and external piping below the moderator tank.

4. Method No. 4, Closed System with Internal Bottom Feed Line (Model 1 Figure 0710-4)

This method is similar in operation to Method No. 3, except that it does

not require that a penetration be made in the lower moderator tank plate. The bottom feed line is located inside the moderator tank, along the bottom and up the side to the top of the tank. This system requires penetrations only in the top of the moderator tank and only one insulated feed line for each control column. The feed line which runs through the high neutron flux area within the moderator tank is not accessible for maintenance.

5. Method No. 5 Closed System Top Feed Concentric Tube (Model 1 Figure 0710-5)

The column consists of two concentric tubes joined at the bottom, with a hollow-cylinder poison element running in the annulus between the tubes. Lift is accomplished by flowing D_2O down through the central supply tube into the bottom of the annulus containing the annular poison element. The D_2O flow raises the poison element and leaves the column through the discharge tube at the top. Insertion is achieved by forcing D_2O down the annulus, driving the poison element into the core. The D_2O returns to the external system through the central supply tube. The entire system is accessible from the top of the reactor for installation, maintenance or replacement. Therefore, the installation of the columns into the reactor is simple.

Use of a hollow-cylinder poison element requires a central D_2O supply tube which is difficult to support. Binding of the long poison element due to close tolerances is possible. This system requires almost twice as much parasitic material in the active core area.

6. Method No. 6 Closed System Top Feed Concentric Tube (Model 2 Figure 0710-6)

This method is similar to Method No. 5 in that it uses two concentric tubes for flow channels; however, the central tube has been enlarged to allow the poison element, in the form of steel balls or short cylinders, to be placed inside the tube. Lift is produced by flowing D_2O down the annulus

and up through the central supply tube, raising the poison element. Flowing D₂O down the center tube drives the poison element into the core.

The entire system is accessible from the top of the reactor and the assembly of columns into the reactor is simplified. The central tube may be simply supported by a spacer wire in the annulus. Close tolerances are not required.

7. Method No. 7 Closed System Internal Bottom Feed Line (Model 2 Figure 0710-7)

This method is the same as Method No. 4 except that it is designed for use with a fluid-interaction diverting valve. This arrangement has the simplest external system of any method considered, and an extremely short valve switching time without water hammer. However, the pumping power requirements are twice those of any other system proposed. In addition, it requires an internal bottom feed line, an ejector pump in the bottom feed line inside the moderator tank, and an integral valve in the top of each column.

B. CONCEPTUAL DESIGNS COMPARISON

A comparison of these concepts relative to cost, core structure and external system required, ease of maintenance and installation, and reliability is shown in Table 0710-1. All of the systems can be designed to fit between the fuel channel positions. The space available for control element actuators above the core is shown in Figure 0710-8.

A preliminary analysis of the thermal-hydraulic requirements of the concentric type column was made based on the experimental data on a similar device*. These both show that an 18-ft column of 1/2-inch spherical absorbers contained in a 9/16-inch ID tube requires a flow of 0.68 gpm to lift the

* SR00-207-30 "Hydraulic Ball (Hy-Ball) Control System Development Program Final Report", General Nuclear Engineering Corporation (June 1964)

TABLE 0710-1

COMPARISON OF SOLID-POISON FLUID ABSORBER COLUMN CONCEPTS

METHOD NO.	1	2	3	4	5	6	7
DESCRIPTION	OPEN SYSTEM BOTTOM FEED	OPEN SYSTEM TOP FEED	CLOSED SYSTEM WITH EXTERNAL PIPING	CLOSED SYSTEM INTERNAL BOTTOM FEED LINE MOD-1	CLOSED SYSTEM CONCENTRIC TUBE MOD-1	CLOSED SYSTEM CONCENTRIC TUBE MOD-2	CLOSED SYSTEM INTERNAL BOTTOM FEED LINE MOD-2
SCHEMATIC DIAGRAM FIGURE NO.	0710-1	0710-2	0710-3	0710-4	0710-5	0710-6**	0710-7
POSSIBLE POISON CONFIGURATIONS	SPHERE or CYLINDER	SPHERE or CYLINDER	SPHERE or CYLINDER	SPHERE or CYLINDER	TUBE	SPHERE or CYLINDER	SPHERE or CYLINDER
RELATIVE PREFERENCE (LEAST COST MAX)	1*	1	5	2	4	2	3
EXTERNAL SYSTEM REQUIREMENTS	1	1	2	2	2	2	3
EASE OF MAINT.	4	1	3	2	1	1	2
SIMPLICITY OF SYS. & EASE OF INSTALL.	4	1	3	2	1	1	2
RELIABILITY ESTIMATES	3	3	2	2	3	1	3
CORE STRUCTURAL MATERIAL (Zr)	1	1	1	1	2	1	1

* THE NUMBERS INDICATE AN ORDER OF PREFERENCE FOR THE VARIOUS SYSTEMS WITH THE UNIT 1 INDICATING THE MOST PREFERABLE

** FLOW DIAGRAM FOR THIS SYSTEM IS SHOWN IN FIGURE 0710-9

column from the core in 10 seconds. The column can be held in the out-of-core position by a flow of 0.26 gpm.

While the column is held above the core, little heat is generated in the column; however, the column is surrounded by uninsulated fuel channels extensions and pigtails containing reactor coolant at temperatures ranging from 500 to 750°F. The D₂O in the control column is insulated by a stagnant gas-gap within the column housing. Preliminary calculations of the heat transferred to the D₂O as it passes through the hot jumper tube area, indicate the water temperature will increase 7.3°F. Considerable heat is generated in the absorbing material when it is in the core and the core is operating at significant flux levels. If boron is the neutron absorbing material, the heat generation rate in a single column worth of 0.3 mk in a reactor producing 2850 Mwt by fission, is approximately 715 BTU/minute.

If it is assumed that the entire heat generated is transferred directly to the coolant with no heat dissipated radially through the column walls, a flow of 1 gpm (9.17 lb/min) will have a rise in the coolant temperature of approximately 78°F. From Figure 0710-9, a flow of 1 gpm produces a pressure drop of 183 psi across an 18-ft column.

The scram time vs scram distance in 70°F water, for a 9-ft stack of 1/2-inch-diameter stainless steel balls in a 9/16-inch-diameter tube, using a straight hopper tube and the effect of the labyrinth buffer serving the function of a dash-pc was obtained experimentally. These data were combined and extrapolated to the HWOCR conditions; i.e. an 18-ft column with a scram

* SR00-207-30 "Hydraulic Ball (Hy-Ball) Control System Development Program Final Report", General Nuclear Engineering Corporation (June 1964)

distance of 18-ft, as shown in Figure 0710-10. The minimum flow required during scram is 1 gpm. If the pressure differential maintained across the column by the external system to produce a flow of 1 gpm with the column at rest is used to scram the column, the flow increases during the 1 to 2 sec scram interval to 2.3 gpm per column.

A preliminary design of a flow system for the closed system concentric tube absorber columns is shown in Figure 0710-11. It is to provide:

- 1) High pressure D₂O for cooling of absorbers while in the in-core position
- 2) High pressure D₂O for rapid insertion (scram) of absorbers
- 3) Low pressure D₂O for withdrawal of the absorbers from the reactor core
- 4) Low pressure D₂O for holding the absorbers in the out-of-core positions.

The D₂O is pumped from the moderator system into a surge tank where it is maintained at 177 psig. Two pumps, one a standby, with a minimum capacity of 200 gpm are provided for this operation. For scram operations and in-core cooling of absorbers, high pressure D₂O from the surge tank is directed through the 4-way solenoid valve to the top of the inner tube of the absorber column. The return flow is from the top of the outer tube of the absorber column through a check valve and the 4-way solenoid valve into the moderator system. For lifting the absorber column to the out-of-core position, high pressure D₂O from the surge tank is directed by the 4-way solenoid valve through a pressure regulating valve to the top of the outer tube of the absorber column. The return flow is from the top of the inner tube through the 4-way solenoid valve into the moderator system. The moderator

surge tank pressure level is maintained by a high pressure helium system. Isolation valves are provided on the inlet and outlet lines of the solenoids and pumps, to permit removal for maintenance.

The concentric tube configuration has been designed to accommodate the length changes produced in the various tubes due to differential thermal expansions. Preliminary analyses indicate that the axial clearance between the end of the column and the bottom moderator tank plate increases 0.590 inch when the reactor is operating at power. The concentric elbow on the column top was designed to accommodate a differential temperature of 100°F between the inner and outer tube without exceeding a safe stress level.

The conceptual design of a mechanical device for use with the solid-poison fluid-actuated absorber column, to permit the poison element to be positioned axially, is shown in Figure 0710-12. Heavy-water flow admitted to the bottom of the control column forces the poison element up the column and against the upper stop. Positioning of the poison element is achieved by changing the axial location of this upper stop with a ball screw mechanism. The power to the ball screw is supplied by either an electric or hydraulic motor. Due to space limitations, the actuator is mounted in a 9-ft horizontal extension at the top end of the column. The actuator is capable of producing up to an 8-ft variation in height at a rate of 10 ft/min.

A literature search revealed that a control system utilizing a soluble poison solution, circulating in a tube through a reactor, had been designed and a prototype system successfully tested on the swimming pool reactor AVOGADRO-RS-1 of the Soluggia Nuclear Center*. From the available data it

* Stability Analysis and Dynamic Performance of a Chemical System
Operated for the Automatic Fine Control of a Nuclear Reactor,
Ambroeni, G. C. et al, (May 1964)

can be concluded that the control system seemed to have satisfactory dynamic characteristics with regard to fine control. Other problems, such as corrosion, radiation damage, poison recovery, and poison deposition on piping were not discussed.

III. EVALUATION OF EFFORT TO DATE

The use of solid-poison and solution-poison fluid absorber columns appears feasible for the HWO CR application from a mechanical design perspective. In addition to these systems, moderator poisoning and booster control rods are under investigation. As the conceptual design effort progresses and the control requirements are further defined, system specifications will be updated and cost analyses conducted.

IV. FUTURE REPORT PERIOD ACTIVITIES

Screening studies of alternate concepts will be continued. Selected candidate designs will be brought to the design layout stage. Combinations of the candidate systems will be studied to determine a reference system. Design of the in-core power distribution monitoring systems will be initiated.

The stress and thermal-hydraulic analysis required to establish the external system and in-core structure material requirements of the reference system(s) will be initiated.

FIG 0710-1. OPEN SYSTEM BOTTOM FEED

FIG 0710-2 OPEN SYSTEM TOP FEED

FIG 0710-3 CLOSED SYSTEM WITH EXTERNAL PIPING

FIG 0710-4. CLOSED SYSTEM WITH INTERNAL BOTTOM FEED MOD-1

FIG 0710.5. CLOSED SYSTEM CONCENTRIC TUBE MOD-1

FIG 0710-6. CLOSED SYSTEM TOP FEED CONCENTRIC TUBE MOD-2

FIG 0710-7. CLOSED SYSTEM INTERNAL BOTTOM FEED LINE MOD-2

FIG 0710-8. SPACE AVAILABLE ABOVE CORE FOR CONTROL DEVICES

FIG 0710-9. DIFFERENTIAL PRESSURE vs FLOW (TAKEN FROM REF 3, p.7)

FIG 0710-10. SCRAM TIME vs DISPLACEMENT (TAKEN FROM REF 3)

FIG 0710-11. SCHEMATIC OF THE FLOW CIRCUIT FOR THE CLOSED SYSTEM CONCENTRIC TUBE ABSORBER COLUMN

FIG 0710-12. SOLID-POISON FLUID ABSORBER COLUMN-HORIZONTAL POISONER

ON-POWER REFUELING

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HWO CR QUARTERLY REPORT	
TITLE: On-Power Refueling	SUBTASK NO.: 0810
PROJECT ENGINEER: R. L. Ferguson	REPORTING PERIOD: Inception through June 1965

I. PROJECT OBJECTIVES

The goal of the fuel management program is to achieve a high uniform burn-up of all fuel assemblies with low enrichment in the feed fuel, while obtaining a maximum power output from a given reactor core volume. The on-power refueling system comes closest to achieving these goals if it can meet the following requirements:

- 1) Maintain a relatively constant spatial power distribution in the core and minimize the maximum-to-average power using a minimum of parasitic absorbers as burnup, conversion, and fission product build-up occurs.
- 2) Minimize plant downtime for the specific purpose of refueling.
- 3) Maintain the excess core reactivity at a minimum, to reduce the loss of neutrons to control absorbers.

The objective of this task is to develop an on-power refueling system which will move fuel through the core in accordance with the fuel management program required to attain these goals.

The HWO CR's size, configuration, temperature, coolant, and goals of high fuel utilization and economic power costs present several unique problems. These include:

- 1) The large number of fuel channels and fuel assemblies reduces the time available for maintenance between refueling operations.

2) The vertical arrangement of fuel channels, where coupled with the bi-directional refueling and coolant flow, produces different force considerations in alternate channels.

3) The combination of the high temperature and poor lubricity of the reactor coolant produces a very severe environment for the complex mechanism of the refueling machine's tool post. For similar reasons, the equipment required to change the fresh fuel from storage to the reactor environment and to change the spent fuel from the reactor to storage environment becomes complex.

4) The length of the fuel channel extensions requires long travel of the machine's tool post.

5) The high performance design of the fuel assembly and fuel channel assemblies requires that a minimum of force be exerted by the machine on these components. The relative travel of these components to each other should be minimized. The fuel assembly travel within the fuel channel should be minimized.

6) Since maintenance will be required, care must be exercised to prevent activation of the machine due to contamination, or exposure to activating neutron flux. In addition, the components must be designed for ease of maintenance; remote maintenance must be possible.

II. PROGRESS DURING REPORT PERIOD

As the result of past work*, bi-directional refueling schemes have been selected for consideration in the initial studies. Because the fuel in adjacent

*CEND-175, "Organic Cooled, Heavy Water Moderated Reactor Power Plant Conceptual Design and Evaluation," Vols. I and II, Combustion Engineering (November 1964).

fuel channels is moved through the core in opposite directions, the average fuel composition of adjacent fuel channels in each axial zone is kept approximately constant; therefore, the reactivity and power distribution remain nearly constant as burnup, conversion, and fission product build up occur. By adjusting the rate at which the fuel progresses through a channel the fuel composition in the radial zones can be adjusted to flatten the radial power distribution of the core.

The design of the refueling machine for the Douglas Point Plant by AECL* has been reviewed and a prototype machine has been developed and tested. At present, the machines to be installed on the reactor are being assembled and tested under simulated reactor conditions.

Design of an alternate refueling machine for application on the Douglas Point Plant by AMF† has been reviewed. The AMF work evaluated commercial parts and materials for refueling machine applications. However, the functioning of the machine and its auxiliary systems was not tested completely. The components and equipment used in this development program have been transferred to the HWOOR Program. Components which perform similar functions to those required by the HWOOR machine may be used for preliminary testing of mechanisms to operate in organic fluids.

The design of the refueling machine for the NPD has been reviewed. By

*F. S. Hummerl, B. C. Stonehill, C. E. Reynon, "Refueling and Fuel Handling for CANDU Reactor," Dilworth, Secord, Meagher & Associates, Ltd.

† AMF On-Power Refueling System for the Douglas Point Plant:
AMF-388-TR-1 Component Feasibility Test Report (February 1962)
AMF-388-TR-2 Task 1 Completion Report, Vols. I and II (July 1962)
AMF-388-TR-3 Adapter Valve Development Report (February 1963)
AMF-388-TR-4 Prototype Refueling Head, Final Development Report (December 1964)

Spring 1965, this machine had been used to insert 1488 fuel bundles into the reactor; 300 of these fuel assemblies were inserted with the reactor operating at full power.

During the refueling operations, these machines perform similar functions to those required for the HWOCR refueling machine. Some components, such as coupling assembly and instrumentation system, may be used with modification. However, the head and assemblies cannot be used with the HWOCR, because this reactor has a larger number of fuel channels, a smaller lattice spacing, vertical fuel channels with longer fuel channel extensions, higher temperature, and less desirable coolant characteristics (i.e., lubricity, radiation damage, and residues), and more brittle fuel cladding material.

A. REFUELING MACHINE PERFORMANCE STUDY

Eight methods of obtaining a bi-directional flow of fuel in a vertical reactor were studied to select feasible concepts which could be used to define the performance requirements of the refueling machine. This screening study consisted of selecting typical reactor parameters (Table 0810-1), defining the refueling cycle, and evaluating the performance and problems of each method. The coolant flow in the machine is assumed to be at a constant temperature throughout a refueling cycle. It is further assumed that the shield plug is not separate from the fuel support structure. For this study the refueling cycle was assumed to be:

- 1) Loading and Discharge - Five fresh fuel assemblies are loaded into one machine, five spent fuel assemblies are discharged from the other machine.

TABLE 0810-1

REACTOR PARAMETERS FOR BI-DIRECTIONAL
REFUELING PROCEDURE EVALUATIONS

FUEL CHANNEL ASSEMBLY ORIENTATION	VERTICAL
FUEL COLUMN LENGTH	200 IN
FUEL ASSEMBLY LENGTH	40 IN.
FUEL ASSEMBLY DIAMETER	3.88 IN.
FUEL ASSEMBLY CROSS SECTION AREA	12 IN. ²
FUEL ASSEMBLY WEIGHT	90 LB
FUEL CHANNEL ID	4.0 IN
FUEL CHANNEL LENGTH	610 IN
FUEL CHANNEL ΔP	125 PSI
FUEL ASSEMBLY ΔP	25 PSI
FUEL FLOW	BI-DIRECTIONAL
COOLANT FLOW	SAME DIRECTION AS FUEL FLOW

2) Locating - Each machine moves from the loading or discharge area to the desired fuel channel.

3) Coupling - Each machine couples to the fuel channel, tests the seal, and removes and stores the seal cap of the fuel channel.

4) Normal Refueling - The machines remove and store the locking rods, remove and store one spent fuel assembly, add one fresh fuel assembly, and replace the locking rods.

5) Uncoupling - The machines replace the seal caps, perform a leakage test, and uncouple from the fuel channels.

6) Repeat - Steps 2 through 5 are repeated for each of four other fuel channels.

7) Homing - The machines return to the loading and discharge areas.

The eight methods considered differ in the manner in which the fuel column is supported and moved and the method of separating the spent fuel assembly from the fuel column during the normal refueling step of the cycle. The method sequences are described below.

1) Method No. 1

a) The fuel is lightly held between the tool posts of machines located at each end of the fuel channel.

b) The fuel column is moved out of the core to the machine at the discharge side.

c) A spent assembly is removed from the column.

d) The fuel column is then moved through the core to the charge side and a fresh fuel assembly added to the column.

e) The entire column is then centered in the core.

2) Method No. 2

- a) The fuel is supported on a central structural rod.
- b) The entire column is pulled into the machine at one end of the reactor.
- c) The central rod is removed and the fuel column rearranged; i.e., the spent assembly is stored, the four (4) irradiated assemblies are repositioned, and a fresh fuel assembly is added.
- d) The central rod is replaced and the fuel column installed in the core.

3) Method No. 3

- a) The fuel assembly is grappled at its end plate.
- b) A fresh assembly is added at the charge end of the column.
- c) The spent fuel assembly is separated from the column in the pressure tube extension and removed by the machine grapple on the discharge end of the fuel channel.

4) Method No. 4

- a) The fuel column has a fixed support on the upstream end of the column; the support at the downstream end is transferred to the tool post ram.
- b) The entire fuel column is moved into the refueling machine by the coolant pressure.
- c) The column is stored in a magazine and then replaced in the core; during this operation, the spent assembly is left in the storage magazine, and the four irradiated assemblies are replaced to effect the movement of one fuel position, and a fresh assembly is added to the column.

5) Method No. 5 The fuel motion is similar to Method No. 4. However, the ram of the machine segment on the downstream end forces the fuel into the machine upstream end.

6) Method No. 6

a) A hydraulic ram installed in the pressure tube extension at the discharge end of the fuel channel forces the fuel column into a latch-can at the charge end.

b) The latch-can deposits the fuel in a storage magazine.

c) The latch-can then picks up the fuel from the storage magazine in the proper order to effect the progression of the four irradiated fuel assemblies, and adds a fresh fuel assembly to the column.

7) Method No. 7

a) A fresh fuel assembly is added to the column by the tool post of the machine at the charge end of the fuel channel. As this assembly is installed in position, the entire fuel column is advanced the length of one fuel assembly. This pushes the spent fuel assembly into a receiver on the machine tool post at the discharge end of the fuel channel.

b) The spent assembly is then removed from the fuel channel.

8) Method No. 8 The fuel movement is similar to Method No. 1. The machine consists of three separate tool posts, one each for removal of each piece of fuel channel hardware, i.e., end seal, shield plug, and fuel assembly.

a) An adapter valve is first clamped to each end of the fuel channel.

b) The tool post tubes are then connected in sequence to perform the operations required.

Schematic drawings (Figures 0810-1 to 0810-6) show the overall nominal dimension and the step-by-step functions required for each method. Schematic diagrams of Methods No. 2 and No. 8 were not prepared, because these methods were found to be less desirable than the other methods. The design problems associated with the machine requirements of Method No. 2, i.e., to remove a central rod from a column of fuel assemblies, and to rearrange and mount the fuel assemblies on a central rod, seems to be difficult to resolve. The number of sealing operations required for Method No. 8 slows the refueling cycle excessively. In addition, the sealing operations increase the maintenance problems associated with seal replacement and increase the leakage of organic coolant.

Each scheme was then studied to determine the following:

- 1) Refueling Time - The time to perform one complete refueling cycle consisting of replacing one fuel assembly in each of five fuel channels, and unloading the spent fuel from the refueling machine and loading new fuel for the next cycle.
- 2) Tool Post Extension - The distance from the end of the fuel channel to the point of maximum tool post extension.
- 3) Tool Post Exposure - The distance from the edge of the fuel column to the point of maximum penetration of the tool post.
- 4) Tool Post Load - The tool post load due to weight of fuel and hydraulic force of coolant flow.
- 5) Fuel Wear Factor - The total travel of one fuel assembly in the fuel channel during its residence in the core
- 6) Pressure Tube Wear Factor - Total travel of one fuel assembly in the process tube during its total residence in the core.

7) Core Disturbance Factor - This factor is an indication of the disturbance of the neutron flux and the coolant and fuel temperatures in the fuel channel being serviced. The magnitude of the disturbance is indicated by the number of fuel assemblies moved from the core at one time. The frequency of the disturbance is indicated by the number of disturbances per cycle.

8) Machine Heat Load Factor - The heat to be removed by the machine's cooling system is indicated by the number of partially irradiated fuel assemblies taken into the machine at one time.

9) Additional Fuel Assembly Structural Material - This indicates if additional structural material is required in the fuel assemblies for the specific purpose of on-power fueling.

Two arrangements for fuel column support were considered. One arrangement supports the fuel between long locking rods, consisting of five 40-in.-long segments. The other arrangement supports the fuel between single-piece 40-in.-long, locking rods. The results of this study are shown on Table 0810-2.

B. REFUELING MACHINE PROCEDURES SELECTION

The results of the study discussed above were also compared on qualitative factors, such as design complexity, anticipated problems, and maintenance requirements. Methods No. 3, No. 7, and No. 1 have been selected as the most desirable procedures, on the basis of the results listed on Table 0810-2 and the arguments presented below:

1) Methods No. 3 and No. 7 - These methods are similar in most respects, differing only in the mode of grappling the fuel. Further work is required to determine the feasibility of fuel grapples and in-channel

TABLE 0810-2

REFUELING SYSTEM CHARACTERISTICS

Method No.		1	2	3	4	5	6	7	8
Figure No.		0810-1		0810-2	0810-3	0810-4	0810-5	0810-6	
	Unit								
<u>Refueling Time</u>									
5 pc Locking Rod	hr	4.05	4.67	3.87	3.55	3.05	4.67	3.87	---
1 pc Locking Rod		3.05	3.33	2.55	3.05	2.38	3.33	2.55	6.0
<u>Tool Post Extension</u>									
5 pc Locking Rod	in.	210	205	205	5	410	165	205	---
1 pc Locking Rod		370	205	205	165	570	165	205	370
<u>Tool Post Exposure</u>									
5 pc Locking Rod	in.	5	None	None	None	205	None	None	---
1 pc Locking Rod		165	None	None	None	265	None	None	165
<u>Tool Post Load</u>	lb	---	1200 Lift	240 Lift	---	---	1200 Lift	240 Lift	---
		1300 Push	---	1300 Push	1200 Push	1300 Push	1200 Push	1300 Push	1200 Push
<u>Magazine Diameter</u>									
5 pc Locking Rod	in.	33.7	24.1	33.1	43.2	43.2	43.2	33.7	---
1 pc Locking Rod		26.0	20.3	26.0	35.6	35.6	35.6	26.0	---
<u>Fuel Wear Factor</u>	in.	4710	3050	610	3050	3050	3200	700	4710
<u>Pressure Tube Wear Factor</u>	in.	2640	1200	240	1200	1200	1200	460	2640
<u>Core Distrubance Factor</u>									
Magnitude	--	5	5	0	5	5	5	0	5
Frequency		2	1	0	1	1	1	0	2
<u>Machine Heat Load Factor</u>	--	1	5	1	5	5	5	1	1
<u>Additional Structural Material Required On Fuel Assembly</u>	--	No	Yes	Yes	No	No	No	No	No

0810-11

fuel support arrangements which do not require additional structural material on the fuel assembly.

2) Method No. 1 - This method is not as desirable as Methods No. 3 and No. 7, since it requires that the entire fuel column be removed from the core area to add a fresh assembly. In addition, the fuel column must be shifted through the core area to the opposing segment of the machine for removal of the spent fuel assembly. However, the space available for fuel manipulation is not as limited in the machine as it is in the fuel channel.

3) Method No. 2 - Extra structural material is required in the fuel assembly. Fuel manipulation required in the machine is complex.

4) Method No. 4 - There is no positive force maintained on the fuel column. Hot coolant from the reactor flows into the machine.

5) Method No. 5 - This system is less desirable than others as shown in Table 0810-1.

6) Method No. 6 - The cost of hydraulic cylinders for each channel is excessive. The difficulty in maintaining these hydraulic cylinders will increase operating costs.

7) Method No. 8 - The number of couplings required to change one fuel assembly is excessive.

C. REFUELING CYCLE TIME CONSIDERATIONS

The above discussion of comparative method advantages and disadvantages is followed by consideration of time requirements in the refueling cycle. The time available for the refueling cycle as a function of burnup, specific power, peaking factor, and number of fuel channels is:

$$t_a = \frac{t_{avg}}{ZP}$$

where: t_{avg} - Time available between successive opening of the average channel
 $t_{avg} = \frac{\text{Burnup} \times (\text{fraction of fuel column changed during refueling cycle})}{\text{Specific Power}}$
 ZP - Zone factor - (peaking factor of radial zone) x (number of channels in zone)

This expression assumes uniform burnup of spent fuel.

The number of fuel columns changed per day is the reciprocal of t_a . The rate at which fuel channels require refueling is given in Table 0810-3 for three enriched fuel reactors containing 600, 1400, and 3400 fuel channels with the radial power distribution shown.

The results in Table 0810-3 indicate that one refueling system of the type selected in the screening studies can be used for the 1000-Mwe and 300-Mwe plants. Its use on an 800-Mwt reactor may be possible if downtime for maintenance can be minimized.

D. BASIC MACHINE ELEMENT REQUIREMENTS

From the screening studies, the basic machine elements required to perform the refueling functions were defined to be:

- 1) A coupling assembly to connect the machine to the fuel channel assembly.
- 2) A tool post assembly to remove the end seal, shield plug, and spent fuel from the fuel channel. This assembly will also insert a fresh fuel assembly and reinstall the shield plug and end seal.
- 3) A storage assembly for fresh or spent fuel, and fuel channel hardware.

TABLE 0810-3

TIME AVAILABLE FOR REFUELING CYCLE

FRACTION OF FUEL COLUMN CHANGED/CYCLE				1/5	1/5	1/10
BURNUP MWD/TONNE				20,000	20,000	20,000
SPECIFIC POWER MW/TONNE				15	20	20
T _{AVERAGE} (DAYS)				265	200	100
ZONE	RADIAL PEAKING FACTOR	NO. OF CHANNELS	ZONE FACTOR	CHANNEL PER DAY		
I	1.35	68	92	0.347	0.458	0.917
II	1.28	84	108	0.408	0.540	1.076
III	1.15	116	135.5	0.503	0.676	1.333
IV	1.04	156	162	0.614	0.813	1.612
V	0.55	176	97	0.367	0.486	0.971
TOTAL	----	600	----	2.239	2.973	5.909
I	1.35	158	213	0.807	1.065	2.13
II	1.28	196	251	0.952	1.265	2.50
III	1.15	270	310	1.176	1.540	3.12
IV	1.04	364	378	1.429	1.885	3.84
V	0.55	412	227	0.855	1.136	2.27
TOTAL	----	1400	----	5.219	6.891	13.86
I	1.35	372	502	1.886	2.50	5.00
II	1.28	476	610	2.330	3.03	6.25
III	1.15	655	753	2.860	3.70	7.69
IV	1.04	885	920	4.170	4.55	9.09
V	0.55	1012	557	2.100	2.78	5.57
TOTAL	----	3400	----	13.246	16.56	33.60

4) A refueling head support assembly to connect the refueling head, i.e., coupling, tool post, and storage assemblies, to its positioning carriage. This assembly will also position the refueling head on the main carriage.

5) A main carriage assembly to transfer the load of the refueling head to the main support structure and provide a means of positioning the refueling head.

6) An environmental control system to maintain the temperature and pressure of the machine

7) An instrument and control system to direct and monitor the functioning of the machine and its components

8) Special support equipment to transfer fresh and spent fuel to and from the machine, to perform routine maintenance upon the machine, and to perform disassembly for overhaul.

Preliminary design specifications have been prepared for each of these components, as discussed in the following pages

1. Coupling Assembly

A study of the coupling assembly functions and environmental conditions established that the coupling seal must achieve an effective seal between the snout of the refueling machine and the flange of the fuel channel each time a coupling is made. The organic coolant temperature of 750°F prevents the use of an elastomeric seal. The use of a metallic seal requires:

- 1) A clamping force of approximately 5000 lb
- 2) A finish of 32 RMS or better on the sealing surfaces
- 3) Sealing surfaces maintained free of foreign material.

4) Frequent replacement due to the limited cyclic life of the seal.

5) The coupling operation must not exert a tensile force on the fuel channel in excess of 1000 lb. Several seal manufacturers have been surveyed for suitable components.

The conceptual design of a commercial coupling assembly and seal modified for remote operation has been evaluated (Figure 0810-7). The metallic seal is 1/8-in.-thick and is capable of withstanding a minimum of 1000 cycles without detectable leaks. Test reports from the vendor indicate that the seal could make a leak-free connection under the service conditions of 315 psig and 750°F. The coupling consists of a gear-driven collar mounted on the snout of the refueling machine, a ring gear mounted on each fuel channel, and the metallic seal pinned to the face of the snout. The gear-driven collar rotates the ring gear until the seal is sufficiently compressed for a leak-free connection. The disadvantage is interference of the ring gear of an adjacent fuel channel with the collar while the connection is being made. Ring gears for each fuel channel are expensive, replacement of these gears is difficult, and the ring gear at the top of the reactor is susceptible to collection of crud.

Three methods of installing thin metallic seals, namely, a snap ring, threaded seal, and seal screws, are shown in Figure 0810-8. The disadvantage of these methods is the difficulty of removing the seal due to formation of crud. The vendor estimates a 50-cycle life to these seals for 315 psig and 750°F service.

Eight methods for clamping the refueling machine to the fuel channel have been investigated. The method of operation and the relative advantages and disadvantages of each are outlined in Table 0810-4. Of the designs investigated,

TABLE 0810-4
COMPARISON OF SEAL CLAMPING METHODS

FIGURE NO.	ACTIVATION METHOD	NO OF JAWS	COMMENTS
0810-11	ELECTRO-MECHANICAL RACK & PINION DRIVEN CAM	4	NO PROVISION FOR ADJUSTMENT* REQUIRES PRECISION MACHINING AND CLOSE TOLERANCES
0810-12	ELECTRO-MECHANICAL RACK & PINION DRIVEN PUSHRODS	4	INSUFFICIENT CLAMPING FORCE
0810-13	ELECTRO-MECHANICAL RACK & PINION DRIVEN CAM	4	REQUIRES PRECISE POSITIONING. SUBJECT TO FREEZING DUE TO CRUD CONTAMINATION NO PROVISION FOR ADJUSTMENT*
0810-14	ELECTRO-MECHANICAL RACK & PINION DRIVEN CAM	4	NO PROVISION FOR JAW RETURN NEGATIVE MECHANICAL ADVANTAGE NO PROVISION FOR ADJUSTMENT*
0810-15	ELECTRO-MECHANICAL RACK & PINION DRIVEN PUSHRODS	4	NO PROVISION FOR ADJUSTMENT* REQUIRES THREE (3) PIVOTS
0810-16	ELECTRO-MECHANICAL RACK & PINION DRIVEN PUSHROD	4	NO PROVISION FOR RETURNING JAWS NO PROVISION FOR ADJUSTMENT*
0810-17	ELECTRO-MECHANICAL RACK & PINION DRIVEN CAM	4	NO PROVISION FOR ADJUSTMENT* REQUIRES PRECISION MACHINING AND CLOSE TOLERANCES
0810-10	MODIFIED CANDU LINKAGE	3	PROVIDES AN EVEN CLAMPING FORCE PROVIDES INCREASED BEARING AREA

* THESE SYSTEMS HAVE NO MEANS OF PROVIDING AN EVEN CLAMPING FORCE OVER THE ENTIRE CIRCUMFERENCE OF THE FLANGE AND SEAL

a modification of the clamp used on the CANDU machine (Figure 0810-9) is the most desirable. These modifications permit the clamp to be used on a vertical refueling machine and to clamp on fuel channels located on a 9.5-in.-square lattice. In this design, a four-segment ring is closed around the fuel channel extension; the refueling machine is then pulled toward the fuel channel, compressing the seal. In this manner, the compression force on the seal is limited to the end of the fuel channel and the force exerted on the fuel channel supports is limited to the amount of force required to move the refueling head against its counterbalance.

2. Tool Post Assembly

The design of the machine's tool post is dependent on the manner in which the fuel column, the end seal, and the shield plug are mounted in the fuel channel. A mounting arrangement for the end seal and shield plug similar to that used on the CANDU has been assumed. Investigations of fuel support arrangements have been made for vertical fuel channels with bi-directional flow.

In the up-flow fuel channels, the fuel column must be supported against gravity forces during low flow conditions, and for forces due to coolant flow during normal flow conditions. Since these forces act in opposite directions, the fuel column must be supported at both ends to prevent fuel movement during changing flow conditions. A variable-length spacer rod is required to compensate for changes in fuel column length due to stack-up of dimensional tolerances and temperature changes during startup and refueling operations. The conceptual design of this spacer has been completed. It is a spring-loaded device which can compensate for approximately 3/4-in. difference in linear expansion between the fuel column and the fuel channel. The spring is preloaded to 570 lb to

support the static weight of the fuel column. At operating temperature, this load increases to 682 lb. Inconel X has been selected as the spring material due to the high operating temperature of 750°F.

Conceptual designs of two methods of accomplishing the selected refueling procedure have been prepared. Each consists of a tool post and associated fuel assembly and plug handling mechanism, and the latches required to support the fuel in the reactor.

One of these tool post and latching mechanism concepts (Figure 0010-17) consists of a primary ram with a permanently attached latch- an and a secondary ram with a permanently attached head assembly. The primary ram will have four hydraulically operated telescoping stages and be capable of extending approximately 294 in. from the refueling machine to the latch position 40 in. above the reactor core. The secondary ram will extend 60 in. and be capable of acting independently of the primary ram. Both rams will be capable of exerting an axial force of 3000 lb.

During a refueling operation, the primary and secondary ram will be extended to permit the head assembly to grasp and unlock the closure plug. When this operation has been completed, the primary ram will withdraw and deposit the closure plug into the magazine. The primary ram must then extend to the latch position where the spacer rod will be unlatched and pushed into the latch-can by the opposing secondary ram. The primary ram must then withdraw, deposit the spacer rod in the magazine, receive and latch onto a fresh fuel assembly, and transfer this fuel assembly into the fuel channel. When the fuel assembly is positioned at the latches, the secondary ram will be extended, pushing the fresh fuel assembly past the latches. The secondary and primary rams will then with-

draw to the magazine, grasp the spacer rod, and, in a similar manner, reseal the spacer rod in the fuel channel. Finally, the rams and head assembly will remove the closure plug from the magazine and replace and lock the closure plug in the fuel channel.

The fuel channel latch mechanism for this design consists of six locking fingers attached to a spring-loaded slider. Opening of these latches is accomplished in the final $1/2$ in. of the primary ram stroke. Contact is first made with the fuel column by the secondary ram head assembly. The first $3/4$ in. of a stroke compresses the spring-loaded ram head and positions in the latch-can $1/8$ in. from the end of the fuel channel latch assembly slider. The next increment of latch-can travel disengages the locking rod from the latch fingers by $1/8$ in. At this point, the latch-can is contacting the latch assembly slider. The remaining $5/8$ -in. travel moves the latch assembly slider, opening the six locking fingers. Closing of the latch finger is accomplished by reversing these sequences.

The second tool post and latching mechanism concept (Figure 0810-18) consists of a primary ram, a secondary ram, and a separate latch-can assembly. The primary ram is ball-screw driven and capable of extending to approximately 250 in. The secondary ram will be a two-stage telescoping cylinder with a stroke of 90 in. and capable of acting independently of the primary ram. Both rams will be capable of exerting an axial force of 3000 lb.

The fuel channel latches are actuated as the latch-can depresses six spring-loaded push rods. Pins integral with the latches ride in cam grooves in the push rods. The motion of the primary ram against the push rods forces the latches to the unlocked position. As the latch-can is withdrawn, the spring pressure on the push rods moves the latches to the locked position.

The pertinent dimensions and configuration of the fuel assemblies which can be used with these fuel latch concepts are shown on Figure 0810-19. The former latch design requires a fuel assembly having the stepped ends ($3/4$ -in.-long by 0.0203 -in.-deep) providing each fingered link of the latch assembly with approximately $1/8$ -in. nominal lip engagement with the fuel assembly centered in the pressure tube. The amount of lip engagement around the periphery of the fuel assembly will vary by the amount of radial clearance provided between the fuel and ID of the pressure tube. A value for minimum link engagement has not yet been set as a requirement; however, $1/6$ in. is considered as an acceptable value until further design studies are made.

The $3/4$ -in. length of the 0.0203 -in. step provides room for engagement of the latch assembly along with distance for overtravel. An overtravel length is needed to insure disengagement of the fingered links prior to actuation of the latch assembly. The overtravel permits complete freedom of the latch assembly to operate without axial fuel column loads and prevents the fingered links from dragging on the cladding of the fuel assembly during a refueling operation.

The latter fuel latch design can operate with a flat fuel element end plate chamfered at the edge. Therefore, a V-groove is formed between adjacent fuel bundles, which enables the fuel latches to separate the bundles by a wedging action.

3. Main Carriage and Refueling Head Support Assembly

The x-y motion of the refueling head over the reactor face is achieved by the main carriage and the refueling head support assembly. Preliminary studies of the main carriage and refueling head support assembly for both the top and bottom segments of the refueling machine have led to selection of an underhung

arrangement for the main carriage of both segments (Figures 0810-20 and 0810-21). This will allow for removal of the refueling head to the associated maintenance area in a relatively simple manner. The weight of the main carriage is estimated to be 12 tons.

The refueling head support assembly will position the refueling head along a line perpendicular to the direction of motion of the main carriage. For the top segment of the refueling machine, it will be an underhung arrangement with the snout projecting from the bottom of the refueling head. For the bottom segment of the refueling machine, it will be an underhung arrangement with the snout protruding from the top of the refueling head. The weight of the refueling head support assembly is estimated to be 8 tons.

The overall length of the refueling head is 21 ft, with an estimated weight of 5 tons. This produces an overall height for the refueling head, its support assembly, and the main carriage, of 23 ft, and a total weight of 25 tons.

4. Environmental Control System

The purpose of the environmental control system is to control the temperature, pressure, and coolant quality in the refueling head. The operating conditions of the refueling head have been studied to determine the source and magnitude of changes in operating conditions which must be controlled. This study has shown that:

1) During the refueling cycle, the refueling head must be coupled to the fuel channel, the spent fuel discharge tube, and the fresh fuel charging tube. The environments of these components are:

a) Fuel channel - 1st Pass Inlet	- 195 psia	550°F max
1st Pass Outlet	- 165 psia	710°F max
2nd Pass Inlet	- 145 psia	690°F max
2nd Pass Outlet	- 85 psia	780°F max

b) Spent fuel discharge tube - to be determined.

c) Fresh fuel receiving tube - to be determined.

2) During the normal refueling cycle, the magazine would sequentially store up to three fuel assemblies irradiated to 20,000 Mwd/Tonne. During an abnormal condition, the machine will be required to store five fuel assemblies from one fuel channel.

3) The coolant pressure in the machine must be adjusted to suit the internal pressure of the component to which it is to be coupled. This must be done to control the flow of coolant to or from the machine.

4) The coolant temperature must be controlled to prevent distortion or undesirable differential expansion of the machine mechanisms and sealing surfaces, to reduce the thermal shock to fresh fuel assemblies being transferred from one environment to another, and to remove heat from irradiated fuel assemblies during their residence time in the machine.

5) The coolant flow distribution in the refueling head must be controlled to provide uniform temperature control in all areas and to prevent stagnant pockets of coolant from forming.

6) The coolant quality must be controlled to minimize deposits of undesirable materials in the refueling head.

7) Separate environmental control systems are required for each machine.

8) It is desirable to use the reactor coolant as the fluid medium in the refueling head. However, special provision must be made for filling and draining the machine for maintenance.

Two methods of environmental control are being considered. The first system would provide temperature control by supplying organic coolant to the

refueling head from an external source. This provides a source of clean coolant which is necessary for the close tolerance mechanisms of the tool post. Temperature control is obtained by mixing of hot and cold organic coolant. Pressure control is obtained by a pressurizing pump. This system is preferred because the bulk of the refueling machine's environmental control system can be isolated from the refueling head. The major problem area is the method by which the organic is transported to the machine. At least two temperature zones are required in the machine, one in the storage area and one in the tool post actuator area. This in turn requires one feed line to each area and one common return line. These lines would be duplicated for safety, requiring a minimum of six separate lines.

The second system proposed would reduce the number of supply lines by taking the coolant for the refueling machines from a fuel channel. However, this places severe temperature requirements on the machine's mechanisms and requires a cooling system mounted on the refueling head to remove the heat produced in the spent fuel. This method does not provide for the control of the fluid's cleanliness; therefore, the machine is subject to more frequent and longer maintenance shutdowns due to crud buildup and contamination. Temperature control would also be more difficult.

The functions that must be performed by the auxiliary fuel handling equipment were defined, and a flow diagram was prepared. The basic functions for handling of new fuel are: receiving, inspection, storage, environmental change, and loading into the refueling machine. For handling spent fuel, they are: unloading from the refueling machine, cleaning, cooling and storage, inspection, and shipping. A separate process line is being considered for cleaning, cooling,

and shipping damaged fuel. The only maintenance function thus far defined is the periodic replacement of the seal in the refueling machine's coupling assembly.

A search of technical reports and literature is being conducted to find materials that have been successfully used for high temperature applications. In addition, information is being gathered on the materials used in the Canadian NPD, CANDU, and WR-1 projects.

Preliminary test specifications are planned for each of the following: coupling assembly, tool post assembly, magazine assembly, refueling head support assembly, environmental control system, instrumentation and control system, sub-assembly life evaluation, complete prototype machine, AMF adaptor valve, and AMF refueling head. These plans describe the components to be tested, the expected operating conditions, and the information desired.

Arrangements were made to obtain experimental hardware from the AMF refueling machine program for evaluation. Receipt of this equipment is expected in July. Refueling Method No. 7 (Figure 0810-6) was selected as the most desirable based on its favorable refueling time, tool post extension, magazine size, wear, and heat load. The design effort will concentrate on the fuel supports, the tool post, and the coupling assembly, since they appear to be the most critical components.

Method No. 1 (Figure 0810-1) is being studied as an alternate system, since the fuel supports for this system may be simpler.

All tests except the very basic functional checks should be performed using organic coolant at the actual operating temperatures. The lubricity of the organic fluid is expected to be different enough from other test fluids

that the use of a substitute fluid would give invalid results. Only one refueling machine segment will be built and tested, the lower position. Operation of the upper machine segment will be simulated through the use of the test equipment.

III. EVALUATION OF EFFORT TO DATE

The screening studies and the evaluations of the AMF and AECL work on refueling have resulted in identification of the major problem areas listed below:

- 1) Sealing the machine to the fuel channel
- 2) Sealing the machine when it is disconnected from the fuel channel
- 3) Maintaining the seals and sealing surfaces
- 4) Supporting the fuel column in a manner compatible with both on-power refueling and the fuel channel environment
- 5) Developing an actuating mechanism for the tool post assembly
- 6) Supporting the fuel assemblies in the storage magazine
- 7) Controlling the machine environment
- 8) Controlling the machine movements
- 9) Diagnosing machine malfunctions
- 10) Loading the machine with fresh fuel and unloading irradiated fuel
- 11) Maintaining a machine containing irradiated fuel
- 12) Developing an end plate configuration for the fuel assembly compatible with on-power refueling and neutron economy
- 13) Minimizing flow disturbances in the fuel channel during the refueling cycle
- 14) Developing an actuating mechanism for the fuel channel end seal
- 15) Limiting the force transmitted to the fuel channel during coupling operations.

The time and motion studies of the refueling operations indicate that the refueling cycle can be completed in the time allotted.

The selected refueling method requires operation of latching devices installed in the fuel channel extension. These latches support the fuel during normal operation and assist in separating the spent element from the fuel column during refueling operations. Design of this latch depends on the fuel assembly end configuration, the allowable forces which can be expected on the fuel assembly during fuel handling, and the effect of long-term operation in a hot organic environment.

IV. NEXT REPORT PERIOD ACTIVITIES

During the next report period the following work is planned:

- 1) The preliminary design of the tool post for Method No. 7 and coupling assembly will be completed.
- 2) A comparison of several different types of latching mechanisms will be completed, and the most feasible design will be selected.
- 3) The heat load and the fuel channel flange temperatures will be determined, and a design specification for the environmental control system will be prepared.
- 4) The design of the refueling head support assembly and its auxiliary equipment, such as heat exchangers and drives, will be initiated.
- 5) A comparison of methods of transporting the refueling head to the maintenance area and methods of positioning the refueling head at the fuel channels, loading stations, and unloading stations will be started.
- 6) The list of recommended material combinations will be updated as additional information is accumulated. The refueling machine will be

analyzed to define the critical wear areas. The loading conditions in each critical area will be estimated and used in a wear test program to determine suitable materials for each application.

7) The functional requirements of the auxiliary fuel handling equipment will be completed. The critical components will be selected for detailed design effort.

8) The design specifications for the refueling machine will be completed.

9) The evaluation test specification will be prepared.

FIG 0810-2. REFUELING SCHEME NO. 3

STEP 1
 REFUELING MACHINE LOCATES ON FUEL CHANNEL 2 0 MIN
 REFUELING MACHINE COUPLES TO FUEL CHANNEL 2 0 MIN
 LEAKAGE TEST OF COUPLING SEAL PERFORMED 5 0 MIN
 MACHINE CLOSURE PLUG REMOVED AND STORED IN MAGAZINE 2 0 MIN
 FUEL CHANNEL END PLUG REMOVED AND STORED IN MAGAZINE 2 0 MIN

STEP 2
 LOCKING ROD REMOVED AND STORED IN MAGAZINE 3 20 18 30 18
 FUEL CLAMP FUEL CLAMP HOLDS REMAINDER OF COLUMN AS LOCKING RODS ARE STORED 4 0 MIN 3 0 MIN

STEP 3
 FIVE FUEL ASSY REMOVED & STORED IN MAGAZINE 5 0 MIN
 FRESH FUEL ASSY PLACED IN COLUMN 1 0 MIN
 FOUR OF PREVIOUS FUEL ASSY REPLACED IN COLUMN 4 0 MIN

STEP 4
 FUEL CHANNEL END PLUG REPLACED 2 0 MIN
 MACHINE CLOSURE PLUG REPLACED 2 0 MIN
 LEAKAGE TEST OF END PLUG AND CLOSURE PLUG PERFORMED 5 0 MIN
 MACHINES UNCOUPLE 0 5 MIN

FIG 0810-3. REFUELING SCHEME NO. 4

STEP 1
REFUELING MACHINES LOCATED ON FUEL CHANNEL 2.0 MIN
REFUELING MACHINES COUPLE TO FUEL CHANNEL 2.0 MIN
LEAKAGE TEST OF COUPLING PERFORMED 5.0 MIN
MACHINE CLOSURE PLUGS REMOVED AND STORED IN MAGAZINE 2.0 MIN
FUEL CHANNEL END PLUGS REMOVED AND STORED IN MAGAZINE 2.0 MIN

STEP 2
DOWNSTREAM TOOL POST PUSHES COLUMN INTO UPSTREAM MACHINE WHERE LOCKING ROD SEGMENTS ARE STORED. TOOL CLAMP RESTRICTS THE COLUMN AS EACH FUEL ASSEMBLY IS REMOVED AND STORED IN MAGAZINE. FUEL FUEL ASSEMBLY IS LATELY REMOVED AND STORED 5.0 MIN 2.0 MIN 5.0 MIN 0.0 MIN

STEP 3
UPSTREAM MACHINE INSERTS FOUR OF THE ORIGINAL FUEL ASSEMBLIES AND ONE NEW FUEL ASSEMBLY. ONE SPENT FUEL ASSEMBLY IS REMOVED IN THE MAGAZINE 6.0 MIN

STEP 4
UPSTREAM MACHINE REPLACES LOCKING ROD SEGMENTS 10.0 MIN 1.0 MIN
DOWNSTREAM MACHINES REPLACE FUEL CHANNEL END PLUGS 2.0 MIN 2.0 MIN
MACHINE CLOSURE PLUGS REPLACED 2.0 MIN 2.0 MIN
MACHINE TEST OF END PLUGS AND CLOSURE PLUGS PERFORMED 5.0 MIN 3.0 MIN
MACHINES UNCOUPLE 0.5 MIN 10.5 MIN

FIG 0810-4. REFUELING SCHEME NO. 5

FIG 0810-5. REFUELING SCHEME NO. 6

FIG 0810-6. REFUELING SCHEME NO. 7

FIG 0810-7. MODIFIED COMMERCIAL COUPLING ASSEMBLY AND SEAL

SEAL MATERIAL - INCONEL 718 OR INCONEL X

FIG 0810-8. INSTALLATION OF THIN METALLIC SEALS

FIG 0810-9. COUPLING ASSEMBLY FOR THE CANDU REACTOR

FIG 0810-10. FUEL CHANNEL COUPLING DESIGN NO. 1

FIG 0810-11. FUEL CHANNEL COUPLING DESIGN NO. 2

FIG 0810-12. FUEL CHANNEL COUPLING DESIGN NO. 3

FIG 0810-13. FUEL CHANNEL COUPLING DESIGN NO. 4

FIG 0810-14. FUEL CHANNEL COUPLING DESIGN NO. 5

FIG 0810-15. FUEL CHANNEL COUPLING DESIGN NO. 6

⊥ FUEL CHANNEL

⊥ FUELCHANNEL

FIG 0810-16. FUEL CHANNEL COUPLING DESIGN NO. 7

FIG 0810-17. FUEL LATCHING ARRANGEMENT NO. 1

CONFIGURATIONS BASED ON FUEL HANDLING
 REQUIREMENTS OF THE TWO SYSTEMS NOW UNDER
 CONSIDERATION.
 DIRECT COUPLING OF FUEL BUNDLES TO EACH
 OTHER IS NOT REQUIRED

FIG 0810-19. FUEL ASSEMBLY

FIG 0810-20. TOP REFUELING MACHINE 1000 MWe PLANT - DESIGN NO. 2

FIG 0810-21. BOTTOM REFUELING MACHINE 1000 MWe PLANT - DESIGN NO. 1

1000 MWE PLANT ENGINEERING

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HWO CR QUARTERLY REPORT

TITLE: 1000-Mwe Plant Engineering and Development

SUBTASK NO.: 0910

PROJECT ENGINEER: G. A. Anderson

REPORTING PERIOD: Inception
through June 1965

I. PROJECT OBJECTIVES

The objective of this task is to provide a preliminary reference design and cost estimate for a 1000-Mwe power plant, and to develop this reference design for eventual construction and operation of the plant. Also under this task, large plant components and process systems will be developed and tested. The work is being coordinated closely between Atomics International and Combustion Engineering. The design and cost estimate work for the reactor and its components, including the reactivity control system and the on-power refueling system, is to be performed by Combustion Engineering with assistance from Atomics International. The design and cost estimate for the remainder of the plant and its components is to be performed by Atomics International with assistance from Combustion Engineering. The overall plant study is to be directed toward:

- 1) Establishing, in close coordination with the Commission, a uniform set of design bases and economic ground rules to permit evaluation on a common basis.
- 2) Evaluation of alternative: fuels, cladding, fuel channels, fuel geometries, primary loop components, process and control systems, and steam conditions and cycles.
- 3) Identification of the research and development requirements for these plants.

4) Preparation of cost estimates to establish plant economics.

5) Establishment of characteristics, component sizes, and criteria which must be incorporated in the design of the 300-Mwe plant to make certain that it will serve as a demonstration plant as well as a prototype for the 1000-Mwe plant.

6) Establishment of characteristics, component sizes, and criteria that can be extrapolated to an 8300-Mwt desalting plant and assure that the 1000-Mwe plant will serve as a prototype for the 8300-Mwt plant.

Within the overall plant guidelines, the objectives of the work performed under this task are the following:

1) Perform the necessary scoping and parameter studies, optimizations, and evaluations for selecting the conceptual and reference designs. The optimization of the reactor will be factored into the primary system and steam plant optimization study to establish the overall plant.

2) Develop preliminary and conceptual designs of the 1000-Mwe reactor.

3) Provide guidance to the research and development program performed under other tasks, such as fuel channel development, fuel assembly development, reactor control development, and refueling development, by identifying the functional requirements and design limitations of systems and components.

4) Factor the results of the research and development programs into the design optimizations and evaluations performed under this task.

5) Provide guidance in establishing those essential features of this plant which are to be checked out in the 300-Mwe demonstration plant.

- 6) Provide guidance for extrapolations to larger plants such as the 8300-Mwt dual-purpose plant.

II. PROGRESS DURING REPORT PERIOD

The problems associated with the Plant Engineering Task can be divided into the following sub-tasks:

- 1) Reactor Physics
- 2) Engineering Analysis
- 3) Mechanical Design
- 4) Cost Estimating and Analysis
- 5) Safety, Control and Instrumentation.

Since the efforts directed to the latter two items were nominal during this report period, progress in these areas is reported in the first three sub-tasks listed and in other tasks in this progress report. The following sections of this task are separated, where convenient, into the areas of Reactor Physics, Engineering Analysis, and Mechanical Design.

A. REACTOR PHYSICS

The work started during this report period consisted of providing preliminary nuclear data to the Project for guidance in making decisions and making preparations for meeting anticipated future demands. Computational techniques, some of which have been developed under the Reactor Physics Task and the remainder being more or less standard techniques, have been utilized in parameter surveys and scoping-type calculations to obtain information in some of the areas described above.

Effort during this report period has been directed to meet certain imme-

diate needs of the project and to establish the state-of-the-art in certain areas. A parameter survey was initiated to provide data for use in the one-pass/two-pass evaluation: preliminary calculations in the shielding area were carried out to provide data on the neutron exposures and gamma heating rates; a literature survey on the subject of xenon stability was initiated to further define the state-of-the-art in this area; investigations into the magnitude of flux peaking resulting from the discontinuity in fuel at the bundle interfaces have been initiated; and a meeting was held at Oak Ridge National Laboratory between AI, CE, ORNL, and SRL personnel to discuss preliminary results obtained by each participating group on a specific lattice cell calculation. A more detailed description of the results obtained in these areas is presented in subsequent sections.

On the basis of the arguments presented in CEND-175,* it has been assumed for initial phases of the investigation that the bi-directional refueling scheme will be employed in the HWOCR. One phase of the investigations in this area will, in fact, involve a comparison between the bi-directional feed system and one or more of the other promising fuel management schemes. Physics calculations to date are based on a simple model in which fuel of all exposures is distributed uniformly throughout the core. Only after the calculational methods have been established under the Reactor Physics Task will an explicit fuel management calculation be factored into the parameter surveys.

1. Physics Parameter Survey

- a. Summary

Physics calculations for the first phase of the 1000-Mwe parameter survey

*CEND-175, Organic Cooled, Heavy Water Moderated Reactor Power Plant Conceptual Design and Evaluation, Combustion Engineering (March 1963).

have been completed. Equations relating the variation of structural and operational parameters to the feed enrichment, assuming continuous feed and withdrawal of fuel in a homogenized reactor, have been generated for 37- and 19-rod bundle configurations at two organic to UO_2 area ratios, 0.580 and 0.824. All calculations assume a fixed value of the D_2O to UO_2 area ratio, 11.92. Data on local power peaking and parameters relevant to fuel cycle cost evaluation are given.

The feed enrichment equations are intended for use in evaluating the relative change in feed enrichment resulting from given changes in structural or operational parameters. These equations should not be used for determining the magnitude of the feed enrichment because: (1) the fuel management scheme is based on a point reactor with continuous refueling and discharge, and not an explicit bi-directional refueling scheme, and (2) fast fission effects are included in a homogeneous rather than a heterogeneous manner.

b. Fuel Bundle Configurations

Figures 0910-1 and 0910-2 illustrate the geometries employed in the 37- and 19-rod fuel channel configurations. In each case, the larger diameter rods are spaced on a uniform triangular pitch in a hexagonal array with the smaller diameter rods replacing the outermost rods in the otherwise uniform array, the ratio of large to small rod diameters being determined by the requirements on maintaining satisfactory coolant ligament dimensions and preserving a circular outer boundary for the fuel bundle. Table 0910-1 summarizes the nuclear and mechanical parameters employed in this phase of the parameter survey. Each fuel rod is contained within a SAP-clad tube with an "equivalent" cladding thickness, since fins are not represented explicitly in the calculations. The assumed fuel density of 10.0 gm/cm^3 is less than the normal pellet density (10.4 to 10.5 gm/cm^3), in order to reflect either the homogenization of a radial gas gap

TABLE 0910-1

SUMMARY OF NUCLEAR AND MECHANICAL PARAMETERS EMPLOYED IN PARAMETER SURVEY

TOTAL FISSION POWER	3200 Mwt
ACTIVE CORE HEIGHT	17.3 ft
FUEL DENSITY	10.0 gm/cm ³
FUEL RODS/CHANNEL	37, 19
FUEL AREA/CHANNEL	6.0376 in. ²
ORGANIC/UO ₂ AREA RATIO	0.580, 0.824
MODERATOR/FUEL AREA RATIO	11.9232
FUEL CLADDING MATERIAL	SAP-895
CLEARANCE GAP (BUNDLE TO PRESSURE TUBE)	0.062 in.
INSULATION GAS GAP	0.125 in.
PRESSURE TUBE MATERIAL	SAP-895
CALANDRIA TUBE MATERIAL	ZIRCALOY-4
CALANDRIA TUBE THICKNESS	0.040 in.
COOLANT MATERIAL	SANTOWAX-OMP (10% HB)
MODERATOR PURITY D/(D+H)	0.9975
MODERATOR TEMPERATURE	190°F
AVERAGE DISCHARGE EXPOSURE	20,000 MWD/MTU
37-ROD BUNDLE PARAMETERS	
CLADDING (LARGE ROD)	0.015, 0.020, 0.025 in.
CLADDING (SMALL ROD)	0.010, 0.015, 0.019 in.
PRESSURE TUBE THICKNESS	0.040, 0.110, 0.180 in.
FUEL DIAMETER (LARGE ROD)	0.481 in.
FUEL DIAMETER (SMALL ROD)	0.293 in.
AVERAGE FUEL TEMPERATURE	1000 - 1400°F
AVERAGE COOLANT TEMPERATURE	600 - 700°F
NUMBER PRESSURE TUBES	800 - 2500

and axial dishing of pellets or the use of vibratory-compacted UO_2 . The number of fuel channels in the reactor, and hence the geometric buckling, is determined by the power per channel which is related to specific power density.

Table 0910-1 contains dimensional data only for the 37-rod assemblies. Data for the 19-rod assemblies, i.e., fuel column OD and cladding thicknesses, were determined in such a manner as to define 19-rod assemblies which are "equivalent" to the 37-rod assemblies. To define an equivalent 19-rod assembly, the following quantities were held constant: (1) the cladding thickness to fuel rod diameter ratio, (2) the total fuel area, (3) the organic coolant area, (4) the pressure and calandria tube thicknesses, and (5) the ratio of D_2O to UO_2 fuel areas. The ratio of small to large fuel rod diameters was determined by reducing the coolant ligaments to zero and requiring that the outer boundary of the clad fuel rod bundle be circular. The geometric relationships are expressed by the following equation:

$$1.5 (D_L + 2t_L) + (D_S + 2t_S) = (\sqrt{3} + .5) (D_L + 2t_L)$$

where:

$$\left(\frac{t_L}{D_L} \right)_{19} = \left(\frac{t_L}{D_L} \right)_{37} = M_L$$

$$\left(\frac{t_S}{D_S} \right)_{19} = \left(\frac{t_S}{D_S} \right)_{37} = M_S$$

D_L, D_S - represent the diameters of large and small fuel columns, respectively

t_L, t_S - represent the clad thickness for large and small fuel columns, respectively

hence:

$$\frac{D_S}{D_L} = (\sqrt{3} - 1) \frac{1 + 2M_L}{1 + 2M_S}$$

Tables 0910-2A and 0910-2B summarize the fuel channel geometries and average reactor conditions employed for both the 37- and 19-rod calculations.

TABLE 0910-2A

PARAMETERS EMPLOYED IN CALCULATIONS FOR 37-ROD ASSEMBLIES^{*†}
 HAVING FUEL ROD DIAMETERS OF 0.481 AND 0.293 FOR THE LARGE
 AND SMALL RODS, RESPECTIVELY

CALC NO.	A_{COOLANT} A_{UO_2}	CLAD THICKNESS (IN.)		PRESSURE TUBE THICK- NESS (IN.)	$\bar{T}_{\text{FUEL}}$ (°F)	$\bar{T}_{\text{COOLANT}}$ (°F)	NO. OF FUEL CHANNELS
		LARGE	SMALL				
A-1	0.824	0.015	0.010	0.040	1200	650	1200
A-2	0.824	0.025	0.019	0.180	1200	650	1200
A-3	0.824	0.020	0.015	0.110	1200	650	1200
A-4	0.824	0.020	0.015	0.110	1400	650	1200
A-5	0.824	0.020	0.015	0.110	1200	700	1200
A-6	0.824	0.020	0.015	0.110	1200	650	2500
A-7	0.824	0.020	0.015	0.040	1200	650	1200
A-8	0.824	0.020	0.015	0.180	1200	650	1200
B-1	0.580	0.015	0.010	0.040	1200	650	1200
B-2	0.580	0.025	0.019	0.180	1200	650	1200
B-3	0.580	0.020	0.015	0.110	1200	650	1200
B-4	0.580	0.020	0.015	0.110	1400	650	1200
B-5	0.580	0.020	0.015	0.110	1200	700	1200
B-6	0.580	0.020	0.015	0.110	1200	650	2500

*SEE FIG 0910-1

†NOMINAL FUEL AREA 6 IN.², CALANDRIA THICKNESS - .040 IN.

TABLE 0910-2B

PARAMETERS EMPLOYED IN CALCULATIONS FOR 19-ROD ASSEMBLIES * †
 HAVING FUEL ROD DIAMETERS OF 0.691 AND 0.497 FOR
 THE LARGE AND SMALL RODS, RESPECTIVELY

CALC NO.	$\frac{A_{\text{COOLANT}}}{A_{\text{UO}_2}}$	CLAD THICKNESS (in.)		PRESSURE TUBE THICKNESS (in.)	$\bar{T}_{\text{FUEL}}$ (°F)	$\bar{T}_{\text{COOLANT}}$ (°F)	NO. OF FUEL CHANNELS
		LARGE	SMALL				
C-1	0.824	0.0287	0.0254	0.110	1200	650	1200
C-2	0.824	0.0287	0.0254	0.180	1200	650	1200
C-3	0.824	0.0359	0.0321	0.110	1200	650	1200
C-4	0.824	0.0287	0.0254	0.110	1400	650	1200
C-5	0.824	0.0287	0.0254	0.110	1200	700	1200
C-6	0.824	0.0287	0.0254	0.110	1200	650	2500
C-7	0.824	0.0287	0.0254	0.110	1200	650	800
D-1	0.580	0.0287	0.0254	0.110	1200	650	1200
D-2	0.580	0.0287	0.0254	0.180	1200	650	1200
D-3	0.580	0.0359	0.0321	0.110	1200	650	1200

* SEE FIG 0910-2

† NOMINAL FUEL AREA 6 in.², CALANDRIA TUBE THICKNESS - .040 in.

c. Physics Computational Model

The physics calculational model consists of a homogenized cell calculation, with thermal group and resonance self-shielding factor (L-factor), and the material and region-wise volume fractions for use in the zero-dimensional fast spectrum calculation, FORM,* and determines the "equivalent" annular geometry for the one-dimensional thermal group calculation (TONG-THERMOS).* Non-unity resonance self-shielding factors are presently employed only for U^{238} in the FORM program. The calculational procedure used in ALFAC* consists of the "rubber-band" model with equivalent slab leakage probabilities to determine the effective surface to mass ratio of the fuel bundle, an analytical expression relating the resonance integral for an isolated rod to temperature and effective cord length of the rod bundle, and the Bell scheme to account for lattice interaction effects.

The nuclear code FORM is employed to compute the neutron spectrum from 10 Mev to the thermal cutoff of 0.625 ev. Spectrally averaged constants are evaluated for three fast groups having breakpoints at 0.821 Mev, 5.53 Kev, and 0.625 ev. Input data to this code consist of homogenized smeared number densities, L-factors for elements having resonances, and the geometric buckling, B_g^2 . The options employed in FORM calculations are the P_1 slowing-down approximation, the self-consistent age approximation, and an exact treatment of hydrogen slowing down (no Selengut-Goertzel approximation). The FORM code output includes fast microscopic cross-section cards for direct input to the point depletion calculation, INGEN-2 (CRUMB).* In the calculations reported herein, the homogeneous, rather than the heterogeneous, fast fission effect has

*For code reference, see Section 0610, Reactor Physics.

been developed. The calculational procedure for incorporating a heterogeneous fast fission effect has been developed, but has not as yet been incorporated into the mechanics of the survey calculational package.

The thermal group calculations employ a modified TONG-THERMOS program. The TONG-THERMOS program solves the space-energy problem by obtaining an integral solution to the Boltzmann equation. This program differs from the BNL version of THERMOS in several ways: (1) dimension statements TONG-THERMOS allows 50 isotopes, 7 mixtures, and 25 mesh intervals, whereas the BNL version allows 10 isotopes, 5 mixtures, and 20 mesh intervals; (2) the output formats TONG-THERMOS computes cell-averaged flux and volume-weighted microscopic cross sections. The program was modified to include a more generalized Radkowsky correlation to apply to deuterium as well as hydrogen, and to provide thermal microscopic cross-section cards for direct input to the point depletion calculation. Input data for TONG-THERMOS consists of region-wise composition data and physical temperatures, and spatial source distribution. The scattering kernels employed were of the Brown-St. John type. Four types of regions or mixtures are employed in the standard TONG-THERMOS calculation: (1) fuel, (2) homogenized cladding and coolant, (3) homogenized pressure tube, insulation gas gap, and calandria tube, and (4) moderator. Figure 0910-3 illustrates the one-dimensional representation of the lattice cell containing a 37-rod bundle. Conversion of the fuel rod bundle to annular fuel rings is accomplished by preserving the fuel area and positioning the center of the fuel annulus on the mean center of the fuel rods in the corresponding ring.

Point depletion calculations were carried out by use of the INGEN-2 (CRUMB) program. This program requires as input four-group thermal and fast

microscopic cross sections, thermal self-shielding factors, specific power density, homogenized cell number densities, and four-group bucklings; a criticality search on buckling was employed for the calculations reported herein. Output edits include such data as material inventories and eigenvalues for each time step. In addition, a time-averaged eigenvalue, k_{eff}^* , is evaluated; this eigenvalue corresponds to a continuous feed (and discharge) homogenized system and is evaluated by the expression:

$$k_{eff}^* = \frac{1}{T(\text{Mwd/MTU})} \int_{T_1}^T k_{eff} dt$$

where: T_1 is usually the equivalent of ~ 50 hr

T is the discharge exposure (Mwd/MTU).

In general, the point depletion calculations were carried out for several enrichments, so as to permit a determination of the feed enrichment necessary to make k_{eff}^* equal to unity for various discharge exposures.

d. Results

The ALFAC, FORM, TONG-THERMOS, and INGEN-CRUMB series of calculations were carried out for the cases listed in Tables 0910-2A and 0910-2B; composition areas for each case are listed in Tables 0910-3A and 0910-3B. The data resulting from these calculations which are of interest in the fuel cycle costing and thermal-hydraulic analyses are: (1) the fissile content of the discharged fuel (tentatively set at 20,000 Mwd/MTU) shown in Tables 0910-4A and 0910-4B; (2) the feed enrichment required to achieve the average discharge exposure (subject to the limitations of the calculational model) shown in Tables 0910-4A and 0910-4B; and (3) the relative thermal power distribution within the fuel bundle. The following tabulation gives representative values of the local-to-average specific power for the 37-

TABLE 0910-3A

COMPOSITION AREAS (CM²) FOR 37-ROD ASSEMBLIES HAVING A TOTAL
FUEL AREA OF 38.952 CM² AND A D₂O AREA OF 464.4316 CM²

CALC NO.	ORGANIC	CLADDING & WIRE	PRESS, TUBES	VOID	TOTAL
A-1	32.0933	5.0432	6.6157	10.3370	557.4728
A-2	32.0933	8.6691	18.8697	11.2776	574.2933
A-3	32.0933	6.8576	12.5796	10.8081	565.7221
A-4	32.0933	6.8576	12.5796	10.8081	565.7221
A-5	32.0933	6.8576	12.5796	10.8081	565.7221
A-6	32.0933	6.8576	12.5796	10.8081	565.7221
A-7	32.0933	6.8576	6.6902	10.4534	559.4780
A-8	32.0933	6.8576	18.6676	11.1628	572.1648
B-1	22.5900	5.0432	6.2102	9.7035	546.9305
B-2	22.5900	8.6691	17.7820	10.6596	563.0844
B-3	22.5900	6.8576	11.8288	10.1825	554.8425
B-4	22.5900	6.8576	11.8288	10.1825	554.8425
B-5	22.5900	6.8576	11.8288	10.1825	554.8425
B-6	22.5900	6.8576	11.8288	10.1825	554.8425

TABLE 0910-3B

COMPOSITION AREAS (CM²) FOR 19-ROD ASSEMBLIES HAVING A TOTAL
FUEL AREA OF 38.952 CM² AND A D₂O AREA OF 464.4316 CM²

CALC NO.	ORGANIC	CLADDING & WIRE	PRESS. TUBES	VOID	TOTAL
C-1	32.0933	7.0634	12.5953	10.8212	565.9568
C-2	32.0933	7.0634	18.6907	11.1759	572.4069
C-3	32.0933	8.9470	12.7384	10.9404	568.1027
C-4	32.0933	7.0634	12.5953	10.8212	565.9568
C-5	32.0933	7.0634	12.5953	10.8212	565.9568
C-6	32.0933	7.0634	12.5953	10.8212	565.9568
C-7	32.0933	7.0634	12.5953	10.8212	565.9568
D-1	22.5900	7.0634	11.8456	10.1965	555.0791
D-2	22.5900	7.0634	17.5911	10.5512	561.1793
D-3	22.5900	8.9470	11.9981	10.3236	557.2423

TABLE 0910-4A

RESULTS OF DEPLETION CALCULATIONS AT 20,000 MWD/MTU FOR 37-ROD ASSEMBLIES

CALC. NO.	DISCHARGE (4)				INTERPOLATED (3)		
	U 235 (1) (Wt %)	U 235 (Wt %)	TOTAL Pu g/KgU (2)	FISSILE Pu g/KgU (2)	U 235 (1) (Wt %)	U 235 (4) (Wt %)	FISSILE Pu g/KgU (2)(4)
A-1	0.72	0.0237	4.2394	2.8768	1.2540	0.1189	2.9516
	1.10	0.0758	4.2253	2.9117			
	1.50	0.1877	4.2492	3.0153			
A-2	0.72	0.0242	4.2464	2.8909	1.4071	0.1634	3.0085
	1.10	0.0769	4.2353	2.9280			
	1.50	0.1895	4.2624	3.0329			
A-3	0.72	0.0240	4.2428	2.8837	1.3306	0.1410	2.9796
	1.10	0.0763	4.2301	2.9196			
	1.50	0.1886	4.2554	3.0236			
A-4	0.72	0.0244	4.2463	2.8820	1.3329	0.1432	2.9839
	1.10	0.0773	4.2368	2.9213			
	1.50	0.1904	4.2261	3.0289			
A-5	0.72	0.0243	4.1884	2.8289	1.3272	0.1415	2.9296
	1.10	0.0773	4.1792	2.8687			
	1.50	0.1903	4.2082	2.9759			
A-6	0.72	0.0236	4.2253	2.8632	1.2999	0.1314	2.9470
	1.10	0.0756	4.2102	2.8967			
	1.50	0.1874	4.2320	2.9973			
A-7	0.72	0.0238	4.2437	2.8815	1.2660	0.1226	2.9647
	1.10	0.0762	4.2363	2.9254			
	1.50	0.1879	4.2539	3.0200			
A-8	0.72	0.0241	4.2471	2.8907	1.3861	0.1570	3.0025
	1.10	0.0766	4.2354	2.9275			
	1.50	0.1890	4.2620	3.0324			
B-1	0.72	0.0236	4.2217	2.9074	1.2169	0.1080	2.9739
	1.10	0.0754	4.2101	2.9429			
	1.50	0.1870	4.2393	3.0490			
B-2	0.72	0.0241	4.2329	2.9252	1.3676	0.1517	3.0346
	1.10	0.0766	4.2243	2.9627			
	1.50	0.1888	4.2566	3.0702			
B-3	0.72	0.0238	4.2272	2.9162	1.2921	0.1297	3.0037
	1.10	0.0760	4.2169	2.9525			
	1.50	0.1879	4.2475	3.0591			
B-4	0.72	0.0242	4.2269	2.9108	1.2945	0.1318	3.0047
	1.10	0.0770	4.2202	2.9509			
	1.50	0.1897	4.2551	3.0615			
B-5	0.72	0.0241	4.1797	2.8681	1.2896	0.1301	2.9599
	1.10	0.0768	4.1726	2.9079			
	1.50	0.1893	4.2063	3.0175			
B-6	0.72	0.0235	4.2093	2.8953	1.2605	0.1199	2.9705
	1.10	0.0752	4.1966	2.9291			
	1.50	0.1867	4.2236	3.0323			

(1) 0 MWD/MTU

(2) g/KgU CHARGED

(3) INTERPOLATED FOR $k_{eff} = 1$

(4) AT 20,000 MWD/MTU

TABLE 0910-4B

RESULTS OF DEPLETION CALCULATIONS AT 20,000 MWD/MTU FOR 19-ROD ASSEMBLIES

CALC. NO.	U 235 (1) (Wt %)	DISCHARGE (4)			INTERPOLATED (3)		
		U 235 (Wt %)	TOTAL Pu g/KgU (2)	FISSILE Pu g/KgU (2)	U 235 (1) (Wt %)	U 235 (4) (Wt %)	FISSILE Pu g/KgU (2)(4)
C-1	0.72	0.0233	4.1515	2.8024	1.3573	0.1463	2.8976
	1.10	0.0747	4.1333	2.8337			
	1.50	0.1860	4.1528	2.9331			
C-2	0.72	0.0234	4.1503	2.8044	1.4253	0.1658	2.9180
	1.10	0.0751	4.1330	2.8364			
	1.50	0.1867	4.1541	2.9368			
C-3	0.72	0.0233	4.1552	2.8067	1.3701	0.1500	2.9050
	1.10	0.0748	4.1371	2.8379			
	1.50	0.1862	4.1569	2.9373			
C-4	1.10	0.0757	4.1367	2.8322	---	---	---
C-5	1.10	0.0757	4.0830	2.7833	---	---	---
C-6	1.10	0.0747	4.1334	2.8340	---	---	---
C-7	1.10	0.0748	4.1331	2.8339	---	---	---
D-1	1.10	0.0746	4.1351	2.8798	1.312	---	---
D-2	1.10	0.0750	4.1346	2.8821	1.380	---	---
D-3	1.10	0.0747	4.1411	2.8861	1.327	---	---

(1) 0 MWD/MTU

(2) g/KgU CHARGED

(3) INTERPOLATED FOR $k_{eff} = 1$

(4) AT 20,000 MWD/MTU

and 19-rod assemblies as a function of enrichment.

Calc. No.	Enrichment	Rod Location			
		Center	Ring 1	Ring 2	Outer Ring
A-3 (37-rod clusters)	0.72	0.7298	0.7745	0.9031	1.1958
	1.10	0.6780	0.7319	0.8821	1.2351
	1.50	0.6304	0.6922	0.8613	1.2727
C-3 (19-rod clusters)	0.72	0.7389	0.8355		1.1375
	1.10	0.6864	0.8029		1.1649
	1.50	0.6384	0.7722		1.1905

It should be noted that the outer ring for both type clusters contains rods of two diameters; hence, to obtain the rod-by-rod power, the cross sectional areas should be taken into account.

The data of Tables 0910-4A and 0910-4B can be used to define an analytic expression relating the feed enrichment to the five dependent variables: pressure tube thickness, cladding tube thickness, fuel temperature, coolant temperature, and the number of pressure tubes. The following approach was taken. For the 37-rod bundle, data were obtained on the variation of enrichment with both pressure tube thickness and cladding tube thickness varying simultaneously, and the variation of enrichment with pressure tube thickness for a constant cladding thickness of 0.020 in. Assuming that the enrichment for these cases is separable into a cladding component and a component associated with the remaining variables, the incremental change in enrichment for variations in the cladding thickness about 0.020 in. was evaluated. Figure 0910-4 shows the dependence

of feed enrichment on: (1) the simultaneous variation of both cladding and pressure tube thickness (solid line); and (2) the variation of pressure tube thickness (dashed line). The data points for the former curve fall in a straight line, whereas the points for the latter effect deviate slightly from a straight line. The data points about the broken curve represent the incremental change in enrichment with variations in cladding tube thickness. Since only a limited range of cladding tube thickness was examined, it was assumed that the functional dependence of enrichment on cladding tube thickness was linear and a least squares fit was made for the cladding tube thickness as well as the pressure tube thickness. Similarly, the dependence of feed enrichment on the remaining independent variables is assumed to be linear. An analytic expression for the enrichment as a function of the independent variables was approximated by the following expression:

$$e = (m_{pt} \bar{T}_{pt} + m_t t + b) \left[1 + \frac{\Delta e}{e_m \Delta T_f} (\bar{T}_f - 1200^\circ\text{F}) \right] \left[1 + \frac{\Delta e}{e_m \Delta T_c} (T_c - 650^\circ\text{F}) \right] \left[1 + \frac{\Delta e}{e_m \Delta N_{pt}} (N_{pt} - 1200) \right]$$

where: e is the enrichment

m_{pt} } are the slopes of the straight line relating enrichment to pressure tube thickness, T_{pt} , and to cladding tube thickness for the large fuel rod, t , respectively,
 m_t }

b is a constant,

$\Delta e/e_m, \Delta T_f, \Delta T_c, \Delta N_{pt} = \text{constants}$

$\bar{T}_c$ is the average coolant temperature ($^\circ\text{F}$),

$\bar{T}_f$ is the average fuel temperature ($^\circ\text{F}$),

N_{pt} is the number of pressure tubes in the reactor.

Table 0910-5 summarizes the enrichment coefficients for both the 37- and 19-rod fuel bundles. It should be noted that the enrichment equation is applicable over a limited range of the independent variables and should only be used to evaluate relative changes in enrichments. The latter comment is important, since the enrichment determinations are based on the "homogenized pot model" described earlier and do not contain a bi-directional fuel management scheme. In addition, the heterogeneous fast fission effect is not included in the calculations. It is estimated that including this effect would increase the initial (beginning-of-life) reactivity by approximately 1.5%. The values of the enrichment given herein could be used to estimate the absolute magnitude of the feed enrichment if an uncertainty of ~ 0.15 wt% were used.

It should also be noted that the enrichments are based on zero excess reactivity. For the case of 20,000 Mwd/MTU discharge, approximately 1% in $\Delta k/k$ corresponds to 0.045 wt% in enrichment.

Table 0910-6 summarizes reactivity coefficient data evaluated for the supercritical fresh fuel or beginning-of-life conditions. The coefficients for variations in pressure and cladding tube thickness are reported in units of reciprocal inches.

e. Reactor Shielding

Information on the subject of gamma heating, neutron dose rate, and cobalt activation in a nonstructural shield tank wall adjacent to the end wall of the calandria vessel was investigated for the axial thermal shields. The purpose of this investigation was to assess the need for a thin nonstructural wall followed by a light water region to be interposed between the calandria wall and the main structural wall for purposes of shielding and cooling of the more massive struc-

TABLE 0910-5
COEFFICIENTS FOR ENRICHMENT EQUATION

NO. OF RODS	37		19	
ORG/VO ₂	0.824	0.580	0.824	0.580
M _{pt}	0.85826	0.85826	0.97143	0.97130
M _f	3.0520	3.2902	3.0000	2.5620
b	1.137	1.170	1.145	1.199
$\frac{\Delta e}{e_m \Delta I_f}$	0.9241×10^{-5}	0.8658×10^{-5}	1.0×10^{-5}	1.0×10^{-5}
$\frac{\Delta e}{e_m \Delta \Gamma_c}$	-0.3918×10^{-4}	-0.5103×10^{-4}	-0.30×10^{-5}	-0.40×10^{-5}
$\frac{\Delta e}{e_m \Delta N_{pt}}$	-0.18851×10^{-4}	-0.1774×10^{-4}	-0.13×10^{-4}	-0.12×10^{-4}

TABLE 0910-6

REACTIVITY COEFFICIENTS FOR FRESH FUEL

	REFERENCE CONDITION	ORG/UO ₂					
		37-ROD ASSEMBLY			19-ROD ASSEMBLY		
		0.824			0.580		
		0.72 1.1 1.5			.72 1.1 1.5		
$\epsilon(w/o)$		0.72	1.1	1.5	.72	1.1	1.5
k_{eff}		0.994	1.165	1.275	1.011	1.179	1.287
$\frac{10^6 \Delta k}{k \Delta T_f (^\circ\text{F})}$	$T_{\text{fuel}} = 1200^\circ\text{F}$	-9.5	-9.1	-8.7	-9.5	-9.0	-8.7
$\frac{10^6 \Delta k}{k \Delta T_c (^\circ\text{F})}$	$T_{\text{coolant}} = 650^\circ\text{F}$	8.0	6.6	5.8	4.9	5.3	5.0
$\frac{10^6 \Delta k}{k \Delta B^2}$	$B^2 = 0.00062$	3.2	2.9	2.7	3.3	3.0	2.8
$\frac{\Delta k}{k \Delta t_p (\text{in.})}$ (-0.070 in.) (+0.070 in.)	$t_{\text{pres. tube}} = 0.110 \text{ in.}$	0.28 -0.24	0.23 -0.20	0.21 -0.18	-	-	-
$\frac{\Delta k}{k \Delta t_c (\text{in.})}$ (-0.005 in.) (+0.005 in.)	$t_{\text{clad}} = 0.020 \text{ in.}$	0.83 -1.25	0.75 -1.08	0.55 -0.97	-	-	-

0910-21

tural wall. Neutron and gamma-ray source terms were evaluated for a core typical of a 1000-Mwe pressure tube reactor with a 1-ft-thick axial reflector. The maximum heat rate was found to be 0.19 watt/cm^3 at the edge of the nonstructural wall adjacent to the reactor, thus indicating that water shielding of the main structural wall is unnecessary.

The choice of material to be used in the shield tank is dependent upon the total fast neutron flux incident on the tank wall over the plant life. The calculated integrated fast neutron flux above 1 Mev is presently estimated to be about $3 \times 10^{18} \text{ nvt}$ for a 30-yr period. This value should not place any severe limits on the choice of shield tank material.

The thermal neutron flux incident on the shield tank wall is estimated at less than $10^{12} \text{ neutrons/cm}^2/\text{sec}$, which should not present any serious cobalt activation problem.

f. Dynamic Studies - Xenon Spatial Flux Oscillations

A characteristic intrinsic to large, high flux, thermal reactors is the need to control spatial oscillations of the neutron flux and of the Xe^{135} concentration. In this sense, a core is considered large if it has two or more regions loosely coupled by diffusion. Such a condition may be satisfied when core dimensions are large compared to the neutron migration length or when the flux distribution is relatively flat between regions. The term high flux is defined as a flux level that gives rise to a Xe^{135} burnout term, in the xenon balance equation, which is large compared to the natural radioactive decay term for Xe^{135} .

In optimizing core size and power densities for high performance systems, there appears, within the literature examined to date, to be no incentive to

impose restrictions on the design in order to preclude these oscillations. Rather, the incentive is to design detection and control systems flexible enough to limit amplitudes of the oscillations so that heat transfer limits on the fuel elements are not exceeded.

To place in perspective spatial flux oscillations due to xenon changes, it is important to recognize that the long duration of the transients removes them from the category of nuclear safety problems (I^{135} and Xe^{135} have 6.7 and 9.2-hr half-lives, respectively). Spatial power oscillations are considered to be operational and design considerations primarily.

HWOGR cores, from the 300-Mwe size and upward, will be large and will operate in the high flux region, thus fulfilling the conditions for spatial oscillations. The problem, then is to choose an analytical model that will yield results applicable to selecting a detector spacing within the core (thermal neutron flux detectors and/or thermocouples). There are operational and economic incentives to minimize the number of such detectors and, consequently, these incentives should determine the degree of sophistication of the analytical models. Since economic optimization of spatial power distributions is a primary goal, instrumentation of the cores for control of spatial power oscillations is a design area in which a certain degree of conservatism may be justified in the initial systems. With the selected analytical model, conservative assumptions should be employed when selecting values for parameters that determine the dimensions of a control region within the core.

The preceding approach to the problem was formulated after reviewing several dozen articles and reports on the subject of spatial oscillations. With the multitude of analytical models in the literature on the subject of spatial

oscillations, documentation of the interpretation of their results for instrumenting a core is most conspicuous in its absence. The majority of the published work has been addressed to the problem of determining what limits to place on flux level, flux shape, core size, and neutron migration length in order to preclude spatial oscillations. Further search for information is underway on how results of one-and-two-dimensional xenon oscillation studies have been factored into criteria used in instrumenting reactor designs.

2. HWO CR Physics Test Problem

a. Statement of the Problem

The basic goal of the comparison study has been viewed not as a core design, but rather as a comparison of physics calculational methods and cross-section libraries in use at present for parameter surveys and evaluations. Additionally, as such, the results do not necessarily represent the characteristics of a reactor design. Therefore, test case questions requiring thermodynamic analysis have been eliminated, e.g., determination of a power coefficient. Nuclear physics analysis has been limited to fuel cell spectral calculations and single space point depletion calculations for an infinite array of fuel cells.

Additional details of the test problem are given in Tables 0910-7, 0910-8, and 0910-9. Figure 0910-5 illustrates the 37-rod fuel element configuration employed in the calculation. The specific test problem results to be calculated are listed below. Items 2-5 are to be evaluated in the supercritical, beginning-of-life condition. In the depletion calculation, the power density is to be 16.0 watt/cm³, and a just critical condition is to be maintained by using a poison (1/v) in the neutron energy range of 0.0625 ev. A square lattice pitch of 9.32 in. is to be used.

TABLE 0910-7

LATTICE CELL DESCRIPTION FOR PHYSICS TEST CARE

MATERIAL		CELL VOLUME (cm ³ /cm)	T̄ (°F)	REMARKS
FUEL	UO ₂	38.952048	1400	(1) $\rho_{\text{eff}} = 10.0 \text{ g/cm}^3$ U ²³⁵ /U = 1.5 wt%
ORGANIC	OMP, 10% HB	32.093304	650	C ₁₈ H ₁₄ $\rho_{\text{eff}} = 0.862 \text{ g/cm}^3$
CLADDING & SPACER WIRES	SAP 895	5.630131	650 ⁽²⁾	
PROCESS TUBE	SAP 895	5.291972	650	
INSULATION GAS GAP	VOID	10.506579	-	
CALANDRIA TUBE	Zr-4	3.495876	190	
MODERATOR	D ₂ O	464.431550	190	D/D+H = .9975 $\rho_{\text{eff}} =$ 1.0724 g/cm ³
		560.401460		

(1) UO₂ SMEARED OVER ID OF CLAD TUBE

(2) FOR PURPOSES OF THIS CALCULATION THE FILM TEMPERATURE DROP IS ASSUMED TO BE ZERO

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TABLE 0910-8

ATOMIC NUMBER DENSITIES FOR PHYSICS TEST CASE*

UO ₂	N^{235}	=	0.0003388
	N^{238}	=	0.0219679
	N^o	=	0.0446135
ORGANIC	N^H	=	0.0315590
	N^C	=	0.0405759
SAP 895	N^{Al}	=	0.0575989
	N^O	=	0.0047985
Zr-4	N^{Zr}	=	0.0424621
	N^{Fe}	=	0.0001483
	N^{Sn}	=	0.0004986
	N^{Cr}	=	0.0000682
	N^{Ni}	=	0.0000100
MODERATOR	N^D	=	0.06436978
	N^H	=	0.00016132
	N^o	=	0.03226555

*Atoms/Barn cm

TABLE 0910-9

MODERATOR AND COOLANT DENSITY VARIATION WITH
TEMPERATURE FOR PHYSICS TEST CASE

COOLANT DENSITY:

$$\rho_{\text{ORGANIC}} \text{ (g/cm}^3\text{)} = 1.159 - 4.575 \times 10^{-4} \times T \text{ (}^\circ\text{F)}$$

MODERATOR DENSITY:

$\rho_{\text{D}_2\text{O}}$ (g/cm ³)	TEMPERATURE (°F)
1.0999	104
1.0906	140
1.0782	176
1.0630	212
1.0459	248

1) Reactivity - A plot of infinite multiplication constant versus fuel exposure for a range of 0 to 30,000 Mwd/MTU.

2) Thermal flux distribution - Thermal flux spatial distribution in a plane transverse to the axis of the fuel bundle

3) Fast effect - No epsilon, but two terms, (a) fission neutrons produced, and (b) the number of neutrons slowing down from above 0.82 Mev per total fission neutrons produced.

4) Reactivity coefficients -

a) Doppler coefficient defined as one over effective resonance integral times the delta resonance integral per delta temperature for a fuel temperature increase of 65°F above ambient.

b) Coolant density coefficient for a coolant temperature increase of 65°F above ambient value.

c) Coolant void coefficient for total voiding of coolant channel.

d) Moderator temperature coefficient for a moderator temperature increase of 95°F above ambient value.

b. HOWCR Physics Test Problem Results

The HOWCR physics test problem was analyzed under the ground rules described above. The calculational method employed was basically a FORM-THERMOS-INGEN (CRUMB) sequence of calculations with a P-3 spatial flux determination used to define heterogeneous self-shielding factors in these FORM multigroups above 0.82 Mev. The results of the calculation are presented below.

(1) Reactivity

A display of the infinite multiplication factor versus exposure over the range 0 to 30,000 Mwd/MTU is presented in Figure 0910-6. It should be noted that

a poison ($\sigma_a(E) \propto 1/v$, thermal group only) was employed in the point depletion calculations. Flux disadvantage factors in both the thermal and fast fission energy regions were taken to be time-independent and were based on a beginning-of-life evaluation.

(2) Thermal Flux Distribution

Figure 0910-7 presents a plot of the thermal distribution across the Seitz-Wigner lattice cell for both the flooded and voided coolant channel conditions. The local to average flux distribution is evaluated for the fresh fuel or beginning-of-life condition.

(3) Resonance Absorption

The resonance absorption above 0.625 ev per total fission produced for a beginning-of-life fuel element is tabulated below for both the flooded and voided coolant channel conditions.

Fast Fission Calc	<u>Flooded Coolant Channel</u>		<u>Voided Coolant Channel</u>	
	Heterog	Homog	Heterog	Homog
(1-p)	0.2065	0.1826	0.2244	0.1967

(4) Fast Fission Effect

The fast fission effect in U^{238} is defined in terms of two quantities:

(a) fission neutrons produced by fissions above 0.82 Mev per total fission neutrons produced, and (b) number of neutrons slowing down from above 0.82 Mev per total fission neutrons produced. The above quantities are summarized below.

Fast Fission Calc	<u>Flooded Coolant Channel</u>		<u>Voided Coolant Channel</u>	
	Heterog	Homog	Heterog	Homog
$\frac{\sum_{E \geq 0.82 \text{ Mev}} \phi \nu \Sigma_f}{10 \text{ Mev}}$	0.04006	0.01137	0.04275	0.01133
$\frac{\sum_{E > 0} \phi \nu \Sigma_f}{10 \text{ Mev}}$				

Fast Fission Calc	<u>Flooded Coolant Channel</u>		<u>Voided Coolant Channel</u>	
	Heterog	Homog	Heterog	Homog
$\frac{\sum_{E \geq 0.82 \text{ Mev}}^{10 \text{ Mev}} (\phi \Sigma_f)}{E > 0}$	0.5416	0.5612	0.5166	0.5375

(5) Reactivity Coefficients*

Reactivity coefficients evaluated for the supercritical beginning-of-life condition (including heterogeneous fast effects) are as follows:

- a) Doppler $-8.95 \times 10^{-6} \Delta k/k-^{\circ}\text{F}$
- b) Coolant Void $+4.21\% \Delta k/k-100\% \text{ voids}$
- c) Coolant Temps $+9.90 \times 10^{-6} \Delta k/k-^{\circ}\text{F}$
- d) Moderator Temp $-2.48 \times 10^{-5} \Delta k/k-^{\circ}\text{F}$

c. Comparison of Test Problem Results Obtained by Atomics International and Combustion Engineering

Table 0910-10 presents a preliminary comparison of test problem results obtained by Atomics International and Combustion Engineering.

*Based on the following eigenvalues:

<u>Condition</u>	<u>k</u>
Flooded coolant channel	1.32550
Voided coolant channel	1.38133
T _{fuel} + 65°F	1.32473
T _{moderator} + 95°F	1.32238
T _{coolant} + 65°F	1.32636

TABLE 0910-10

PRELIMINARY COMPARISON OF TEST PROBLEM RESULTS
OBTAINED BY ATOMICS INTERNATIONAL AND COMBUSTION ENGINEERING

	AI	CE
INITIAL k_{eff}	1.316	1.3255
RESONANCE ABSORPTION	0.2107	0.2065
FAST EFFECT		
TERM (a)	0.0377	0.04006
TERM (b)	0.530	0.5416

REACTIVITY COEFFICIENTS

FUEL TEMP	$\frac{10^{-6} \Delta k}{k^{\circ}\text{F}}$	-7.95	-8.95
COOLANT TEMP	$\frac{10^{-6} \Delta k}{k^{\circ}\text{F}}$	6.43	9.90
COOLANT VOID	$\Delta k/k$	0.0467	0.0421
MODERATOR TEMP	$\frac{10^{-5} \Delta k}{k^{\circ}\text{F}}$	-8.00	-2.48

B. ENGINEERING ANALYSIS

This phase of the program deals with the engineering analysis efforts required to establish a HWOCR design. Design selection necessitates evaluation of many concepts as well as determination of the characteristics and adequacy of a particular reference concept. Methods must be developed for engineering analysis work for both evaluation and design efforts. Results of analyses must be made available to other tasks as a guide for design and development. As work proceeds, experimental information will be required to supplement analytical efforts.

Initial analytical efforts have been based on earlier studies* of an organic cooled heavy water moderated reactor. These efforts were directed primarily towards rapid selection of an initial reference concept while surveying a limited number of concepts. As faster calculational methods become available, more extensive surveys will be initiated.

The Engineering Analysis section of this report is divided into the following sub-headings:

- 1) Development of a Criterion for Fuel Utilization: The Dynamic Utilization
- 2) Fuel Utilization for Selected HWOCR Cases
- 3) Coolant Mixing Studies
- 4) Thermal and Hydraulic Code Development
- 5) One- and Two-Pass Reactor Studies
- 6) Present Reference Design Characteristics.

* CEND-175, Organic Cooled Heavy Water Moderated Reactor Power Plant - Conceptual Design and Evaluation, Combustion Engineering, March 1963.

1. Development of a Criterion for Fuel Utilization: The Dynamic Utilization

The general term "fuel utilization" is applied to nuclear power reactors to designate their effectiveness in minimizing the drain on the natural resource of fissile isotope while supplying some specified demand for nuclear energy. One of the principal objectives of HWOCR is the improvement of fuel utilization over that afforded by current types of conventional reactors. A quantitative method of specifying fuel utilization is needed for the project, not only to measure the progress made toward this objective, but to serve as a means of evaluating alternative features of reactor design within the project. The purpose of this analysis is to set up such a method, and to reduce it to a figure of merit which can be applied to different reactor designs without ambiguity.

Because of the critical mass requirement, at typical power reactors an inventory of fissile isotope several times greater than the actual depletion in a year of power operation is maintained at all times. The increasing demand for nuclear reactors as a commercial power source is increasing the amount of fissile isotope required for inventory in the additional reactors built, as well as increasing the net fissile isotope consumed in nuclear reaction. Proper characterization of the fuel utilization of a particular reactor must take into account the fissile isotope destroyed by nuclear reaction (generally referred to as burnup), as well as that tied up in inventory. A problem, then, exists in that the ratio of the requirement to the burnup depends upon the rate of growth of the installed nuclear capacity--that is, upon the rate of growth of the nuclear power industry.

The difficulty is to arrive at a scheme for weighting the two types of fuel requirements--burnup and inventory--in an equitable manner. This is a most important consideration, since past deficiencies in characterizing fuel utiliza-

tion have resulted from failure to recognize the importance of the inventory requirement.

In the past, the exponential has been the simplest mathematical function by which the growth of the electric utility industry can be described. It is true that the doubling time has not remained precisely constant, and it is also true that ridiculous conclusions can be reached by the thoughtless extrapolation of an exponential; but it seems entirely legitimate to utilize the simplifying properties of the exponential, in order to arrive at a figure of merit for fuel utilization in a rapidly expanding nuclear power industry. This is the course followed here. It is important to recognize that the intent is only to generate a figure of merit in terms of the exponential function, and that this is not equivalent to predicting that the growth of the industry will in fact proceed along an exponential with a constant doubling time.

Two terms must first be defined, as follows. Assume a nuclear power industry which has been growing for a long (effectively infinite) time at an exponential rate characterized by a doubling time T_d ; the defined quantities are:

a) The Exponential Investment (EI) of natural uranium is the ratio of the number of tons of natural uranium which have been fed into the industry up to any time, t , for both burnup and inventory, to the number of megawatt years of electric energy produced up to the time, t . Obviously, the value of the EI will depend upon the doubling time, and it will be necessary to select an appropriate doubling time in order to arrive at a value of the EI which has a meaningful relationship to the fuel supply problem.

b) The Dynamic Utilization (DU) of natural uranium is the reciprocal

of the EI It is the DU which is proposed as the figure of merit, evaluated for a suitably selected doubling time

The remainder of this discussion will develop the two quantities defined above in terms of more familiar numbers, and will relate them to some other quantities of importance.

Let:

$C = C_0 e^{0.693 t/T_d}$ Installed capacity at time, t , in electric megawatts

S = Specific installed inventory metric tons of natural uranium required to provide the 235 in 1 Mwe of installed reactor capacity.

A = Ratio of total inventory (including ex-reactor) to in-reactor inventory.

F = Plant factor.

B = Specific burnup, in metric tons of equivalent natural uranium per megawatt-year of electrical energy produced.

W = Total electric energy produced from $t = 0$ to $t = T$

N = Total natural uranium put into the system for burnup plus inventory, from $t = 0$ to $t = T$.

Then:

$$W = \int_0^T FC_0 e^{0.693 t/T_d} dt = \frac{FC_0 T_d}{0.693} \left[e^{0.693 t/T_d} - 1 \right] \quad (1)$$

$$N = BW + SAC = \frac{BFC_0 T_d}{0.693} \left[e^{0.693 t/T_d} - 1 \right] + SAC_0 e^{0.693 t/T_d} \quad (2)$$

Assume:

$$e^{0.693 t/T_d} \gg 1, \text{ so}$$

$$EI = \frac{N}{B} = \frac{\left[\frac{BFC_0 T_d}{0.693} + SAC_0 \right] e^{0.693 t/T_d}}{\frac{FC_0 T_d}{0.693} e^{0.693 t/T_d}} = B + \frac{C_0 693 SA}{FT_d} \quad (3)$$

The applications of the EI and the DU are illustrated in Table 0910-11. In preparing the table, the intent has been to cover a wide range of reactor characteristics rather than to make a critical comparison of the utilization performances of different reactor types. The characteristics of the light water reactor were derived from the recent Oak Ridge evaluation study (ORNL-3686)*, while the two sets of fast breeder characteristics represent the extremes of the range reported in the Argonne evaluation of fast breeder studies by four different contractors (COO-279)[†]. In all of these cases it was necessary to arrive at some components of the utilization performance by inference from related numbers which were reported. The upper set of HWOCR numbers is taken from the current reactor physics reference characteristics in use on the project. These are essentially the same as those in the design study of the 500 Mwe plant (CEND-175)[§]. The lower sets of HWOCR numbers are from past studies on large HWOCR plants, which represent somewhat more advanced characteristics than those of the current reference.

* ORNL-3686, A Comparative Evaluation of Advanced Converters, Oak Ridge Laboratory, January 1965.

† COO-279, An Evaluation of Four Design Studies of a 1000 Mwe Ceramic Fueled Fast Breeder Reactor, Chicago Operations Office, U. S. Atomic Energy Commission, December 1964.

§ CEND-175, Organic Cooled Heavy Water Moderated Reactor Power Plant -- Conceptual Design and Evaluation, Combustion Engineering, March 1963.

TABLE 0910-II

DYNAMIC UTILIZATION FOR SEVERAL REACTORS

REACTOR	$\frac{SA}{F}$ METRIC TONS NATURAL U PER MWe	B METRIC TONS NATURAL U PER MWe-yr	EI, METRIC TONS NATURAL U PER MWe-yr $T_d =$			DU, MWe-yr PER METRIC TON NATURAL U; $T_d =$		
			5 yr	10 yr	∞	5 yr	10 yr	∞
LWR (ORNL-3686)	0.683	0.154	0.248	0.201	0.154	4.0	5.0	6.5
HWO CR (CURRENT REF)	0.318	0.087	0.131	0.109	0.087	7.6	9.1	11.5
<u>LARGE HWO CR:</u>								
ENRICHED U	0.23	0.067	0.099	0.083	0.067	10.1	12.1	14.9
NATURAL U	0.40	0.061	0.116	0.089	0.061	8.6	11.2	16.4
THORIUM (C.R.=0.95)	0.63	0.027	0.114	0.071	0.027	8.8	14.1	37.0
FBR RANGE (COO-279)	1.82	-0.037	0.215	0.089	NEGATIVE*	4.7	11.2	~1000*
	0.64	-0.064	0.024	NEGATIVE*	NEGATIVE*	42	~1000*	~1000*

* WHEN T_d APPROACHES OR EXCEEDS THE DOUBLING TIME OF THE BREEDER, THIS ANALYSIS IS NO LONGER APPLICABLE; THE DU IS THEN DETERMINED BY REPROCESSING LOSSES RATHER THAN NEUTRON PHYSICS AND BECOMES OF THE ORDER 1000, SO LARGE THAT RESOURCE DEPLETION IS NO LONGER A PROBLEM

It is evident from the table that in terms of the dynamic utilization it is possible to compare breeders and converters directly. It is also possible to cover combined systems -- such as converters producing plutonium inventory for fast breeders -- but that aspect of the subject will not be explored at this time.

As to the appropriate value of T_d to be used in a reactor evaluation, opinion may vary. However, most current projections of the nuclear power industry show doubling times less than ten years for some decades to come, and it seems likely that most evaluators would agree on a T_d range between 5 and 10 years.

It is useful to point out the relationship between the exponential investment and two other important quantities -- the lifetime fuel commitment and the fuel cost.

The life time fuel commitment -- the equivalent quantity of natural uranium used by a particular reactor type for inventory and burnup over the total life of a typical plant -- has sometimes been used as a criterion of fuel utilization. The specific lifetime commitment (i.e. the commitment per Mwe-year of electrical energy produced) is given by:

$$S.L.C. = B \frac{SA}{FL} \quad (4)$$

where L is the life of a typical plant, in years. Comparison with equation (3) shows that the exponential investment will have the same value as the S.L.C., if the doubling time of the installed capacity (T_d) happens to be equal to $0.693 L$. For the usually assumed lifetime of 30 years, this would happen for a doubling time of 20.8 years.

If the inventory expense can be represented by a fixed annual charge rate, R , against the cost of the material in inventory, then that component of the fuel cost which is due to the cost of the raw natural uranium is given by an expression having the same form as equation (3). Specifically, if the cost of natural uranium is U' dollars per metric ton, and the fuel cost component due to natural uranium costs is G' dollars per Mwe-year, then:

$$G' = U' \left(B + \frac{SAR}{F} \right) \quad (5)$$

Comparison with equation (3) shows that G' is just U' times the exponential investment when the latter is evaluated for a doubling time, T_d , of $0.693/R$:

$$G' = U' \quad (EI \text{ evaluated at } T_d = 0.693/R) \quad (6)$$

Or, more conveniently, if G is the fuel cost component due to natural uranium costs, in mills/kwh, and U is the cost of natural uranium in dollars per pound of U_3O_8 , then:

$$G = 0.297 U \quad (EI \text{ evaluated at } T_d = 0.693/R) \quad (7)$$

The tabulation below gives the annual charge rates (R) which, in equation (7), would go with the values of T_d of 5, 10, and 20 years. For the sake of example, the table also gives the values of G which would correspond to a natural uranium price of \$8 per pound of U_3O_8 with the values of EI previously listed for the large enriched-uranium HWO CR.

T_d (Yr)	Equivalent inventory charge rate (R , per yr)	EI metric tons Nat U/Mwe-yr	Fuel cost component due to Nat U cost (G) (mills/kwh)
5	0.139	0.099	0.23
10	0.069	0.083	0.20
20	0.035	0.075	0.18

The main significance of the above tabulation is to show that the selection of a doubling time (T_d) in the 5 to 10 year range, for evaluation of the EI and DU, will give about the same relative emphasis to burnup and inventory as fuel cost evaluations do.

2. Fuel Utilization for Selected HWO CR Cases

For the early guidance of the HWO CR design effort, fuel utilization calculations have been made for a few selected designs of 1000-Mwe reactors. Some of these do not represent practical cases, for some of the design characteristics were varied arbitrarily to show their effects on utilization.

Three different aspects of the fuel utilization are considered, each of which has a particular significance and a particular usefulness for the understanding of the subject. These aspects are described by the terms defined below:

a) The static reactor utilization is the number of megawatt days of thermal energy produced per net gram of fissile isotope burned.

b) The static system utilization is the number of megawatt days of thermal energy produced per net gram of fissile isotope burned or discarded. It differs from the static reactor utilization in that it takes into account the fissile isotope which is made unavailable when it is discarded in diffusion plant tailings or spent reactor fuel, at enrichments too low for economic recovery.

c) The dynamic system utilization is the number of megawatt years of electrical energy produced per metric ton of equivalent natural uranium used up (for burnup, inventory and discard) in a system of nuclear plants whose installed capacity and annual output are increasing

exponentially with some specified doubling time. This is the dynamic utilization which is defined and discussed in the preceding section (II.B.1). Its essential difference from the static system utilization defined above is the inclusion of the inventory requirement. The change of units, from megawatt days (thermal) per gram of fissile isotope to megawatt years (electrical) per metric ton of equivalent natural uranium, is incidental, and is made for the purpose of relating the dynamic utilization more directly to the natural uranium resource.

In all of the above definitions, the term "burned" means fissioned or converted to a non-fissile isotope by neutron capture. All fissile isotopes — U^{235} , U^{233} , Pu^{239} , and Pu^{241} — are regarded as equivalent on a gram-for-gram basis. The conversion from grams of fissile isotope to equivalent tons of natural uranium is made simply by dividing by 7.2×10^3 .

The numbers presented here are derived from the results of the Reactor Physics activity: Specifically, from the results of some of the INGEN-2(CRUMB) calculations described in section II.A. of this subtask. The cases which have been utilized, and the characteristics of the reactors to which they apply, are listed in Table 0910-12. It has been necessary in using these results to apply them to some sort of a fuel management program. The very simple case which has been assumed has sometimes been designated "uniformly graded irradiation". This is equivalent to the assumption that the refueling cycle has reached equilibrium, and that refueling is carried out continuously at all points within the reactor, with the rate adjusted at each point to be proportional to the local fission rate, so that the fuel composition is uniform throughout the reactor. This is expected to be approximately equivalent to bi-directional push-through. For enrichments which give rather short reactivity lifetimes,

TABLE 0910-12. CHARACTERISTICS OF REACTORS USED IN ANALYSIS

ALL REACTORS HAVE THE NOMINAL POWER CAPACITY OF 1000 MWe, AND CONSIST OF 1200 PRESSURE TUBES WITH 17.3-FOOT FUELED LENGTHS. THE SIDE AND END REFLECTORS ARE 1 FOOT THICK. THE AVERAGE MODERATOR TEMPERATURE IS 190 °F, THE COOLANT TEMPERATURE 650 °F, AND THE FUEL TEMPERATURE 1200 °F. THE FUEL AREA PER PRESSURE TUBE IS 6.0 SQUARE INCHES OF UO₂.

INGEN-2 CASE NO.	FEED ENRICHMENT, Wt% U235	SAP CLAD THICKNESS FOR LARGE FUEL ELEMENT, INCHES	SAP PRESSURE TUBE THICKNESS, INCHES	ZIRC. CALANDRIA TUBE THICKNESS, INCHES	VOLUME FRACTION ORGANIC IN PRESSURE TUBE	VOLUME RATIO, MODERATOR/FUEL
161	0.720	0.020	0.110	0.040	0.34	11.9
162	1.100	0.020	0.110	0.040	0.34	11.9
163	1.500	0.020	0.110	0.040	0.34	11.9
225	1.100	(0.5) (0.020)	(0.5) (0.110)	(0.5) (0.040)	0.34	11.9
226	1.100	(0.1) (0.020)	(0.1) (0.110)	(0.1) (0.040)	0.34	11.9
224	1.100	0.020	0.110	0.040	0.34	20.0
144	1.100	0.020	0.110	0.040	0.43	11.9

the bi-directional push-through may yield a somewhat longer reactivity lifetime than the uniformly graded irradiation; but for enrichments which yield rather long lifetimes (in the order of 20,000 Mwd/T), the situation will probably be reversed.

a. Waste of U^{235} by Discard

If the diffusion plants for uranium enrichment are operated under anything like attractive economic conditions, they must discharge, as tailings, uranium which contains substantial quantities of U^{235} . For the purposes of this analysis, it is assumed that the plants, at the time which is of interest, will discharge tailings containing 0.25% U^{235} . This means that about 1/3 of the U^{235} fed to the plants in the form of natural uranium is wasted in the tailings, and that for every gram of enriched U^{235} which is burned in a reactor, about 0.53 gm have been discarded in diffusion plant tailings. This would be essentially the case for light water reactors. The U^{235} content of the spent fuel from these reactors is usually well above 0.25%, so it is economical to re-enrich the spent fuel, either by sending it back to the diffusion plants or by adding highly enriched U^{235} to it. Whichever course is taken, the material balance is the same, and so far as wastage in diffusion plant tailings is concerned, it is as though the reactor were fed with highly enriched U^{235} . Hereafter, this case will be designated as " U^{235} recycle" (see Fig. 0910-8). In this cycle, there is no waste of U^{235} through discard in the spent reactor fuel, since all of the spent fuel is re-enriched and reused.

For the case of the heavy water reactors, the U^{235} content of the spent fuel is almost always below 0.25% if the exposure has been long enough to be economically attractive; this is usually true whether the reactor is using

natural or enriched uranium as feed. The spent fuel, therefore, is discarded (after the plutonium has been recovered), and whatever U^{235} remains in the fuel is wasted. In the heavy water reactor there is consequently a feed enrichment which will minimize the sum of the U^{235} wastage in diffusion plant tailings, plus the wastage in discarded fuel. It is believed that it will usually be profitable to use a feed enrichment high enough to burn the U^{235} content down to 0.1%, and possibly less; but this of course will depend on many considerations.

Figure 0910-8 shows the wastage of U^{235} through discard as a function of feed enrichment, for two different discharge enrichments. If one assumes, for example, that in some particular reactor a feed enrichment of 1.0% will give an exposure lifetime such that the U^{235} content of the spent fuel is 0.1%, then one might examine the possibilities of decreasing the wastage by increasing the feed enrichment, and, therefore, burning the fuel to a lower discharge enrichment. If, for example, the discharge enrichment could be decreased to 0.05% by increasing the enrichment to about 1.15% (Line A, Figure 0910-8), there would indeed be an improvement. If an increase of feed enrichment to, say, 1.6% were required (Line C, Figure 0910-8), then the wastage would be increased. If an increase of feed enrichment to about 1.3% were required (Line B, Figure 0910-8), then the wastage would stay about the same. It is, of course, to be remembered that the wastage of U^{235} by discard is only one consideration in the utilization picture, and that changes in feed enrichment will have other effects that may be equally or more important. For example, a higher feed enrichment and a resultant longer fuel exposure will increase the average fission product burden of the reactor, and, hence, will reduce the average conversion ratio.

b. Use of Enrichment to Reduce Discard Wastage

Table 0910-13 shows a specific case in which the enrichment has been changed over a rather wide range. The reactor remains essentially the same for the three cases shown in the table, and only the feed enrichment of the fuel is changed. This particular reactor will barely operate on natural uranium. The discharge exposure of 3,000 Mwd/MTU leaves the spent fuel with a U^{235} content of 0.442%. In practice, this spent fuel would probably be re-enriched and reused, but for the sake of this illustration, the assumption is that it is discarded after recovery of plutonium. The static system utilization is quite low -- 0.55 Mwd/gm. When the feed enrichment is increased to 1.1%, the exposure increases to 15,900 Mwd/MTU, and the discharge enrichment decreases to 0.142%. The static utilization of the system increases to 1.60 Mwd/gm. When the enrichment is increased further, to 1.5%, the static system utilization decreases to 1.49 Mwd/gm. There are at least three contributing reasons for the decrease:

- 1) The decrease of wastage in the spent fuel is compensated, more or less, by increased wastage in diffusion-plant tailings.
- 2) The higher fission product burden decreases the conversion ratio of the reactor.
- 3) A larger fraction of the total fissions occurs in the plutonium isotopes which yield lower values of η than U^{235} , this also decreases the average conversion ratio.

The effects of the last two processes named above may be seen by examining the change in the reactor utilization as the enrichment is increased. This change, from 2.96 for the natural uranium case to 2.20 for the 1.5% case, is quite substantial.

TABLE 0910-13. USE OF ENRICHMENT TO REDUCE DISCARD OF U235 IN SPENT FUEL

(1 GRAM U235 BURNED YIELDS 0.78 MWD)

CASE NO.	FEED ENRICHMENT, Wt % U235	DISCHARGE EXPOSURE, MWD/MTU	DISCHARGE ENRICHMENT, Wt % U235	FISSILE Pu IN DISCHARGE, Wt %	STATIC UTILIZATION	
					REACTOR	SYSTEM
					MWD PER GRAM FISSILE ISOTOPE BURNED	MWD PER GRAM FISSILE ISOTOPE BURNED OR DISCARDED
161	NATURAL	3,000 *	0.442	0.176	2.96	0.55
162	1.10	15,900	0.142	0.305	2.44	1.60
163	1.50	24,000	0.100	0.307	2.20	1.49

* THIS CASE HAS BEEN FUDGED SLIGHTLY FOR THE SAKE OF ILLUSTRATING THE POINT. THE ACTUAL INGEN-2 (CRUMB) CALCULATIONS INDICATE THAT THE REACTOR WILL NOT QUITE RUN ON NATURAL URANIUM. THE ASSUMPTION HAS BEEN MADE THAT REACTIVITY WOULD BE SLIGHTLY HIGHER, AND THAT IT WOULD RUN. IF IT WILL RUN AT ALL, IT WILL GIVE A DISCHARGE EXPOSURE OF ABOUT 3,000 MWD/MTU, BECAUSE OF THE "PLUTONIUM HUMP" IN THE BURNUP CURVE.

The three enrichment cases examined above are not enough to establish an optimum enrichment for highest utilization, but it probably would be not far from the 1.1% point. At any rate, the enrichment of 1.1% has been adopted as the enrichment at which other characteristics will be examined in the remainder of this analysis. The fuel exposures which this enrichment yields (in the range of 15,000 Mwd/MTU) are lower than the 20,000 Mwd/MTU which is usually assumed in design studies, but there is no reason to assume that the utilization must be evaluated at the enrichment which yields near-optimum short-term economics, provided only that the reactor is capable of generating full power at both enrichments.

c. Effect of Parasitic Absorption on Utilization

The parasitic absorption of neutrons in moderator and structural material reduces the fuel utilization, both because it reduces the conversion ratio and because it increases the general level of feed enrichment for a given fuel exposure, and, thus, increases the wastage of U^{235} by discard. No doubt there is a different optimum feed enrichment for each value of parasitic absorption, but for this study the feed enrichment has been held constant at 1.1%. The parasitic absorption is expressed as the fraction of neutrons absorbed parasitically when the enrichment of the fuel is equal to the feed enrichment (1.10%). This absorption fraction is the fraction calculated by a 4-group analysis, and it includes both epithermal and thermal absorptions.

Table 0910-14 gives reactor utilizations and system utilizations for three different values of the parasitic absorption fraction. Only the highest of these fractions corresponds to a practical reactor design. The main geometrical features of that design are summarized in Table 0910-15, and the

TABLE 0910-14. EFFECT OF PARASITIC ABSORPTION ON UTILIZATION

(FEED ENRICHMENT = 1.10 Wt % U235)

CASE NO.	PARASITIC ABSORPTION FRACTION	DISCHARGE EXPOSURE, MWD/MTU	DISCHARGE ENRICHMENT, Wt % U235	FISSILE Pu IN DISCHARGE, Wt %	STATIC UTILIZATION	
					REACTOR	SYSTEM
					MWD PER GRAM FISSILE ISOTOPE BURNED	MWD PER GRAM FISSILE ISOTOPE BURNED OR DISCARDED
162	0.086	15,900	0.142	0.305	2.44	1.60
225	0.068	17,400	0.114	0.304	2.55	1.75
226	0.053	19,600	0.080	0.302	2.73	1.96

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TABLE 0910-15. BREAKDOWN OF PARASITIC ABSORPTION FRACTION FOR CASE 162

(ENRICHMENT = 1.10 Wt % OXIDE FUEL)

ITEM	QUANTITY	FRACTION OF NEUTRONS ABSORBED	INCREASE OF QUANTITY CORRESPONDING TO INCREASE OF 0.0075 IN TOTAL PARASITIC ABSORPTION FRACTION *
D ₂ O	$\frac{\text{VOL. MOD.}}{\text{VOL. FUEL}} = 11.9$	0.0150	--
H ₂ O IN D ₂ O	0.25 %	0.0055	0.34 %
CALANDRIA TUBES (ZIRC.)	0.040 IN. THICK	0.0065	0.046 IN.
ORGANIC	0.34 V ^F	0.0270	0.094 V ^F
FUEL CLAD (SAP)	0.020 IN. THICK	0.0105	0.014 IN.
PRESSURE TUBES (SAP)	0.110 IN. THICK	<u>0.0212</u>	0.039 IN.
TOTAL		0.0857 *	

* THE FRACTION ABSORBED IN FUEL (U235 AND U238 FOR THIS CASE) IS $1.0000 - 0.0857 = 0.9143$.

** FOR ALL THE ABSORBERS LISTED EXCEPT ORGANIC, THIS WOULD CORRESPOND TO A REDUCTION OF ABOUT 0.05 IN THE STATIC SYSTEM UTILIZATION WHEN THE BASE VALUE OF THE STATIC SYSTEM UTILIZATION IS IN THE NEIGHBORHOOD OF 1.6; FOR ORGANIC THE EFFECT ON SYSTEM UTILIZATION IS ABOUT TWICE AS GREAT (SEE FIG 0910-10.)

parasitic absorption is broken down among the various components of the reactor composition. The two lower absorption fractions shown in Table 0910-14 correspond to reactors which are identical to that of Table 0910-15, except that, arbitrarily, the number densities of aluminum (SAP) and zirconium have simultaneously been reduced by 50% (Case 225) and 90% (Case 226).

It is clear that the reactor utilization is rather sensitive to the parasitic absorption fraction and that the system utilization is even more sensitive to it. These results can be understood better by reference to Figure 0910-9, which shows the variation of the U^{235} content and the fissile plutonium content of the fuel with exposure. As the parasitic absorption fraction is changed, the resulting change in these curves is almost imperceptible. The important result of a decrease of parasitic absorption is to lengthen the exposure of the fuel. After an exposure of about 10,000 Mwd/T, the plutonium isotopes have almost reached their equilibrium concentrations and further exposure does not change appreciably the amount of plutonium recovered from the spent fuel. Since the U^{235} in the spent fuel is to be discarded, the system utilization increases linearly with discharge exposure (i.e. with the amount of energy extracted from each unit of fuel) once the fissile plutonium content approximates its equilibrium value. In figuring the reactor utilization, on the other hand, credit is taken for the U^{235} in the spent fuel; therefore, the change in the reactor utilization with discharge exposure is smaller, percentage-wise, than the change in the system utilization.

d. Effects of Other Variables

Other than the feed enrichment and the parasitic absorption fraction, the remaining important physics design variables are the coolant fraction and the moderator/fuel ratio. Changes in either of these quantities will, of course,

affect the parasitic absorption fraction, but will also affect the neutron energy spectrum, in particular the ratio of epithermal to thermal neutron fluxes, and, therefore, the ratio of neutron absorption in U^{238} to absorption in U^{235} for a given fuel enrichment. In Table 0910-16 two cases have been added to those previously considered in Table 0910-14. One of these involves an increase in the coolant fraction from 0.34 to 0.43, while the other involves an increase in the moderator/fuel ratio from about 12 to 20. In Figure 0910-10 the static utilizations for all five of these cases have been plotted as functions of the parasitic absorption fraction. For the change in coolant fraction, the loss in both reactor utilization and system utilization appears to be about twice that predicted by extrapolation of the curves of parasitic absorption fraction. The deviation from the curve is much more pronounced for the reactor utilization in the case of the increased moderator/fuel ratio. This is because the resonance absorption in U^{238} has been decreased, and therefore, over the life of the fuel, the fissile and fertile isotopes compete less effectively with the parasitic absorbers that would be expected on the basis of the feed enrichment. This is to say, when an important alteration is made in the neutron energy spectrum, the parasitic absorption fraction at the beginning of life is no longer a valid indicator of the average parasitic loss over the fuel lifetime. A secondary reason for the departure from the curve is that the diffusion length is longer in the more dilute reactor, and there is, therefore, an extra loss of neutrons by leakage. An adjustment has been made for the latter effect in Figure 0910-10, as indicated. Surprisingly enough, the rather large deviation of the dilute reactor from the reactor utilization curve

TABLE 0910-16. EFFECT OF PARASITIC ABSORPTION, COOLANT FRACTION, AND MODERATOR/FUEL RATIO ON UTILIZATION
(FEED ENRICHMENT = 1.10 Wt % OXIDE FUEL)

CASE NO.	PARASITIC ABSORPTION FRACTION	COOLANT, V _F	VOLUME FRACTION, MODERATOR/FUEL	DISCHARGE EXPOSURE, MWD/MTU	DISCHARGE ENRICHMENT, Wt %U235	FISSILE Pu IN DISCHARGE, Wt %	STATIC UTILIZATION	
							REACTOR	SYSTEM
							MWD PER GRAM FISSILE ISOTOPE BURNED	MWD PER GRAM FISSILE ISOTOPE BURNED OR DISCARDED
162	0.086	0.34	11.9	15,900	0.142	0.305	2.44	1.60
225	0.068	0.34	11.9	17,400	0.114	0.304	2.55	1.75
226	0.053	0.34	11.9	19,600	0.080	0.302	2.73	1.96
144	0.096	0.43	11.9	14,900	0.172	0.296	2.36	1.49
224	0.099	0.34	20.0	15,400	0.117	0.247	2.09	1.46

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is not reproduced in the system utilization curve. This is because the more dilute reactor burns a larger fraction of the U^{235} out of the fuel; i.e., the discharge enrichment is lower, as shown in Table 0910-15, and, therefore, most of the loss in reactor utilization is compensated by a decrease in the wastage of U^{235} by discard.

It is concluded that design changes which make appreciable changes in reactor physics characteristics other than simple absorption ratios are apt to affect the utilization results in rather subtle ways, and these affects must be investigated specifically when such design changes are contemplated. Nevertheless, in the case of coolant fraction, it appears unlikely that volume fractions much outside the range 0.34 to 0.43, which was investigated here, will be used. For the moment, it appears reasonable to use the rule of thumb that, changes in coolant fraction will have an effect on the static utilization which is about twice as great as would be inferred from the change in parasitic absorption fraction alone.

e. Change of Fuel Material

Calculations for carbide fuel have not yet been analyzed for utilization, but, for the sake of illustrating a point, the assumption will be made that the approximate utilization attainable at a given feed enrichment will be simply a function of the parasitic absorption fraction.* Thus, if uranium oxide were replaced in case 162 by uranium carbide, without changing any other feature of the design, the parasitic absorption fraction would decrease in approximately the inverse ratio of the uranium densities in the two fuels: $1/1.4 = 0.71$. The result would be to increase the static system utilization from 1.60 to

* This despite the reservations expressed in the paragraph immediately above.

about 1.82 (from Fig. 0910-10). Since the power output of the oxide fuel design is presumably limited by surface temperature considerations rather than by the central temperature of the fuel, one would expect that such a simple replacement of oxide by carbide would result in a decrease of the specific power in the same inverse ratio of the uranium densities; i.e., 0.71. The net result of the increase of static utilization and the decrease of specific power on the dynamic utilization is shown in Table 0910-17.

It is, of course, unlikely that one would change from an oxide design to a carbide simply by substituting one fuel for the other. One would do something in the way of finning or increasing the coolant fraction, or both, to increase the power output of the carbide design. Some increase in the volume ratio moderator/fuel might also be required for reactivity reasons. The net result would probably be a decrease in the static utilization below that indicated for the carbide case in Table 0910-17 and an increase in the specific power, to give some net increase in the dynamic utilization over that for the oxide fuel. The point of Table 0910-17 is to illustrate the importance of specific power, and to indicate that particular attention should be given to this characteristic in the carbide designs.

f. Computation of Utilization

Table 0910-18 has been included to show how the values of static utilization have been computed from the results of the INGEN-2 (CRUMB)* calculations. The values of the discharge exposure were computed as described in the first section of this discussion, on the basis of uniformly graded irradiation. It should be clear from Table 0910-18 that the system utilization is quite sensitive

* See Section 0610, Reactor Physics, for code references.

TABLE 0910-17. DYNAMIC UTILIZATION (DOUBLING TIME OF INSTALLED CAPACITY TAKEN AS 7 YEARS)
(1.10 Wt % FEED ENRICHMENT)

	UO ₂ FUEL	U DENSITY INCREASED 40 Wt %
STATIC SYSTEM UTILIZATION, MWD/GRAM FISSILE ISOTOPE BURNED OR DISCARDED	1.60	1.82
SPECIFIC BURNUP, B * = METRIC TONS OF NATURAL U REQUIRED TO SUPPLY BURNUP FOR 1 MWe-YR (TAILS LOSSES INCLUDED)	0.0882	0.0776
SPECIFIC POWER = MWe OF RATED OUTPUT PER TON U IN REACTOR	4.42	3.16
SPECIFIC INVENTORY, S ** = METRIC TONS NATURAL U REQUIRED TO FURNISH INVENTORY FOR 1 MWe INSTALLED CAPACITY (TAILS LOSSES INCLUDED)	0.510	0.716
EXPONENTIAL INVENTORY REQUIREMENT, I = METRIC TONS NATURAL U INVESTED IN INVENTORY PER MWe OF ENERGY PRODUCED, WHEN INSTALLED CAPACITY IS DOUBLING EVERY 7 YEARS $= \frac{0.693 SA}{7 F} ***$	0.0787	0.1110
B + I	0.167	0.189
DYNAMIC UTILIZATION = $\frac{1}{B + I} \frac{\text{MWe-YR}}{\text{MTU (NAT.)}}$	6.00	5.30

* THE SPECIFIC BURNUP IS SIMPLY THE RECIPROCAL OF THE STATIC SYSTEM UTILIZATION (SSU), WITH A CONVERSION FACTOR TO CHANGE UNITS AND TO CONVERT FROM THERMAL TO ELECTRICAL ENERGY. FOR THIS CASE THE NET THERMAL EFFICIENCY WAS ASSUMED TO BE 36%, AND THE RELATIONSHIP IS: $B = 0.141/(\text{SSU})$

** CALCULATED ON ASSUMPTION THAT THE REACTOR CALCULATED PRODUCES 1000 MWe

***A, THE RATIO OF TOTAL INVENTORY TO IN-PILE INVENTORY, WAS TAKEN AS 1.25; F, THE PLANT FACTOR, TAKEN AS 0.80

TABLE 0910-18. COMPUTATION OF STATIC UTILIZATION

A. CASE NUMBER	161	162	163	225	226	224	144
B. DISCHARGE EXPOSURE, MWD/KgU	3.00	15.9	24.0	17.4	19.6	15.4	14.9
C. FEED ENRICHMENT, gm U235/KgU	7.200	11.00	15.00	11.00	11.00	11.00	11.00
D. DISCHARGE ENRICHMENT, gm U235/KgU	4.423	1.42	1.00	1.14	0.80	1.17	1.72
E. Pu 239 + Pu 241 IN DISCHARGE, gm/KgU	1.764	3.05	3.07	3.04	3.02	2.47	2.97
F. NET F. I. BURNED = C - (D + E), gm/KgU	1.013	6.53	10.93	6.82	7.18	7.36	6.31
G. REACTOR UTILIZATION + B/F, MWD/gm	2.96	2.44	2.20	2.55	2.73	2.09	2.36
H. GRAMS U235 FED TO SYSTEM IN NATURAL U PER KgU REACTOR *	7.200	13.00	19.15	13.00	13.00	13.00	13.00
J. NET F. I. BURNED OR DISCARDED IN SYSTEM = (H - E), GRAMS F. I./KgU FED TO REACTOR	5.436	9.95	16.08	9.96	9.98	10.53	10.03
K. SYSTEM UTILIZATION = B/J, MWD/gm	0.552	1.60	1.49	1.75	1.96	1.46	1.49

* $H = \frac{(E - 2.5)}{(7.2 - 2.5)} \times 7.2$ FOR THE ASSUMPTIONS USED HERE, WHERE E IS THE FEED ENRICHMENT, IN GRAMS U235/KgU

to the value of the discharge exposure, and that any careful calculation of the utilization will have to give close attention to this quantity, and will have to be consistent with the actual fuel management program as postulated for the reactor. The computation of the dynamic utilization should be clear from Table 0910-17.

g. Implications for the Reactor Design

The breakdown of the parasitic absorption fraction given in Table 0910-15 should be useful in connection with decisions which involve either increasing or decreasing the amounts of parasitic materials in the reactor design. Note that the last column of the table gives the amount by which any one of the absorbing components could be increased if one were willing to accept an increase of 0.0075 in the parasitic absorption fraction. This increase, if it occurred in any of the materials listed, except D₂O or organic, would result in a decrease of static system utilization by about 0.05 Mwd/gm, provided the static system utilization were initially in the neighborhood of 1.6 Mwd/gm. If the increase in parasitic absorption were due to an increase in organic fraction, the effect on system utilization would be about twice as great. If one were interested in adding zirconium, SAP, or aluminum to the reactor for purposes other than those listed in Table 0910-15, the results would be the same on a gram-for-gram basis, except that some allowance would have to be made for a difference in disadvantage factor. For such things as pressure tube supports, which would cross the reactor lattice in a direction perpendicular to the pressure tubes, the physics calculations indicate that the average thermal neutron flux would be about 20% higher than that in the pressure tubes or calandria tubes. Therefore, the addition of one gram of

zirconium or aluminum per pressure tube, as a transverse support, would have about the same effect on utilization as 1.2 gm * of the same material added to the wall of each pressure tube or calandria tube.

The numbers given in the last column of Table 0910-15 are derived from the slope of the curve of utilization versus parasitic absorption fraction. It is clear from Figure 0910-10 that the slope is not constant as the absorption fraction is changed, and it is for this reason that the base value of the utilization must be stated when giving a simple relationship between quantity of material and utilization, as is done in Table 0910-15. In order to give an idea of the total effects of parasitic absorption on utilization, Figure 0910-11 has been included; this is a replot of the system utilization on the same sheet with a bar graph showing the parasitic absorption of each reactor component.

3. Coolant Mixing Studies

A major problem in reactor design is to determine the influence of fuel element interchannel coolant mixing on reactor thermal performance. Studies have been initiated, therefore, to determine the possible influence of mixing on the maximum surface temperature of 37-rod fuel elements. Thermal performance calculations using various degrees of coolant mixing between subchannels were performed for the "dry" (0.34 organic volume fraction) and the "wet" (0.42 organic volume fraction) 37-rod fuel element design. Due to the absence of specific experimental information on this type of fuel element, the inter-subchannel cross-flow rate was retained as a parameter in the calculations.

Figure 0910-12 shows a typical 1/6 section of the fuel elements used for calculations. The whole bundle consists of 31 rods of 0.513-inch OD and

6 rods of 0.313 inch OD. The large rods have one spiral-wrapped spacer. The direction of the spiral is the same for all rods. The small rods do not have a spacer spiral. As shown in Figure 0910-12, there are four different types of subchannels.

The computer code which was used for the calculations employs an inter-subchannel cross-flow rate, g , which is used to describe mixing. It was assumed that between a large rod and an unwrapped small rod, as well as at the outside of the peripheral large rods, the cross-flow rate is half that between large central rods, as indicated in Figure 0910-12. The neutron flux depression within the fuel bundle was assumed to be symmetric and all rods of one fuel "ring" were considered to have the same relative radial power peaking factor.

All computer cases were run for the following one-pass reactor conditions:

Power-to-coolant	3000 Mwt
Number of pressure tubes	2592
Mean reactor coolant outlet temp	750°F
Coolant temp rise through reactor	86°F

Figures 0910-13 and 0910-14 show results of the calculations. The ratio of subchannel coolant temperature rise to pressure tube coolant temperature rise is plotted against the inter-subchannel cross-flow rate. It is seen that the dry element displays a very large coolant rise ratio for subchannel 8 for no mixing. This is due to the combination of relatively large heat input into subchannel 8 and an unfavorable flow distribution among the subchannels, with subchannels 6, 7, and 12 using up a relatively large portion of the total flow. Corresponding coolant rise ratios for the wet

element are much lower due to the more open lattice and more uniform sub-channel flow distribution. It is also apparent that thermal performance can be improved significantly if mixing can be obtained. Flow tests will be required to confirm this. It is expected that a given amount of mixing would be attained more easily for wetter elements, because of greater fuel element spacing.

In Figure 0910-15 the reduction of the maximum surface temperature (considering all channels in the dry and wet elements) is indicated. Although there is great incentive to promote mixing by spiral or fin additions, it will have to be determined how costly it is in terms of additional pressure drop and increased neutron absorption.

4. Thermal and Hydraulic Code Development

Due to the nature and scope of the HWOCR Project, two distinct types of computer programs are required. Evaluation programs are needed for scoping a large number of designs and operational parameters. Other programs are needed to examine in detail a particular design as to its thermal and hydraulic adequacy. Development of both types of programs has been started.

Thermal and hydraulic performance was evaluated by hand calculations which required simplified assumptions and techniques at the outset of the project. The electronic computer programming development was initiated with the procurement of a program (U3 Code) developed by the Canadians at Chalk River. This code had been developed as an evaluation tool for fuel bundles in test loops.

The program as developed performed multichannel rodged fuel bundle calculations for a single-pass organic cooled system. A chopped cosine axial power distribution was employed with inherent capabilities for evaluating flow redistribution and inter-subchannel mixing effect. The program is capable of analyzing almost any multi-rodged fuel bundle configuration. The code determines coolant temperature and velocity, and surface temperature, at each axial position for each subchannel.

In order to increase the flexibility of this program, revisions were made to many of the calculational techniques. Provisions were incorporated to permit the use of an arbitrary axial power distribution and for the determination of maximum fuel temperature. The dimensional units were converted from the c.g.s. system to the f.p.s. system. The capability for handling a two-pass hydraulic system was added. A flow search procedure was added which would require the program to search for the flow rate which would result in a desired maximum surface temperature. Also, a provision was devised for the elimination of non-varying input parameters for successive problems in a series.

The modified U3 code is now available for reactor design work and certain evaluation studies. In order to provide for evaluation studies with different input-output requirements, a new program, THEME-1, is being developed and is in the final stages of preparation. This program, which is for a one-pass hydraulic system, will perform a hot subchannel calculation while maintaining the flow redistribution characteristic of the U3 code. The input preparation is not extensive compared to the U3 code, with reactor exit temperature, maximum surface temperature, and either maximum velocity or temperature rise, as the main input parameters. The program will then determine the number of pressure tubes which correspond to the input parameters.

A variation of this program, THEME-2, is in the development stage. This program will conform to THEME-1 with the option of either a one- or two-pass hydraulic system. Also, it will have a subroutine for computing reactor costs associated with the particular design being evaluated.

5. One and Two-Pass Reactor Studies

In the study of various coolant flow arrangements, one- and two-pass reactors were considered. In this one/two-pass study, a total of 57 reactor plants were compared, of which 30 were based on one-pass and 27 on two-pass reactor designs. All reactor concepts are based on a maximum fuel elements surface temperature of 850°F. Costs are in differential form and are based on a constant plant output of 1000 Mwe. Reactor thermal power ratings were determined using the net cycle efficiency of the plant, considering such variable requirements as primary coolant and feed pump loads. Results are presented in Table 0910-19. In Figures 0910-16 and 0910-17, cost comparisons are made on the basis of a constant maximum velocity (30 ft/sec) in the core. In Figure 0910-16, the organic volume fraction in the flow tube was held constant at 0.43 as the reactor exit temperature was varied. In Figure 0910-17, the reactor exit temperature was fixed at 750°F, while the organic fraction was varied. Figure 0910-14 shows the type of oxide fuel element used for reactor calculations.

6. Discussion

The data in Table 0910-19 are based on fuel cycle costs using \$100/kg fabrication costs. For cases DEF and MNO in Table 0910-19, fuel cycle costs based on \$50/kg are shown for comparison. Velocity was not held constant for the cases run, since it was not clear that a maximum permissible velocity

TABLE 0910-19

FUEL CYCLE COST DATA

0910-63

	A	A	A	B	B	B	C	C	C	D	D	D	E	E	E
Steam Pressure (psig)	600	900	1200	600	900	1200	600	900	1200	600	900	1200	600	900	1200
No. Passes	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
Reactor Exit Temp (°F)	750	750	750	750	750	750	750	750	750	750	750	750	750	750	750
Organic V ^F	0.340	0.340	0.340	0.340	0.340	0.340	0.340	0.340	0.340	0.430	0.430	0.430	0.430	0.430	0.430
Reactor ΔT (°F)	81	81	81	84	84	84	87	87	87	90	90	90	100	100	100
Reactor ΔP (psi)	530	530	530	150	150	150	73	73	73	420	420	420	140	140	140
Thickness Of Flow Tube (in.)	0.210	0.210	0.210	0.076	0.076	0.076	0.046	0.046	0.046	0.173	0.173	0.173	0.070	0.070	0.070
Thickness Of Clad.(in.)	0.026	0.026	0.026	0.017	0.017	0.017	0.015	0.015	0.015	0.024	0.024	0.024	0.017	0.017	0.017
Equilibrium Fuel Enrichment (Wt % U ²³⁵)	1.434	1.434	1.434	1.279	1.279	1.279	1.230	1.230	1.230	1.456	1.456	1.456	1.337	1.337	1.337
No. Of Flow Tubes	1093	1031	1017	1799	1706	1653	2635	2483	2426	706	668	657	1049	999	977
Max Core Velocity ft/sec	68	68	68	36	36	36	25	25	25	66	66	66	37	37	37
MWt	3380	3190	3145	3085	2925	2835	3040	2865	2800	3260	3085	3035	3055	2910	2845
<hr/>															
Costs, Mills/kwh															
Reactor Capital	0.467	0.445	0.441	0.712	0.680	0.662	0.989	0.936	0.916	0.330	0.317	0.313	0.452	0.434	0.426
System Operation & Capital	0.369	0.376	0.374	0.086	0.098	0.106	0.020	0.047	0.043	0.314	0.314	0.329	0.052	0.063	0.079
D ₂ O	0.199	0.188	0.186	0.316	0.300	0.292	0.454	0.434	0.424	0.130	0.130	0.130	0.190	0.180	0.180
Organic Makeup	0.266	0.252	0.248	0.242	0.229	0.220	0.236	0.223	0.218	0.319	0.301	0.297	0.293	0.280	0.273
Fuel Cycle \$100/kg	1.450	1.370	1.350	1.230	1.160	1.130	1.180	1.110	1.090	1.430	1.350	1.330	1.260	1.200	1.170
Total \$100/kg	2.751	2.631	2.599	2.586	2.467	2.410	2.879	2.750	2.691	2.523	2.412	2.399	2.247	2.157	2.128
Fuel Cycle \$50/kg	---	---	---	---	---	---	---	---	---	1.050	1.000	1.000	0.910	0.860	0.840
Total \$50/kg	---	---	---	---	---	---	---	---	---	2.224	2.142	2.148	1.987	1.907	1.887

TABLE 0910-19 (Cont.)

FUEL CYCLE COST DATA

	F	F	F	G	G	G	H	H	H	I	I	I	J	J	J
Steam Pressure (psig)	600	900	1200	600	900	1200	600	900	1200	600	900	1200	600	900	1200
No. Passes	1	1	1	1	1	1	1	1	1	1	1	1	2	2	2
Reactor Exit Temp (°F)	750	750	750	780	780	780	780	780	780	780	780	780	750	750	750
Organic V	0.430	0.430	0.430	0.430	0.430	0.430	0.430	0.430	0.430	0.430	0.430	0.430	0.340	0.340	0.340
Reactor ΔT (°F)	115	115	115	63	63	63	68	68	68	74	74	74	147	147	147
Reactor ΔP (psi)	24	24	24	400	400	400	150	150	150	65	65	65	1100	1100	1100
Thickness Of Flow Tube (in.)	0.040	0.040	0.040	0.180	0.180	0.180	0.080	0.080	0.080	0.047	0.047	0.047	0.045	0.045	0.045
Thickness Of Clad. (in.)	0.015	0.015	0.015	0.025	0.025	0.025	0.018	0.018	0.018	0.015	0.015	0.015	0.056	0.056	0.056
Equilibrium Fuel Enrichment (Wt % U235)	1.274	1.274	1.274	1.456	1.456	1.456	1.337	1.337	1.337	1.276	1.276	1.276	1.896	1.896	1.896
No. Of Flow Tubes	2330	2217	2166	1051	1004	991	1481	1414	1385	2137	2023	1988	943	894	871
Max Core Velocity ft/sec	15	15	15	65	65	65	40	40	40	25	25	25	> 60	> 60	> 60
MWt	2975	2830	2765	3320	3170	3130	3085	2945	2885	3010	2850	2800	3410	3230	3150
<hr/>															
<u>Costs, Mill/kwh</u>															
Reactor Capital	0.896	0.857	0.839	0.462	0.433	0.429	0.606	0.577	0.567	0.836	0.792	0.782	0.412	0.397	0.388
System Operation & Capital	-0.040	-0.030	0.011	0.325	0.337	0.329	0.111	0.146	0.113	0.033	0.045	0.049	0.427	0.437	0.486
D ₂ O	0.400	0.380	0.380	0.192	0.183	0.182	0.264	0.253	0.248	0.373	0.355	0.348	0.174	0.165	0.162
Organic Makeup	0.276	0.263	0.257	0.400	0.384	0.378	0.370	0.355	0.348	0.358	0.341	0.334	0.233	0.221	0.215
Fuel Cycle \$100/kg	1.180	1.120	1.090	1.457	1.390	1.370	1.270	1.124	1.184	1.188	1.124	1.104	1.820	1.720	1.680
Total \$100/kg	2.712	2.590	2.577	2.836	2.727	2.688	2.621	2.455	2.460	2.788	2.657	2.617	3.066	2.940	2.931
Fuel Cycle \$50/kg	0.840	0.800	0.780	---	---	---	---	---	---	---	---	---	---	---	---
Total \$50/kg	2.493	2.388	2.383	---	---	---	---	---	---	---	---	---	---	---	---

TABLE 0910-19 (Cont.)

FUEL CYCLE COST DATA

	K	K	K	L	L	L	M	M	M	N	N	N	N
Steam Pressure (psig)	600	900	1200	600	900	1200	600	900	1200	1000*	600	900	1200
No. Passes	2	2	2	2	2	2	2	2	2	2	2	2	2
Reactor Exit Temp (°F)	750	750	750	750	750	750	750	750	750	750	750	750	750
Organic V ^f	0.340	0.340	0.340	0.340	0.340	0.340	0.430	0.430	0.430	0.430	0.430	0.430	0.430
Reactor ΔT (°F)	157	157	157	167	167	167	200	200	200	233	233	233	233
Reactor ΔP (psi)	540	540	540	260	260	260	424	424	424	170	170	170	170
Thickness Of Flow Tube (in.)	0.173	0.173	0.173	0.100	0.100	0.100	0.145	0.145	0.145	0.063	0.063	0.063	0.063
Thickness Of Clad. (in.)	0.025	0.025	0.025	0.019	0.019	0.019	0.023	0.023	0.023	0.017	0.017	0.017	0.017
Equilibrium Fuel Enrichment (Wt % U ²³⁵)	1.414	1.414	1.414	1.310	1.310	1.310	1.443	1.443	1.443	1.340	1.340	1.340	1.340
No. Of Flow Tubes	1291	1220	1187	1641	1561	1525	716	683	653	1000	1056	998	968
Max Core Velocity ft/sec	54	54	54	39	39	39	55	55	55	33	33	33	33
MWt	3175	3000	2920	3040	2895	2825	3070	2940	2800	2830	2990	2825	2740
<u>Costs, Mills/kwh</u>													
Reactor Capital	0.441	0.512	0.500	0.658	0.630	0.617	0.334	0.323	0.311	0.440	0.454	0.434	0.420
System Operation & Capital	0.154	0.176	0.237	0.020	0.049	0.121	0.053	0.114	0.183	0.011	0.011	0.061	0.086
D ₂ O	0.232	0.220	0.215	0.290	0.277	0.270	0.140	0.130	0.120	0.180	0.190	0.180	0.180
Organic Makeup	0.212	0.200	0.195	0.198	0.189	0.184	0.232	0.223	0.212	0.201	0.212	0.201	0.195
Fuel Cycle \$100/kg	1.350	1.280	1.240	1.230	1.160	1.140	1.330	1.270	1.210	1.190	1.230	1.160	1.130
Total \$100/kg	2.389	2.388	2.387	2.396	2.305	2.332	2.089	2.060	2.036	2.000	2.075	2.036	2.011
Fuel Cycle \$50/kg	---	---	---	---	---	---	0.980	0.940	0.890	---	0.890	0.840	0.810
Total \$50/kg	---	---	---	---	---	---	1.820	1.810	1.795	---	1.826	1.806	1.779

* (Once-Through Steam Generator)

TABLE 0910-19 (Cont.)

FUEL CYCLE COST DATA

	O	O	O	P	P	P	Q	Q	Q	R	R	R	S	S	S
Steam Pressure (psig)	600	900	1200	600	900	1200	600	900	1200	600	900	1200	600	900	1200
No. Passes	2	2	2	2	2	2	2	2	2	2	2	2	1	1	1
Reactor Exit Temp (°F)	750	750	750	780	780	780	780	780	780	780	780	780	725	725	725
Organic V	0.430	0.430	0.430	0.430	0.430	0.430	0.430	0.430	0.430	0.430	0.430	0.430	0.430	0.430	0.430
Reactor ΔT (°F)	266	266	266	138	138	138	153	153	153	168	168	168	130	130	130
Reactor ΔP (psi)	33	33	33	620	620	620	215	215	215	118	118	118	77	77	77
Thickness Of Flow Tube (in.)	0.040	0.040	0.040	0.230	0.230	0.230	0.092	0.092	0.092	0.058	0.058	0.058	0.040	0.040	0.040
Thickness Of Clad. (in.)	0.015	0.015	0.015	0.028	0.028	0.028	0.019	0.019	0.019	0.016	0.016	0.016	0.015	0.015	0.015
Equilibrium Fuel Enrichment (Wt % U235)	1.283	1.283	1.283	1.511	1.511	1.511	1.359	1.359	1.359	1.305	1.305	1.305	---	---	---
No. Of Flow Tubes	2142	2033	1993	856	816	792	1312	1255	1227	1652	1585	1554	1110	1057	1028
Max Core Velocity ft/sec	15	15	15	69	69	69	38	38	38	27	27	27	30	30	30
MWt	2935	2785	2730	3210	3060	2970	3005	2875	2810	2950	2830	2775	3025	2890	2810
<hr/>															
Costs, Mills/kwh															
Reactor Capital	0.831	0.794	0.780	0.386	0.372	0.362	0.548	0.527	0.519	0.670	0.643	0.629	0.470	0.450	0.440
System Operation & Capital	0.007	0.043	0.057	0.243	0.243	0.277	0.011	0.022	0.075	-.046	-.025	0.037	-.040	0.014	0.071
D ₂ O	0.360	0.360	0.370	0.160	0.151	0.148	0.235	0.225	0.222	0.292	0.280	0.276	0.200	0.190	0.190
Organic Makeup	0.194	0.185	0.175	0.337	0.323	0.314	0.308	0.295	0.290	0.292	0.282	0.277	0.250	0.240	0.230
Fuel Cycle \$100/kg	1.170	1.110	1.090	1.436	1.371	1.331	1.250	1.190	1.168	1.192	1.140	1.116	1.220	1.170	1.140
Total \$100/kg	2.562	2.492	2.472	2.562	2.460	2.432	2.352	2.259	2.274	2.400	2.320	2.335	2.100	2.060	2.070
Fuel Cycle \$50/kg	0.830	0.790	0.660	---	---	---	---	---	---	---	---	---	---	---	---
Total \$50/kg	2.336	2.285	2.155	---	---	---	---	---	---	---	---	---	---	---	---

(30 ft/sec) would always result in minimum cost power. In all of the cases, it did develop that this was so.

Figures 0910-15 and 0910-16 represent 30 ft/sec cross-plots of data. The spread shown in these curves represents cost differences associated with attainable steam pressures. The upper curve represents 600 psig steam condition costs. The lower curve represents costs for the estimates of the highest attainable steam pressure for that condition. The maximum steam pressure considered was 1200 psi. It is well to keep in mind that various trends apparent from these cases can change when other design parameters are considered, because of the strong inter-relationships of many design parameters. For example, the influence of the organic volume fraction shown in Figure 0910-15 may change when fins or spiral spacers are considered. Different organic make-up costs will certainly influence cost differentials associated with changes in reactor exit temperature and T .

The lowest reactor exit temperature considered for the one-pass and two-pass concepts is 725 and 750°F, respectively. Although these lower limits were chosen due to reactor inlet temperature limitations as they effect the attainable steam pressure, both concepts can probably be designed to run at some lower exit temperature with conventional recirculating steam generators. The use of once-through steam generators can also permit lower reactor exit temperature or increase attainable steam pressures, although the improvement probably will be more significant in the higher ΔT cases.

The following is a tabulation of observations that have been made from these studies:

- a) The two-pass concepts are associated with higher ΔT 's than the one-pass concept.

b) Costs with the two-pass concepts are as low or lower than corresponding one-pass design costs.

c) The relative one-pass/two-pass picture for a given exit temperature and organic volume fraction is not altered by comparisons at a different exit temperature or organic volume fraction.

d) Higher exit temperatures result in higher power costs due primarily to increases in the organic make-up costs.

7. Guidelines and Assumptions

The following range of reactor variables was used for this study.

Reactor Test Temperature	725-780°F
Reactor ΔT	100-300°F
Estimated Reactor ΔP Range	100-400 psi
Organic Fraction	0.34-0.43
Fuel Element	Unfinned 37-rod (31 large and 6 small)
Fuel Area	6 in. ²

Assessment of the concepts was on the basis of total power cost differentials. These were made up of reactor capital costs, equilibrium fuel cycle costs, system capital and operating cost differentials, D₂O costs, and organic make-up costs. The influence of reactor temperatures and pressures on cladding and flow tube thicknesses was considered in determining enrichment and fuel cycle costs.

The following summarizes the guidelines used in the cost estimates of the one- and two-pass reactor concepts. No detailed information concerning the ex-reactor portion of the plant is included, as this part of the evaluation is being performed by Atomics International.

8. General Guidelines

The reactors considered for these evaluations are heavy water moderated organic cooled, and use vertical pressure tubes with a calandria design. All reactors use on-line, bi-directional fuel management and bi-directional coolant flow. Reactor life is considered to be 30 years. Other assumptions of a general nature are tabulated below.

Reactor Size	Sufficient to provide plant output of 1000 Mwe (net)
Plant Capacity Factor	80%
Active Core Length	17.3 ft
Core Diameter	Variable
Lattice Pitch	Square
Radial and Axial Reflector Thickness	1 ft
Reactor Coolant	Santowax OMP with 10% HB, composition $C_{18}H_{14}$
Fuel Cycle	"Throwaway"
k_{ex}^*	1% $\Delta k/k$
Calandria Tube Material	Zr-4, 0.040-in-thick
Flow Tube Material	SAP-895, Variable thickness
Pigtails	Carbon Steel, 3.6-in.ID, two per pressure tube

All design calculations were performed for full power steady state operation conditions. No assessments of transient behavior, reactor stability, or safeguard considerations were made.

*Provided for limited power operation maneuverability.

9. Reactor Design Guidelines

- a) For considerations of the two-pass concept, the first pass was assumed to be in the central region of the core and occupying 60% of the reactor volume.
- b) Flow orificing was utilized as required.
- c) Fission power was taken 7% higher than thermal power.
- d) Maximum fuel element cladding outer surface temperature of the fuel rods was 850°F.
- e) A 4-foot-long shield plug was assumed to exist at each end of the pressure tubes.
- f) No credit was taken for the gas pressure between the flow tube and calandria in establishing flow tube thicknesses.
- g) Flow maldistribution between flow tubes was represented by a factor of 1.05.
- h) The operating pressure in the reactor discharge header was 65 psia.
- i) The core nominal peaking factors were:

<u>Radial</u>	basic	1.35
	local	1.06
<u>Axial</u>	basic	1.60
	local	1.04

Additional factors relating to geometry variation and flow redistribution are handled within the code.

- j) UO_2 density was 10 gm/cm³.
- k) Moderator composition, D/(D+H), was 0.9975.
- l) Moderator temperature was 190°F

- m) Discharge fuel exposure was 20,000 Mwd/MTU.
- n) D_2O/UO_2 area ratio was 11.92.
- o) The physics calculations employed a zero-dimensional calculation for the non-thermal energy groups and a one-dimensional space-energy calculation for the thermal energy group. Lifetime calculations were based on a point depletion model and the beginning-of-life spectrum. The interpretation of these calculations utilizes a reactor model which assumes a uniform distribution of fuel of all exposures (0 to discharge value) throughout the active core volume.

10. Costing

a. General

The power cost components that were examined in each generating plant case were:

- 1) Total Plant Cost
- 2) Equilibrium Fuel Cycle Costs
- 3) Heavy Water Inventory Charges and Losses
- 4) Organic Make-up Costs
- 5) All Variable Operating Costs

The power conversion plant capital costs were supplied as differentials, with the reference point being a case where the reactor ΔT was 200°F, reactor ΔP was 100 psi, and the steam pressure was 900 ps'. Other cost components are essentially on an absolute basis. The division of costing between AI and CE was taken at the outside of the biological shield. Carrying charges on new fuel in storage were assumed to be sufficiently small and invariant, in the cases considered, to be ignored. This assumption was checked using private financing carry charge rates, and it was found that in the reactor cases considered, the power cost extremes associated with a stored 90-day fuel supply

were 0.06 and 0.03 mills per kwh (net electric). Operation and maintenance charges for the reactor were assumed to be sufficiently invariant in the generating plant cases examined that they could be ignored in relative comparisons. Nuclear liability insurance costs were ignored, assuming that their contribution to power costs would be small and relatively invariant in the cases considered. The Guide to Nuclear Power Cost Evaluation* indicates that the maximum amount of such insurance would amount to 0.052 mills/kwh (net electric), and the minimum amount of such insurance (corresponding to the lowest powered reactor) would be 0.049 mills/kwh (net electric).

The carrying charges that were applied to the reactor and power conversion plant direct capital investment, and the cost of heavy water, are 12 and 10%, respectively. Indirect construction costs were taken to be 37%. This was based on information developed in other studies.**

In the following paragraphs the detailed methods used to obtain the various cost components are presented. All calculations were performed using private financing carrying charges and an 0.8 plant factor.

b. Reactor Capital Investment

The capital investment charges were assumed to relate to the following categories:

- 1) Flow Tubes
- 2) Calandria Tubes
- 3) Calandria Tank

* Guide to Nuclear Power Cost Evaluation, Vol. 1, Reference Data and Standards as amended by Supplement 1, Kaiser Engineers, 1962, pp 15, Section 110.

** J. W. Wade, "A Computer Program for Economic Studies of Heavy Water Power Reactors," DP 707, May 1962, pp 105.

- 4) Control System
- 5) Axial Shields
- 6) Pigtails, Headers, and Downcomers
- 7) Three Refueling Machines
- 8) Biological Shield

It was recognized that there would be further contributions to these components, but it was felt that these components provided reasonable numbers for relative comparisons. Costs of these components varied in the one/two-pass considerations, except for the pigtails, headers, downcomers and the refueling machines. The pigtails, headers and downcomers were assumed to be a small enough contribution to reactor costs that a constant nominal value could be used for this preliminary survey. These will be variables for later studies.

The weight of the calandria tank as a function of the number of pressure tubes was determined, based on the active core diameter. Heavy water reflectors, 1-ft-thick, were then added axially and radially.

The costs of the axial shields were determined in a similar manner. The volume of the biological shield was calculated using two right circular cylinders. The one was 6-ft-thick and without ends. The other, 4-ft-thick and with a 6-ft-thick cap on the top. The dimensions used for the biological shield were determined from work done in CEND-175*, with adjustments made for reactor size. No adjustments were made for changes in shielding thickness.

c. Fuel Cycle Costs

The "throwaway" fuel cycle model that was used is tabulated below:

* CEND-175, Organic Cooled, Heavy Water Moderated Reactor Power Plant Conceptual Design and Evaluation, March 1963.

<u>Cost Component</u>	<u>Value Used</u>
U ₃ O ₈	\$8/lb*
Fabrication	\$100/kg and \$50/kg
New Fuel Shipping	\$1.50/kg
Irradiated Fuel Storage	\$3/kg

d. Heavy Water Costs

The costs associated with heavy water were developed using the following data:

<u>Parameter</u>	<u>Value Used</u>
Thickness of Axial and Radial Reflectors	1 ft
D ₂ O Area per Cell	71.99 in ²
D ₂ O Costs	\$14/lb ²
D ₂ O Annual Losses	1% of Inventory

The heavy water inventory in the calandria tank was determined by determining the active diameter and adding on the reflectors. The heavy water inventory in the inlet and outlet pipes to the calandria tank was not considered, for it was felt that variations in this inventory would be small.

e. Organic Make-up Costs

Organic make-up costs were determined by the following formula:

$$\text{Annual Make-up Cost (\$)} = \frac{1050 \text{ Mw} f Y \text{ VF}}{0.34} + P$$

* J. W. Wade, "A Computer Program for Economic Studies of Heavy Water Power Reactors," DP 707, May 1962, pp 105.

where:

Mwf = total reactor fission power

Y = radiolysis reaction rate, a function of average coolant temperature and fission energy deposited in the coolant

VF = organic volume fraction

P = annual cost of make-up due to pyrolysis, a function of reactor coolant temperature.

As a first order approximation, the cost of make-up due to radiolysis was adjusted proportionally to the organic volume fraction in the cell. This assumption will receive further consideration during the next report period.

11. Present Reference Design Characteristics

During this report period, several reactor concepts were established for possible use as a "point of departure" for initial design and development efforts. These concepts were based on bi-directional, on-line refueling with channel coolant flow in the direction of fuel movement. Both one and two coolant pass designs were considered. Table 0910-20 presents the reactor design characteristics of the reference concept evolved during this period and presently in use as a basis for design and development efforts. The parameters listed must of necessity be considered tentative since future design efforts may indicate the desirability of certain changes. A cross-sectional outline of this type of fuel element in Figure 0910-1 shows the arrangement of the 31 large and 6 small fuel rods. Fuel rods of two different diameters are used to reduce flow maldistribution normally associated with rod bundles in circular pressure tubes. The coolant flow paths through the reactor are shown schematically in Figure 0910-18. Coolant from the steam generator enters the inlet header

REACTOR DESIGN CHARACTERISTICS

GENERAL

NUMBER OF COOLANT PASSES	2
PLANT OUTPUT (MWe NET)	1000
REACTOR THERMAL POWER (MW)	2850
COOLANT FLOW RATE (lb/hr)	92×10^6
REACTOR OUTLET TEMP (°F)	750
REACTOR INLET TEMP (°F)	550
COOLANT	SANTOWAX OMP, 10% HB
STEAM PRESSURE (psig)	900

FUEL ELEMENT

TYPE OF FUEL	UO ₂
NO. FUEL RODS PER BUNDLE	
LARGE	31 (0.521 in. DIA)
SMALL	6 (0.323 in. DIA)
CLADDING MATERIAL	SAP-895
ACTIVE CORE LENGTH (ft)	17.3

PRESSURE/CALANDRIA TUBES

NUMBER	1190
--------	------

MATERIAL

PRESSURE TUBE	SAP-895
CALANDRIA TUBE	Zr-2

NUCLEAR

D ₂ O/UO ₂ VOLUME	11.9
ORGANIC VOLUME FRACTION IN PRESSURE TUBE	0.45
AVG MODERATOR TEMP (°F)	190
RADIAL REFLECTOR THICKNESS (ft)	1.0
FUEL BURNUP (MWD/TONNE U)	20,000

THERMAL-HYDRAULIC

MAX FUEL CLAD SURFACE TEMP (°F)	850
HEAT FLUX (BTU/hr - ft ²)	
AVERAGE	97,000
MAXIMUM *	312,000
MINIMUM DNB RATIO	2.5
PRESSURE DROP (psi)	
HEADER TO HEADER	135
FIRST PASS FUEL	31
SECOND PASS FUEL	59
MAXIMUM COOLANT VELOCITY (ft/sec)	30
FLOW AREA RATIO (1 st /2 nd PASS)	60/40

* INCLUDES ALLOWANCE FOR FRESH FUEL INSERTION

connecting pipe which leads to the upper and lower inlet headers. From these headers the coolant goes through jumper tubes to the pressure tubes of the first pass. In half of these tubes flow is upward, in the other half it is downward. From the first-pass pressure tubes, coolant exits through jumper tubes to upper and lower interpass headers. Flow then proceeds to the second pass through the jumpers, half of the flow going upward, half downward. Coolant exits from the second pass sequentially into jumper tubes, the upper and lower exit headers, and to the reactor discharge piping to the pumps.

Power peaking factors shown in Table 0910-20 are preliminary, representing best estimates at this time. The shapes of the axial and radial power distributions are shown in Figures 0910-19 and 0910-20, respectively. The general "roof topping" effect shown in Figure 0910-19 is due to bi-directional fuel management. Maximum power is obtained from this arrangement with enriched reactors, when the coolant flow is in the same direction as fuel movement, placing the element with the highest heat flux in the region of lower temperature coolant. The power distribution shown represents that which occurs after fresh fuel has just been added to the fuel channel. For preliminary studies, it was assumed that the fuel in the core was divided into five pieces. This is the reason for the five discontinuities shown in the axial power distribution.

A summary of hot channel coolant and surface temperatures is shown in Figure 0910-21. The hot channel represents the hottest flow channel within the fuel element having the highest surface temperature. At this time, no credit has been taken for coolant mixing within the fuel element, to be conservative. Design studies are presently based on a fuel cladding maximum outer surface temperature of 850°F, neglecting fouling.

Selection of primary reactor design parameters was based on the considerations listed in the ensuing paragraphs.

a. Reactor ΔT and Exit Temperature

Studies¹ have shown high ΔT (180-250°F) systems to be desirable. Plant arrangement problems are reduced because of smaller piping and valve requirements associated with high ΔT 's (low flow rates). These arrangement problems have been identified² as a limiting factor in determining maximum sizes of the HWO CR concept.

One- and two-pass studies have shown the high ΔT systems to be economically somewhat better than the low ΔT systems. It is expected that some additional economic gains can be shown, since the high ΔT two-pass reactors in the studies were not, under the assumptions made, thermally and hydraulically optimized designs, and the low ΔT one-pass reactors were. Further studies on a system with a 750°F outlet temperature and a 200°F ΔT have substantiated this conclusion, and the present reference two-pass design differential cost is shown for comparative purpose superimposed on Figure 0910-19. While studies have shown economic incentives for increasing reactor ΔT above 200°F, the potential cost savings must be balanced against the fact that there is less choice in the selection of steam plant pressures and equipment for ΔT 's much over 200°F.

Reactor outlet temperatures above 750°F do not appear to be warranted at this time, because of increased power costs. Lower outlet temperatures would reduce steam plant pressures; however, organic costs and damage rates are subject to continual study. As a result of the design efforts in the one-

¹CEND-175, "Organic Cooled Heavy Water Moderated Reactor Power Plant Conceptual Design and Evaluation," Combustion Engineering (March 1963).

²DP-899, "Large Heavy Water Moderated Power Reactors; An Engineering Feasibility Study," DuPont Co., SRL (November 1964).

and two-pass reactor comparison, and for the above reasons, a reactor ΔT of 100°F and an outlet temperature of 750°F were chosen as initial bases for design.

2. Reactor Header Arrangements

The one-pass reactor requires two upper and two lower coolant headers. The two-pass reactor requires three upper and three lower headers. Since the flow rate in the one-pass reactor is about twice that of the two-pass reactor, its headers are greater in size. The number of connecting pigtails is essentially the same for both concepts. Running the large piping of the one-pass reactor through the shielding presents more problems than with the smaller piping associated with the two-pass reactor. In summary, therefore, mechanical design problems in this area do not seem to pose more of a problem for a two-pass reactor than for a one-pass design.

C. MECHANICAL DESIGN

As the first stage in the conceptual design of the reactor complex (calandria, end shields, thermal and biological shields, fueling machines and organic coolant collection and distribution system within the biological shields), it was considered desirable to investigate in a preliminary way the more promising types of component construction, materials and equipment layout, together with suitable methods of fabrication, assembly and erection. Following this, a single concept would be selected for more detailed study leading to a reasonably accurate cost estimate for a typical reactor complex which could be used as a base for the comparison of alternative concepts. Careful attention was to be paid to the integration of the complex into the complete reactor system.

This effort is described by the two following sections; viz., (1) the present reference design discussing the calandria and end shield in a vertical arrangement, and (2) alternate studies, which during this report period, dealt with a crossed tube lattice configuration.

1. Calandria and End Shield

The calandria vessel will be of the order of 35 ft in diameter and 20 ft high, with an operating pressure approximately atmospheric at the top. Solid austenitic stainless steel was selected as the vessel material because of compatibility with the D_2O moderator, ease of fabrication, and freedom from radiation embrittlement. Stainless clad carbon steel was considered, but rejected because of possible radiation embrittlement and the cost of cladding the tube sheets. Zircaloy calandria tubes were selected in preference to aluminum, because of reported galvanic corrosion problems with the latter in a D_2O -stainless steel system. The calandria would

weigh about 85 tons with tubes, and could be completely shop fabricated, but shipping could then be only by water. Since water access to the site cannot be guaranteed, the initial detail design of the vessel for costing purposes will be based on field assembly of partially prefabricated sections.

The calandria will be supported by the bottom end shield with provision for accommodating differential expansion. Methods of distributing the moderator coolant within the calandria, of venting radiolytic gases from the calandria, and of providing cooling of the upper portions of the vessel in the event of a temporary fall in the moderator level are being investigated.

The upper and lower end shields will be of the order of 37 ft in diameter and 5 ft high. Total by weight will be in the order of 480 tons each. These vessels, again, could be shop fabricated in one piece, but could then be shipped only by water. Shop fabrication is nevertheless preferable, because of the large number of accurately located and bored tubes required, and it was decided to design the shields in completely shop fabricated sub-sections that could be shipped by rail, since the location of the plant is unknown. Two basic concepts were considered:

- a) Pie-shaped segments
- b) Parallel beam sections

The maximum size of a pie-shaped segment is controlled by the maximum allowable shipping dimensions. For rail shipments, this would result in a large number of segments. The parallel beam concept was chosen, therefore, with each shield consisting of three beam segments spanning the shielding opening. The beam sections are independent, with mechanical interlocking where necessary. For convenience in fabrication, the sections are flat-sided, forming an octagonal shield when assembled.

The use of steel balls, castings, or plates as shielding materials, together with water within the sections, was investigated. A concept was developed with steel balls which would allow field installation of the ball charge through manways at the ends of the sections, thus reducing the shipping weight. The sections which would be tilted about their ends while being filled are approximately 12 ft 4 in. wide, and have longitudinal and transverse internal stiffeners to limit twisting and deflection due to their own weight and external loading (the weight of the fuel channels for the upper shield and the weight of the calandria and moderator for the lower shield). Preliminary calculations indicate a maximum fast (> 1 Mev) integrated neutron flux into the shields of 3×10^{18} nvt. At this time, it would appear that ordinary carbon steel (eg. SA 212) will be an acceptable material under these conditions. Figure 0910-22 shows the details of the lower end shield assembly, which is typical. A cost estimate of the shields has been started.

Details for the support of the tank section and their alignment with the calandria are not shown on the drawing at this time. Several design concepts for these requirements are presently under study, together with the necessary handling and lifting attachments for shipment and field installation operations.

One of the supporting and alignment concepts is a peripheral skirt by which the tanks would be suspended from the supporting structure. This method would provide for concentric radial expansion of the shield tanks, while at the same time providing uniform resistance to lateral shock loadings. Another concept is simply to support the tank sections at the periphery; one end of the sections would be constrained, and the opposite end set on a sliding or roller pedestal for minimum expansion resistance, similar to supporting

arrangements for long bridge spans. Expansion along the reactor centerline is effected by a slide and groove arrangement which also provides restraint against lateral shock loads. Lateral expansion is guided but unrestrained. With both upper and lower end shields so supported and initially concentrically aligned, relative concentricity is maintained, providing shield temperatures are nearly equal. In both these concepts the calandria is supported on, and concentrically aligned to, the lower shield, thereby effecting concentric relative expansion of these three major components.

During the report period, work on reactor layout was concerned mainly with methods of distributing the organic coolant to, and collecting it from, the pressure tube extensions. Scoping studies were made for a reactor with bi-directional coolant flow and fueling having a 9.5-in lattice pitch. Maximum coolant velocity was taken to be 30 ft/sec, and both one and two-pass arrangements were considered.

A brief investigation of manifolding the pressure tube extensions into headers was made. The problems which became apparent immediately are discussed in the following paragraphs.

a. Differential Thermal Expansion

A differential thermal expansion exists between the header and the upper shield tank which supports the pressure tubes. For a 33 ft 4 in. core diameter, this differential would be approximately 0.84 in. between a center pressure tube and a peripheral tube. This would develop bending stresses in the pressure tube extensions and cause a large deflection (approximately 0.5 in. for a freely expanding header) in the extension which would present a severe alignment problem in the fuel channel. The use of expansion bellows to accommodate the differential expansion would present a difficult problem, due to

the restricted space envelope available and the necessity to provide for the end thrusts developed by the pressure in the bellows.

b. Pressure Tube Extension Manifolding

The manifolding of the pressure tube extensions would require headers to provide sufficient flow area and which would be able to pass diagonally between the rows of pressure tube extensions (because of the bi-directional flow requirement), in order to reach the collection headers without interfering with the refueling equipment. The largest round header possible would be approximately 6 in. OD, and could not take the flow from more than four fuel channels. The use of rectangular ducts would require special fittings and heavier wall thicknesses.

c. Control Column Insulation

The type of control columns presently under consideration run parallel to the pressure tubes. Space must be provided between the headers, in order to permit passage of these columns. Since the pressure tube extensions will operate in the 500 to 750°F range, and the control column will operate below 200°F, space must be provided for adequate insulation.

The problems become more severe when two-pass coolant flow is considered. The use of coolant collection and distribution plenums was also given brief consideration. This arrangement has a thermal expansion problem similar to that encountered in the header design discussed previously, with the additional complication that inlet pressure tube extensions for flow in one direction must pass through the outlet plenum for flow in the opposite direction, and vice versa. Such a scheme would appear more suited to a uni-directional flow reactor, provided that the differential expansion problem could be solved, perhaps by supporting the fuel channels from the upper plenum instead of the end shield.

Attention was then turned to arrangements using individual pigtails to connect the pressure tube extensions to the collection headers outside of the pressure extension area. Numerous pigtail arrangements were studied, in order to develop the most advantageous design on the basis of accessibility, compactness, and pigtail length. The arrangement of connecting pigtails to pressure tube extensions, which was selected for further examination, is shown in Figure 0910-23. It has the following features:

- 1) It provides for a quadrant arrangement of the pigtails, which permits a uniform distribution of the flow around the outer circumference of the reactor and, consequently, the use of multiple small headers.
- 2) It provides unobstructed vertical access to each of the couplings connecting the pigtails to the pressure tube extensions.
- 3) It provides for each alternate vertical bank of pigtails to consist of pipes carrying flow in one direction only.

A preliminary layout of the pigtails and headers using this basic pigtail arrangement was developed for a one-pass-flow reactor. This design is shown in Figure 0910-24. Typical characteristics used in defining this arrangement were the following:

- 1) 1388 tubes arranged in quadrant symmetry
- 2) Tube size, 3 in. schedule 10 pipe
- 3) Grid spacing, $9\frac{1}{2}$ in.
- 4) Total coolant flow, 218×10^6 lb/hr
- 5) Header size, 36 in. OD

The headers were assumed to be fixed at the outer biological shield, so that thrusts from external piping would be reacted by the header supports. The pigtails were designed to absorb the differential expansion between the outermost and innermost pressure tube extensions and the headers (approximately 0.84 in.), and the differential vertical expansion between the extensions and the headers (up to 0.75 in. in the upper pigtails and 3.75 in. in the lower pigtails for a SAP pressure tube). The pigtail configurations were designed to reduce thermal expansion loads imposed on the pressure tube extensions by the pigtails to 100 lb or less in the horizontal plane.

A preliminary layout of the pigtails and headers for a two-pass flow reactor using the same basic pigtail arrangement is shown in Figure 0910-25. The same total flow, grid spacing, pigtail size, and total number of pressure tubes were

used as in the one-pass design. The first pass consisted of 832 pressure tubes in the central region of the reactor core, the boundary of which is indicated by a line in Figure 0910-23; the second pass consisted of 556 tubes in the outer region of the core. The design is similar to the one-pass design, except for the addition of the inter-pass mixing headers at the top and bottom of the reactor. The inter-pass headers were located at the same level as the outlet headers, but closer to the reactor in order to simplify the pigtail arrangement. All three headers were assumed to be supported and anchored on the biological shield. Sufficient flexibility was provided in the pigtails, to limit the individual loads on the pressure tube extensions to 100 lb maximum, as in the one-pass design described previously.

For both one- and two-pass reactor designs, the headers would be manifolded in the external coolant system to suit the number of coolant loops and to provide cross connection between the loops.

Work was started to define the space requirements for the fueling machines, including space for routine maintenance of the fueling machine heads, for new- and spent-fuel handling, and for spent fuel storage.

2. Crossed-Lattice HWOCR Concept

a. Introduction

The crossed tube lattice concept represents an attempt to solve the principal mechanical problem of large size pressure tube reactors in which the fuel channel assemblies are closely spaced tubes, disposed in a parallel array. There is a severe space limitation for bringing coolant connections to the fuel channel tubes. This restriction on space is made still more severe if on-power refueling is specified. On-power refueling probably requires that the ends of the tubes lie on one plane and remain unencumbered by piping connections. As a result, coolant jumper tube (pigtail) connections to headers are forced into the narrow space between fuel channel tubes. Such an arrangement of connections is feasible, but only if the jumper tubes are banked. For any scheme so far proposed with banked jumpers,

removing one fuel channel tube for repair is possible only after the removal of many other jumper tube connections. The crossed lattice concept eliminates or at least greatly reduces this difficulty.

In the crossed lattice design, fuel channels lying in two adjacent planes, separated by the lattice spacing, cross the reactor in one direction while the tubes in the next two planes cross the moderator in another direction which may lie 90° from the first. In this way, the ends of the tubes are spread over four faces of the reactor. On any one face, the space where tubes do not penetrate that face is available for headers and jumper tube connections.

The crossed lattice was invented on January 4, 1957 at General Nuclear Engineering Corporation, which subsequently became a wholly-owned subsidiary of Combustion Engineering, Inc. Patent No. 3190807 on this invention has been issued to Combustion Engineering, Inc.

b. Principal Problems of a Crossed Lattice Design

In its original concept, the crossed lattice design visualized process tubes lying in horizontal planes. While it is possible to conceive a crossed lattice design with one set of process tubes running vertically and another set horizontally, it does not appear probable that such a design would be attractive. It would require solving the problems of fuel support for the case where the weight of the fuel column is borne in some manner by the column itself, and also the case where the weight of the fuel column is borne by the pressure and calandria tubes. At least two kinds of refueling machines would be required. Hence, this arrangement of the crossed lattice reactor would require solving some of the more difficult problems of both the vertical tube and horizontal tube reactors. For these reasons, the further discussion of the crossed lattice in this report will be limited to process tubes lying in horizontal planes.

The problems to be solved with the horizontal tube crossed lattice design fall into two areas. The first is the amount of material needed in the calandria and pressure tubes in order to support the weight of the fuel and to sustain

whatever stresses are placed otherwise on these members. The second is the accommodation of the change in length due to temperature change. Included in this problem are situations which may result from non-steady-state operation of the reactor. Ideally, the process tube should be anchored in the calandria structure and be free to expand, with very small additional load imposed by stresses in piping connections.

In order to explore the stress and geometrical situation, a single concept of a 1000 Mwe reactor has been chosen and the necessary analyses made. This design is described in subsequent paragraphs. It should be pointed out that no claim is made that this design even approaches an optimum one. It is a design, however, in which the stresses are acceptable for the materials and conditions envisaged. The geometrical arrangement of headers and coolant connections also appears feasible.

Table 0910-21 gives the principal dimensions and conditions of the design concept.

c. Description of the Reactor

For a crossed lattice design, the principal dimensions of the moderator (calandria) tank are established by the lattice spacing and the number of process tubes required. In this design, it was decided arbitrarily to limit the number of process tubes in any one plane to 30. To achieve a total of 1200 tubes, the number of tubes in the vertical bank is 40. Staggered lattices are not employed, in order to leave spaces which are free of obstruction for the full vertical height of the reactor. The calandria tank is of square cross-section, with a width of 24.5 ft and a height of 32.87 ft. It is of stainless steel construction, because of possible radiation damage in areas where process tubes lie near the tank wall. A carbon steel tank has been considered, but there was no readily apparent

TABLE 0910-21

PRINCIPAL DIMENSIONS OF THE 1000 Mwe CONCEPT

ASSUMED LATTICE PITCH (in.)	9.5
ASSUMED TOTAL NUMBER OF PROCESS TUBES	1200
ASSUMED NUMBER OF PROCESS TUBES IN A SINGLE HORIZONTAL BANK	30
REFLECTOR THICKNESS (ft) -- SIDE	0.75
END	3.5
INSIDE WIDTH OF MODERATOR TANK (ft)	24.54
NUMBER OF PROCESS TUBES IN A VERTICAL BANK	40
INSIDE HEIGHT OF MODERATOR TANK (ft)	32.87
ASSUMED THICKNESS OF SHIELD (ft)	5
LENGTH OF EXTENSION OF PROCESS TUBE EXTERIOR TO SHIELD WALL (ft)	5
OVERALL LENGTH OF PROCESS TUBE (ft)	44.54
ID OF PROCESS TUBE (in.)	4.0
PROCESS TUBE MATERIAL	SAP
WALL THICKNESS, PROCESS TUBE (in.)	0.093
THICKNESS OF INSULATING GAP BETWEEN PROCESS AND CALANDRIA TUBES (in.)	0.25
ID OF CALANDRIA TUBE (in.)	4.686
WALL THICKNESS OF CALANDRIA TUBE (in.)	0.080
PROCESS TUBE INLET AND OUTLET CONNECTION	3 in. SCHEDULE 10
WALL THICKNESS OF JUMPER TUBE (in.)	0.120
FLOW PATTERN	TWO PASS
TOTAL WEIGHT OF CONTENTS OF PROCESS TUBE (lb) (UO ₂ FUEL)	800
MAXIMUM CHANGE IN TEMPERATURE OF PROCESS TUBE (°F)	680
MAXIMUM CHANGE IN LENGTH OF PROCESS TUBE (in.)	3.5
MAXIMUM VERTICAL DEFLECTION OF PROCESS TUBE (in.)	0.030

solution for providing a thermal shield without increasing the length of the process tubes. Further design could show the feasibility of a carbon steel moderator tank.

Due to the crossed lattice, the lengths of the calandria and process tubes are determined by the lattice spacing and the number of tubes in a horizontal bank. The side D₂O reflector in this design is only 0.75 ft, while the end reflector is 3.5 ft if the active fuel length is assumed to be 17.25 ft. Presumably, further analysis would lengthen the active fuel column or make other dimensional adjustments to eliminate the excess reflector.

The Zircaloy calandria tubes are placed across the calandria tank and fixed in the tank wall by rolling or, if provided with stainless steel ends, by welding. Since this work is done at room temperature and the operating temperature of the moderator may be 200°F, the Zircaloy will be placed in tension by the expansion of the tank relative to the calandria tube expansion. To substantially eliminate the reaction at the tank wall, it was assumed that sufficient deflection could be built into panels of the tank wall to accommodate the differential expansion. Panels sealed into the tank wall by omega seals are envisaged.

The moderator tank is backed up by a steel-water shield or possibly a concrete shield. In any case, rough estimates indicate that a 5 ft shield thickness should be ample to reduce neutron activation outside the shield to tolerable levels. No attempt has been made here to establish the design of either the calandria tank or shield. The thickness of the shield can only be determined by detailed analysis. Its thickness directly affects the overall length of the process tube.

Insofar as possible, the conditions of flow, temperature, weight of fuel, and other pertinent parameters in the crossed lattice design are the same as in the vertical tube design described in this report.

In order to bring the stresses within acceptable limits, it has been necessary to include a center support for the calandria tubes, and three middle supports for the process tube within the calandria tube. The center support is visualized as a zirconium tube of rectangular cross section which is mounted vertically, and through which the calandria tubes pass. The holes for passage of the calandria tube are oversize and provided with a saddle on which the calandria tube lies. The vertical central support tube is put under tension by suitable jacks at the point where it passes through the upper shield; by so doing, the sagging of the lower calandria tubes due to stressing the support tube, or to thermal expansion of the support tube, is minimized and kept within acceptable levels.

The reactor is envisaged to have two-pass flow with bi-directional refueling. The cold (550°F) inlet coolant is brought to the process tubes occupying the central 18 positions in any bank of tubes in a horizontal plane. These tubes, therefore, form the first pass. The outlets of these tubes are connected to a mixing header from which the fluid enters the second pass. The second pass tubes are the outer six on each side of the central bank of 18 in any horizontal bank. Temperature rise of the coolant is 150°F in the first pass and 50°F in the second pass. Because there are fewer process tubes in the second pass, the coolant velocity is greater in the second pass than in the first. The process tubes are anchored to the reactor shield at the inlet for the first pass and at the outlet end for the second pass. The mixing header which forms the outlet for the first pass and the inlet for the second pass floats on the process tube extensions, and moves away from the face of the reactor shield when the process tubes expand. Incorporating the floating header in the space between banks of process tubes is effective in reducing the stress which the piping connections must endure. On

any one face, the space between the process tubes is occupied alternately by inlet and outlet jumper tubes and the mixing header. This is required, to permit bi-directional refueling and two-pass flow. Since the process tubes are anchored at the inlet of the first pass and at the outlet of the second pass, the connecting jumper tubes at those points experience little stress, because the length of process tube free to expand and create the stress is quite small. At the face, these inlet and outlet jumper tube connections are carried across the face of the reactor and connected to vertical inlet and outlet headers as is shown in Figure 0910-26. While these jumper tubes are arranged in a bank 4 high and 7 deep, each process tube can be disconnected from its jumper without disturbing the connection of the jumpers to other process tubes.

Consideration has been given to using flexible bellows-type tubing to make the coolant connections to the process tubes. From a strictly geometrical and mechanical point of view, such connections can be made with an estimated very long service life. The unknown in the situation is the life of the bellows tubing when subjected to hot, organic fluid. Corrosion data indicate that the life ought to be extensive, but actual testing of bellows tubing for such service has not yet been performed. The conceptual design depicted in this report does not use the bellows tubing connection, since it was found that for the two-pass coolant configuration, pipe connections could be made. The connections shown in Figures 0910-26 and 27 have been analyzed. Small and acceptable forces are exerted on the process tubes.

Table 0910-22 gives further details of the stresses calculated for the configuration shown in Figure 0910-26. Figure 0910-27 has been prepared to assist in visualizing the jumper connections at inlet and outlet.

TABLE 0910-22
CALCULATED STRESSES

	σ_a	σ_h	σ_r	T	$S_{allowable}$	FACTOR OF SAFETY	DESCRIPTION
CALANDRIA TUBE	- 226	1,028*	- 15.2	200 °F	17,000		Hydrostatic Pressure
	3,066						Max Moment At Support
	1,059						Moment Due To Misalignment Of Middle Support And End Supports (0.060 in.)
	+ 3,899						
	- 4,351					3.9	
PROCESS TUBE	1,767	3,534	- 154	750 °F	4,600		Internal Pressure
	2,285						Moment At Middle Support
	491						Misalignment of 0.060 in.
	4,543					1.012	
	- 2,776						

σ_a = AXIAL STRESS (psi)

σ_h = HOOP STRESS (psi)

σ_r = RADIAL STRESS (psi)

+ TENSION

- COMPRESSION

$S_{allowable}$ IS STRESS TO 1% CREEP IN 30 yr.

* STRESS DUE TO EXTERNAL PRESSURE PLUS STRESS DUE TO RADIAL MOMENT FOR AN OUT OF ROUNDNESS OF 0.030 in.

d. Neutron Absorption Penalty Apparently Required for the Crossed Lattice Design

A comparison has been made between the amount of material needed for the process and calandria tubes for the crossed lattice design and for what appears to be acceptable in a vertical tube design. A SAP process tube of acceptable thickness is relatively ineffective in supporting the fuel load; hence, the process tube thickness was increased by only 0.020 in. over the thickness in the vertical design. A thicker calandria tube with a central support can support the fuel load in the horizontal reactor. The actual increase in thickness of the calandria tube wall over the current vertical design is 0.025 in. The central support tube needed to carry the load of all the tubes in a vertical line at the midpoint of the reactor is equivalent to adding another 0.0044 in. to the calandria tube wall thickness. As stated in Section II. B. 1, the net effect of the increases in calandria and process tube walls required in the crossed lattice design is to lower the static system utilization by 0.06, with respect to the current vertical reactor design. The objective is to estimate the difference in fuel utilization between a reactor using a simple vertical lattice and a reactor using a horizontal cross-tube lattice. For mechanical reasons the horizontal (cross-tube) lattice requires somewhat greater process and calandria tube thicknesses. These are the only differences taken into account here. The estimate is made by comparison with the reactor of Table 0910-5 in Section II. B. 1, and uses also Figure 0910-11 of that Section. The parameters employed and the results of the estimate calculation are shown in Table 0910-23.

e. Crossed Lattice Concept for Very Large Reactors

Time has not permitted study in any detail of the crossed lattice design

TABLE 0910-23

NET EFFECT OF WALL THICKNESS CHANGES

	CROSSED-TUBE LATTICE	VERTICAL-TUBE LATTICE	REFERENCE LATTICE ⁽²⁾
A. THICKNESS OF SAP PROCESS TUBE (in.)	0.110	0.090	0.110
B. EFFECTIVE THICKNESS ⁽¹⁾ OF ZIRCONIUM CALANDRIA TUBE (in.)	0.0844	0.055	0.040
C. ADDITIONAL PROCESS TUBE MATERIAL = A - 0.110 (in.)	0	-0.020	
D. PARASITIC ABSORPTION FRACTION DUE TO ADDITIONAL PROCESS TUBE MATERIAL = $\frac{0.0075}{0.039} C$	0	-0.00384	
E. ADDITIONAL CALANDRIA TUBE MATERIAL = B - 0.040 (in.)	0.0444	0.015	
F. PARASITIC ABSORPTION FRACTION DUE TO ADDITIONAL CALANDRIA TUBE MATERIAL = $\frac{0.0075}{0.046} E$	0.00724	0.00244	
G. TOTAL ADDITIONAL PARASITIC ABSORPTION FRACTION RELATIVE TO REFERENCE LATTICE = D + F	0.00724	-0.00140	
H. PARASITIC ABSORPTION FRACTION = 0.0857 + G	0.0929	0.0843	0.0857
J. STATIC SYSTEM UTILIZATION (Mwdt) PER NET GRAM OF FISSILE ISOTOPE BURNED OR DISCARDED ⁽³⁾	1.56	1.62	1.60

1. INCLUDES EFFECT OF ZIRCONIUM USED FOR LATERAL SUPPORT.

2. FROM TABLE 0910-15.

3. FROM H, AND FIGURE 0910-11.

for very large reactors. The 1000 Mwe design is based upon thermal power of approximately 2.4 Mwt per tube. If it is assumed that in the development of a large reactor this could be raised to 4.0 Mwt/tube, then an 8000 Mwt reactor would require 2000 tubes. If, at the same time, the lattice spacing could be reduced from 9.5 in. to 9 in., the very large reactor could be accommodated by a moderator tank having an inside dimension in the horizontal direction of 24.5 ft and a height of 52 ft. This assumes that the maximum of 30 tubes in a horizontal plane is retained. Further study is needed to determine if a height of 52 ft possibly could be acceptable. Increasing the number of tubes in a horizontal plane looks difficult, but may not be out of the question. Flexible tubing connections may be required, to permit an increase. There is another serious objection to increasing the number of tubes in a horizontal plane. The length of the process and calandria tubes also is increased. Increased length makes all the problems of support and expansion more severe and, at the same time, there is doubt that the increased length can be used effectively in the generation of power.

f. Conclusion

The study of a single design of a 1000 Mwe crossed lattice reactor has not revealed any unfeasible areas either in geometry, stress, or parasitic absorption. For very large reactors, further study would be needed to ensure that the height of the reactor does not become unreasonable.

The crossed lattice design has not been examined from the point of view of its arrangement in a reactor building. This may pose some severe layout problems because of the necessity of providing four working places for refueling. The crossed lattice design without staggered tubes provides vertical open channels in the reactor, which can be used conveniently for control elements of either the mechanical or hydraulic type.

III. EVALUATION OF EFFORT TO DATE

The efforts to date have consisted primarily of scoping studies and preliminary designs. As a result of the efforts during this period, a preliminary set of reference characteristics was chosen (see Table 0910-20) and more detailed design initiated. In addition, several design methods of supplying coolant to the various flow tubes were considered and a reference method selected.

Initially, the development of adequate methods for the two-dimensional treatment of heat transfer and hydraulics in a bundle geometry caused some delay in the core design. However, suitable methods have been developed and automated, and evaluation surveys can now be performed quite rapidly.

In general, the progress during the report period has been satisfactory and corresponds to expectation.

IV. NEXT REPORT PERIOD ACTIVITIES

A. REACTOR PHYSICS

The parameter survey will be directed toward evaluating reactivity coefficients; e.g., Doppler, void, power, and coolant temperature coefficient, as a function of the D_2O /fuel ratio, volume fraction of organic, and volume fraction of fuel in the fuel channel. It is anticipated that this type of information will further define the range of variables of interest in the HWOCR concept and provide data for the nuclear safety and stability studies. In addition, the more general goals of the survey will be to extend the range of the pertinent variables to include all values of interest; e.g., the range of cladding thickness will be extended to approximately 0.070 in. to account properly for heavily finned fuel rods. The following variables are presently anticipated to be included in the survey: (1) organic volume fraction, (2) fuel area, (3) D_2O /fuel ratio, (4) cladding thickness, (5) pressure tube thickness, (6) calandria tube thickness, (7) number of pressure tubes, (8) active core length, (9) average coolant

temperature, (10) average fuel temperature, (11) average moderator temperature, (12) number of fuel rods, and (13) the type of fuel.

During the next quarter, it is anticipated that the fuel management schemes of interest will be defined, and both the fuel management scheme and a definitive criterion in fuel utilization will be factored into the survey calculational procedures. The end-cap flux peaking calculations will be analyzed and reported. Shielding investigations will be extended to include radial thermal and biological studies, streaming shields for the coolant channels, and a centering review of the shield designs.

In areas of nuclear safety and xenon stability, literature surveys will continue until the state-of-the-art in both areas can be defined. It is anticipated that one-and possibly two-dimensional calculations will be initiated for further investigation of xenon stability conditions. In particular, these calculations will be directed toward defining the instrumentation required to monitor the effects of the xenon oscillations. Safety investigations will be directed toward defining the safety criteria to be employed in the HWOCR concept.

B. ENGINEERING ANALYSIS

During this period, limited scoping studies have been made of one-and two-pass reactors, and an initial reference concept for reactor design efforts has been established. A major portion of engineering analysis activities for the coming period will be centered around design efforts related to the concept shown in Table 0910-20. Work on development of codes and other analytical design methods will taper off and reactor evaluation efforts will become greater. Work related to the development of the reactor design and cost code for the overall plant cost optimization programs will be initiated. It is expected that the development of such an overall plant optimization code will not be

complete before FY 1967. During the next period, analytical studies of coolant mixing will continue and related experimental test program requirements will be established.

C. MECHANICAL DESIGN

Detailed design of the reactor complex will be continued. This design will be utilized for the ORNL evaluation studies. Detailed cost estimates of the various components will be prepared and the erection sequence will be studied in depth.

FIG 0910-1. 37 ROD FUEL CHANNEL

FIG 0910-2. 19 ROD FUEL CHANNEL

FIG 0910-3. ANNULAR FUEL ASSEMBLY REPRESENTATION EMPLOYED
IN THERMAL CALCULATIONS FOR 37 ROD ASSEMBLIES

FIG 0910-4. FEED ENRICHMENT VERSUS SIMULTANEOUS VARIATION OF PRESSURE AND CLAD TUBE THICKNESSES (SOLID LINE), AND VARIATION OF PRESSURE TUBE THICKNESS (DASHED LINE). THE BROKEN LINE REPRESENTS THE INCREMENTAL CHANGE IN ENRICHMENT WITH CLAD TUBE THICKNESS

FIG 0910-5. CROSS SECTIONAL VIEW OF A 37-ROD FUEL BUNDLE,
FLOW TUBE AND CALANDRIA TUBE

FIG 0910-6. INFINITE MULTIPLICATION FACTOR VERSUS MWD/MTU
USING THERMAL POISON SEARCH-PHYSICS TEST CASE

FIG 0910-7. PHYSICS TEST PROBLEM - LOCAL TO AVERAGE FLUX DISTRIBUTION
(0 - 625 EV)

FIG 0910-8 EFFECT OF FEED ENRICHMENT ON U-235 DISCARD

FIG 0910-9. VARIATION OF U-235 AND FISSILE PLUTONIUM CONTENTS WITH EXPOSURE FOR CASES 162, 225, AND 226

FIG 0910-10. EFFECTS OF DESIGN VARIABLES ON STATIC UTILIZATION

FIG 0910-11 BREAKDOWN OF PARASITIC ABSORPTION FRACTION

FIG 0910-12. TYPICAL SECTION OF 37-ROD FUEL ELEMENT

FIG 0910-13. RELATIVE SUBCHANNEL COOLANT TEMPERATURE RISE AS FUNCTION OF MIXING FLOW

FIG 0910-14. RELATIVE SUBCHANNEL COOLANT TEMPERATURE RISE AS FUNCTION OF MIXING FLOW

FIG 0910-15. INFLUENCE OF INTER-SUBCHANNEL MIXING
 ON MAXIMUM SURFACE TEMPERATURE OF CLAD

FIG 0910-16. COST COMPARISON OF VARIOUS REACTOR PLANT DESIGNS

FIG 0910-17. COST COMPARISON OF VARIOUS REACTOR PLANT DESIGNS

FIG. 0910-18. TYPICAL FLOW DIAGRAM OF BIDIRECTIONAL FLOW TWO-PASS REACTOR

FIG. 0910-19. AXIAL POWER DISTRIBUTION vs FRACTIONAL CORE LENGTH

FIG. 0910-20 RADIAL POWER DISTRIBUTION vs FRACTIONAL CORE RADIUS

FIG 0910-21. HOT CHANNEL CONDITIONS OF THE REFERENCE DESIGN REACTOR

FIG 0910-23. SCHEMATIC ILLUSTRATING
JUMPER TUBE - PRESSURE TUBE FLOW PATHS

FIG 0910-24. PRELIMINARY LAYOUT OF THE JUMPER TUBES FOR A ONE-PASS REACTOR

FIG 0910-25. PRELIMINARY LAYOUT OF THE JUMPER TUBES FOR A TWO-PASS REACTOR

SECTION B-B

FIG 0910-26. PLAN AND ELEVATION VIEWS OF THE HORIZONTAL TUBE, CROSSED LATTICE REACTOR

FIG 0910-27. FLOW TUBE CONNECTION METHODS FOR THE HORIZONTAL TUBE, CROSSED LATTICE REACTOR

HWO CR QUARTERLY REPORT	
TITLE: 1000-Mwe Plant Engineering and Development	SUBTASK NO.: 0950
PROJECT ENGINEER: J. Jacobson	REPORTING PERIOD: Inception through June 1965

I. PROJECT OBJECTIVES

The overall objective of this task is to provide a preliminary reference design (conceptual) and budgetary cost estimate for a 1000-Mwe HWO CR power plant. This overall objective will be attained by means of the following contributory objectives:

- a) Evaluation of alternative fuels, cladding, fuel channels, fuel geometries, primary loop components, process and control systems, and steam conditions and cycles.
- b) Identification of research and development requirements.
- c) Establishment of characteristics, component sizes, and criteria which must be incorporated in the design of the 300-Mwe plant to assure that it will serve as a demonstration plant as well as a prototype for the 1000-Mwe plant.
- d) Establishment of characteristics, component sizes, and criteria that can be extrapolated to an 8300-Mwt desalting plant and assure that the 1000-Mwe plant will serve as a prototype for the 8300-Mwt plant.

In order to achieve these objectives, alternate systems designs will be prepared and evaluated. The areas toward which the Atomics International effort is primarily directed include:

- a) Reactor heat transfer system, including pumps, valves, steam generating equipment, and interconnecting piping.

b) Turbine-generator system, including steam turbine, electrical generator, and all related equipment and facilities.

c) Development of an optimization code for the reactor heat transfer system, including provisions for integrating this code with a similar code for the reactor for overall optimizations.

d) Coolant purification system, including methods and means for controlling high boiler content and coolant purity, and for minimizing makeup cost.

e) D₂O moderator system, including D₂O processing equipment and utilization of heat generated in the moderator.

f) Plant auxiliary systems, including decay heat removal, coolant pressurization and degasification, drain and vent, pressure relief, inert gas, and waste disposal.

g) Plant instrumentation and control requirements, including nuclear, process, and protective.

h) Plant structures and equipment layout, including economic and piping stress studies, and electrical systems.

i) Conceptual design of a loop capable of testing large organic components, including main coolant pumps and valves.

j) A nuclear and heat transfer parameter evaluation of UC fuel.

k) Safety analysis, including containment requirements and studies of potential accidents.

II. PROGRESS DURING REPORT PERIOD

A. MAJOR ACCOMPLISHMENTS

The task effort was initiated with the establishment of design criteria and the delineation of individual activities.

Preliminary process and instrumentation (P&I) diagrams were prepared for the reactor heat transfer system. A procedure was developed for conducting the optimization study for this system, design limitations and ground rules for the study were established, and the determination of cost data for major equipment items was initiated. An optimization computer code was completed for cases involving recirculating boilers. Development of a code for coolant systems utilizing once-through steam generators was initiated. Another code was initiated for the calculation of heat balances using a number of variable inputs and for the calculation of steam cycle data for partial load plant operation. Steam cycle heat balances were calculated for an arrangement employing two half-capacity turbine-generator units; also, plans were formulated to conduct a similar effort for a plant having a single 1000-Mwe (nominal) unit. The improvement of steam cycle heat rate resulting from the utilization of D_2O moderator heat for feedwater heating was calculated; this calculation led to the tentative selection of this method of moderator heat rejection.

Preliminary results of a study performed to assess the capability of industry to provide pumps (not including drivers) for the HWOCR main heat transfer system confirmed that adequate commercial technology and experience exists for their design and manufacture. However, as the study proceeded, it became evident that the pump design would be closely related to driver availability. In some cases, the optimum pump shaft speeds were different from the nominal electric motor shaft speeds; in all cases, the horsepower requirements exceeded those

for which electrical motors are commonly used in central station power plants. For these reasons, the use of turbine-driven horizontal pumps equipped with double-suction impellers (to reduce the required NPSH) is being considered.

A similar study was performed with regard to shaft seals for these pumps. In this respect, too, manufacturers indicated confidence in their capabilities to provide adequate components.

The study of several steam generator concepts of the natural recirculating and once-through types was initiated. Designs were prepared and estimates of the cost of each type are being obtained. The ability of manufacturers to fabricate very large units was investigated.

Three principal methods of organic coolant purification were investigated:

1. Distillation

Distillation removes decomposition products [high boilers (HB)] from the coolant; the rate of distillation affects the concentration of HB in the circulating coolant. In the course of removing HB, distillation also removes particulate material and film-forming compounds. Following investigation of the coolant distillation method, a preliminary conceptual design of such a system for the 1000-Mwe HWOCR plant was prepared. This system was based on the general method and criteria being utilized by the Piqua Nuclear Power Facility (PNPF). In addition, an investigation and preliminary conceptual design of another distillation method was initiated. This method, termed "deep distillation," utilizes a high system through-put rate to maintain impurity concentrations (particulates and film formers) at lower levels, but maintains the desired concentration of HB by further distilling a portion of the bottoms of the high-rate distillation and returning the overhead to the main coolant stream.

2. Adsorption

Adsorption removes film-forming compounds from the coolant. Following investigation of past and current experience in the utilization of this method, a preliminary conceptual design of such a system sized for the 1000-Mwe HWOCR plant was prepared.

3. Hydrocracking

Hydrocracking reclaims a portion of the HB and returns it to the coolant stream, thus reducing coolant makeup requirements. Investigations were made on the general application of this process in chemical and refinery installations, and on the laboratory work performed by Phillips Petroleum Company specifically on organic coolant. Based on the results of these investigations, economic and technical evaluations of the hydrocracking process were made. The former evaluation indicates no economic justification for the process in small (~ 300 Mwe) plants with high ($\sim 40\%$) HB coolant concentrations, but a marginal justification becomes evident and increases to a significant level as the plant size is enlarged and/or the HB concentration is reduced.

Technically, the hydrocracking process was concluded to be generally feasible, but certain aspects involve uncertainties which require resolution.

With respect to moderator system design, the literature on heavy water systems was reviewed; also, knowledge of the design and operating experience of reactor plants employing heavy water was obtained by visits and through reports. An analysis was made of the most economical method of rejecting the heat generated in the moderator. A conceptual design of the moderator system was initiated.

Design criteria for the auxiliary process systems were determined. Studies of past and current auxiliary process systems in related installations, such as PNPP, were made. Where alternate concepts exist, comparative evaluations were

conducted. Preliminary flow diagrams and P & I diagrams were prepared for the selected concepts for a number of the auxiliary organic and steam, cooling water, and waste systems. In addition, preliminary process systems descriptions were started.

Several reactor building equipment layout and piping drawings were made for heat transfer system designs under consideration. Preliminary comparative containment cost studies were completed. A study of the electrical hazards classification requirements of HWOCR plant areas was initiated.

The objectives of the large organic components test loop effort, evaluation of the need for proof-testing large organic equipment, and conceptual design of a suitable test facility were completed.

Core physics parameter surveys for UC fuel were initiated over a broad range of variables of fuel rod diameter, fin height, fuel enrichment, and deuterium-to-uranium ratios. Heat transfer evaluations of these configurations were initiated. A physics test case to compare calculational techniques was completed.

A review of hazards reports and accident analyses for plants with features similar to the HWOCR was made in order to ascertain previously encountered problem areas. As a result of this review and detailed consideration of the HWOCR features, a comprehensive list of accidents requiring safety analysis was prepared.

B. HEAT TRANSFER SYSTEM

1. Organic Coolant System

A major effort during this report period included the development of a procedure for conducting the optimization study, the establishment of the design limitations and ground rules for the study, and the determination of cost

data for the major equipment components of the primary coolant loops. A computer program was developed for performing the numerous calculations required for evaluating the economics associated with varying the design parameters of the heat transfer system; a more detailed discussion of this computer program is presented in section II-B-3.

The major plant parameters chosen for variation were: (a) reactor coolant outlet temperature, (b) reactor coolant temperature rise (reactor ΔT), (c) steam pressure, and (d) steam temperature. The values of these parameters to be studied are:

a)	Reactor outlet temperature ($^{\circ}\text{F}$)	700, 750, and 800
b)	Reactor ΔT ($^{\circ}\text{F}$)	100, 150, 200, and 250
c)	Steam pressure at turbine throttle (psig)	600, 900, and 1200
d)	Steam temperature at turbine throttle ($^{\circ}\text{F}$)	675, 725, and 775

These design condition ranges were selected as being representative of the most probable operating conditions attainable and of the most interest for the initial optimization effort.

On the basis of the reactor design studies being conducted by Combustion Engineering, a range of reactor pressure drops was selected for study also. The pressure drops chosen were 100, 200, 300, and 400 psi.

In conjunction with this project, a similar type of economic study is being conducted for optimizing the steam portion of the plant. This work is discussed in section II-B-2. The results of the steam plant study will be incorporated in the primary heat transfer system optimization work. All parametric studies will be based on a fixed gross plant electrical output of 1088 Mwe. Consequently, the reactor thermal power will be permitted to vary. The optimum plant design

conditions will be determined by combining the costs for the organic coolant system and steam plant with those costs associated with the reactor.

The equipment costs associated with the organic heat transfer loops include the coolant pumps and drives, piping and valves, and steam generators. These components are shown in relation to each other in the typical schematic diagram of one coolant loop presented in Figure 0950-1. Some cost data and outline dimensions have been obtained from equipment manufacturers, and other equipment costs have been developed by the Pricing and Estimating Department. This information was used to develop cost curves which were then incorporated in the computer program.

Figure 0950-2 illustrates how these equipment costs vary with changes in reactor ΔT for a constant reactor outlet temperature, reactor pressure drop, and steam pressure. The costs of the pumps and drives vary directly as the coolant flow rate and total pump head. As the reactor ΔT increases, the total flow rate decreases for a fixed reactor thermal power level. Costs for the piping and valves also vary directly as the coolant flow rate and pump head. The steam generator costs, however, are governed by the reactor inlet temperature, and increase with an increase in reactor ΔT .

The reactor cost information has not been included in the computer program as yet. However, a number of trial runs have been made to determine the optimum operating conditions for the primary heat removal system excluding the effect of the reactor costs.

A typical example of a CRT (cathode ray tube) graph automatically plotted by the computer is shown in Figure 0950-3. This graph shows the relative variation in power generation cost (in mills/kw-hr) with reactor ΔT for a fixed reactor outlet temperature, steam pressure, reactor pressure drop, and annual fixed charge rate.

The preliminary results from the parametric studies conducted to date indicate that the major factor affecting the economics of the primary heat removal system is the cost of the steam generators. The steam pressure available from the reactor plant is determined by the coolant temperature and the approach temperature in the steam generator. Figure 0950-4 shows a typical temperature enthalpy diagram for a reactor steam generator and illustrates how the coolant temperature drop limits the attainable steam pressure. The design of a natural circulation type steam generator is limited by a minimum temperature difference between the reactor inlet temperature and the saturated steam temperature, commonly referred to as the "pinch-point" of the boiler. Designing to the lowest practical pinch-point results in the maximum attainable steam pressure. However, this results in a large log mean-temperature difference in the boiler requiring a large heat transfer surface area. Although the higher steam pressures yield the greatest electrical output, the additional cost for the steam generator surface may not be economically justified.

Figure 0950-5 illustrates how the steam generator surface area varies with the log mean-temperature difference and boiler "pinch-point."

In the design of a once-through steam generator or a natural circulation boiler with a separate economizer, the "pinch-point" of the boiler is greater than the "pinch-point" for the natural circulation boiler without an economizer. This is also illustrated in Figure 0950-4. Consequently, a higher steam pressure is possible with the once-through unit or with the natural circulation unit if an economizer is used.

The computer code is presently programmed to compute the size and cost of the natural circulation type steam generators. Equations for computing the once-through type steam generator have been prepared and will also be programmed for comparison purposes.

The preliminary studies indicate that the steam generator is the limiting size parameter in the primary coolant loop. The final selection of the maximum size and type of steam generator will have a definite influence on the selection of the number of loops.

Figure 0950-6 illustrates the effect of steam pressure on the coolant flow rate, pumping power, and steam generator surface area when the reactor ΔT is varied, while the reactor outlet temperature and pressure drop are held constant. The surface areas of the boiler and super-heater increase as the pressure is increased due to a reduction in the log mean-temperature difference. Conversely, increasing the steam pressure reduces the coolant flow rate and the coolant pumping power. This occurs since the plant efficiency improves with an increase in steam pressure, resulting in a reduction in reactor thermal power level.

The values indicated in Figures 0950-2 through 0950-6 are for specific cases; however, these figures suffice to illustrate typical effects of parameter variation.

In the initial main coolant pump investigations, thirty-four organizations believed to be capable of manufacturing large centrifugal pumps were asked to supply recommendations for pumps designed to meet a preliminary series of operating conditions as listed in the following table:

<u>Pump</u>	<u>Units Required</u>	<u>Design Flow Pump</u>	<u>Hydraulic Horsepower at 750°F & $\Delta P = 350$ psi</u>
A	1	270,000	53,500
B	2	135,000	26,750
C	3	90,000	17,800
D	4	62,500	13,400
E	5	54,000	10,700
F	6	45,000	8,900

Data sought included a predicted performance curve, outline dimensions, estimated weight, approximate cost, delivery time, NPSH required (they were told that 100 ft would be available at design flow), pump shaft diameter at the seal region, and space available for the shaft seal if not furnished with the pump.

Two companies, Newport News Shipbuilding and Dry Dock Co. and Baldwin-Lima-Hamilton, indicated that they have the hydraulic experience to design and manufacture all the pump sizes for which information was sought. Four other companies, Ingersoll-Rand, Worthington, Byron Jackson, and Bingham, indicated capability for all but the two largest pumps. Only two manufacturers, Byron Jackson and Bingham, furnished any information about their pumps; Bingham furnished a predicted performance curve, and Byron Jackson furnished both performance and dimensional data. Preliminary outline dimensions for a range of pump sizes are given in Figure 0950-6A. Byron Jackson also furnished preliminary cost data. None of the manufacturers had ever built pumps of the size and operating temperature required.

For the range of reactor ΔT 's and reactor pressure drops under consideration, pumping power requirements for the primary heat transfer systems range between ~ 20 and 130 Mw of electrical energy, resulting, in some cases, in motor drives in excess of 20,000 hp/pump. Studies of current power plant practice showed that for loads of over 5000 hp, turbines are preferred over electric motors as pump drives. Because of the power required by the HWOCR coolant pumps and the nonavailability of vertical turbines, interest has centered on using turbine-driven horizontal pumps equipped with double-suction impellers. Double-suction impellers reduce the NPSH required by the pumps and permit operation on higher shaft speeds, thereby reducing their sizes and costs.

Inquiries were sent to nine manufacturers of mechanical seals to obtain their recommendations for pump shaft seals. Information sought included instal-

lation space requirements, disassembly space needed, estimated leak rates, effects of varying pump shaft speed upon seal leak rate, amounts of radial and axial movement that can be tolerated by the seal, and approximate delivery and price. Six replies were received and all of these stated that the sealing requirements posed could be met with extrapolations of existing seal designs; however, as in the case of the pump response, the amount of **requested data** that was actually furnished varied.

Since a great amount of effort would be required to evaluate steam generator designs for all the possible combinations of variables, the following reference set of parameters was selected to begin the steam generator study:

Reactor outlet temperature (°F)	750
Reactor inlet temperature (°F)	550
Steam conditions at turbine throttle	
Temperature (°F)	725
Pressure (psig)	900
Net plant electrical output (Mwe)	1000

From these conditions, a total steam generator thermal power requirement of 2950 Mwt was established, with a steam flow rate of 10.6×10^6 lb/hr. This then enabled steam generator heat transfer area requirements and physical sizes to be determined.

Following some initial steam generator sizing effort, steam generator manufacturers were visited. Discussions with these manufacturers confirmed that the types of steam generators which were being considered (i.e., vertical and horizontal natural convection recirculation and once-through) were appropriate for the service conditions. It was also determined that manufacturers are fabricating very large reactor vessel heads and shells, components which are similar to

steam generators from a fabrication viewpoint. Large tubesheets, approximately 24 in. thick by 10 ft diameter, are also being fabricated. These large-size component fabrication capabilities are currently being matched by appropriate inspection and testing techniques. It became apparent, as a result of these visits, that the fabrication of large units would not be a serious limitation in steam generator selection. Table 0950-1 lists some fabricating capabilities which were determined during these visits.

Subsequent to visiting the above manufacturers and utilizing supplementary information furnished by them, effort was returned to analyzing the applicable types with consideration of performance characteristics, size, and cost data. Table 0950-2 lists the quantities and sizes required for the types of steam generator considered.

Briefly discussed below are several aspects of the types of steam generator being considered:

a) In an organic-cooled reactor plant, the steam side of the steam generators operates at a considerably higher pressure than does the organic side. For this reason, it is economically advantageous to contain the steam inside the tubes and to have the lower pressure organic on the shell side, thus minimizing the required shell thickness. Generally, once-through generators are designed with steam in the tubes and recirculating generators with steam on the shell side. However, the reverse conditions are possible in both cases.

b) In a once-through steam generator, all of the heat transfer occurs in a single shell; i.e., preheat, steam generation, and superheat are all accomplished in one continuous tube. Hence, for given coolant inlet and exit temperatures, the once-through generator can produce higher

TABLE 0950-1
FABRICATION CAPABILITIES OF STEAM GENERATOR MANUFACTURERS VISITED

	MANUFACTURER NO. 1	MANUFACTURER NO. 2	MANUFACTURER NO. 3
Shell Capability	<ol style="list-style-type: none"> 1. Presently fabricating a number of 8-10' dia x 8-10" wall thickness (w.t.) reactor vessels 2. Also fabricating commercial vessels, of order of 100' long x ~10' dia x ~4" w.t. 3. Fabricating many commercial boiler drums ranging in length from 8-10' to 50-60', dia 4-5' to 6-8', and w.t. ~4-8" 4. Building reactor vessel 251" ID x 6-8" w.t., 650 tons 	<ol style="list-style-type: none"> 1. Presently fabricating large reactor vessels 12' dia, order of magnitude: x ~10" w.t. 2. Presently fabricating large steam generators, shells ~12' dia x ~4" w.t. 	<ol style="list-style-type: none"> 1. Presently fabricating many large reactor vessels, largest shell ~20' dia x 10" w.t. 2. Many shell courses going through fabrication of 10-12' dia x 6" and 8" w.t. variety 3. Fabricating horizontal steam generators, ~10' dia x ~4" w.t. x ~40' long
Tubesheet Capability	<ol style="list-style-type: none"> 1. Largest tube plate fabricated at present is 6' dia x 14" thick 2. Fabricating replacement steam generators ~5' dia x ~40' long, tubesheets 12-14" thick 3. Fabricating large ring forgings for reactor head hold-down rings of order of 18" thick (in radial direction) x ~30" high, drilled in 30" direction for bolt holes 	<ol style="list-style-type: none"> 1. Presently fabricating steam generators with ~10' dia x 24" thick tubesheets (vert. U-tube type) 2. Also fabricating other vertical U-tube type steam generators slightly larger 3. Tubesheet drilling limited to ~1" dia 	<ol style="list-style-type: none"> 1. No vertical steam generators being fabricated at present, but do not feel that large dia thick tubesheets, order of 10' dia x 24" thick, any particular problem 2. Tubesheet drilling limited to ~1" dia

TABLE 0950-1 (continued)

	MANUFACTURER NO. 1	MANUFACTURER NO. 2	MANUFACTURER NO. 3
Tubesheet Capability (cont'd)	4. Have no particular reservations about fabricating steam generators with large thick tubesheets, ~10' dia x 24" thick 5. Tubesheet drilling limited to about 1" dia.		
Tube-to-Tubesheet Welding Capability	1. Have perfected automatic tube-to-tubesheet welding method which produces excellent uniform & consistent welds 2. Tube-to-tubesheet joints are of the through-tube variety	1. Have perfected and are using automatic tube-to-tubesheet welding method which produces excellent, uniform, and consistent welds 2. Tube-to-tubesheet joints are of the through-tube variety	1. Have perfected tube-to-tubesheet welding method for naval nuclear steam generators 2. Tube-to-tubesheet joints are of the through-tube variety
Machining Capability	1. Grid plates and tubesheets up to 20' dia	1. Grid plates and tubesheets up to 20' dia	1. Grid plates and tubesheets up to 20' dia
Heat Treating Capability	1. Car bottom furnace up to 12' dia x 50' long 2. Local ring heating - any size	1. - 2. Local ring heating - any size	1. Car bottom furnace up to 14' dia 2. New furnace 35 x 35 x 55' long
Quenching Capability	1. Brine quench tank, 14' dia x 15' deep	1. -	1. -
X-Ray Capability	1. Up to 16" thick	1. Up to 16" thick	1. Up to 36" thick
Weight Capacity	1. 500 ton crane	1. 250 ton crane	1. 1000 ton crane
Shipping	1. 14' dia x 50-60' by rail 2. No limit by barge	1. 14' dia x 50-60' by rail 2. No limit by barge	1. 14' dia x 50-60' by rail 2. No limit by barge

TABLE 0950-2

STEAM GENERATORS ANALYSIS DATA

TYPE	NO. REQUIRED* AND DIMENSIONS	REMARKS
RECIRCULATING NATURAL CONVECTION VERTICAL SHELL DRUM	(6) EVAPORATORS 14' DIA x 60' HIGH	
	(6) SUPERHEATERS 6' DIA x 45' LONG (OPTIONAL HORIZONTAL OR VERTICAL)	
RECIRCULATING NATURAL CONVECTION HORIZONTAL KETTLE DRUM	(3) EVAPORATORS 10-12' DIA x 90' LONG	
	(3) SUPERHEATERS 8' DIA x 45' LONG	
ONCE-THROUGH	(24) SHELLS 2 1/2' DIA x 120' LONG (VERTICAL ORIENTATION PREFERRED)	ACHIEVE HIGHER STEAM PRESSURES FOR GIVEN COOLANT CONDITIONS. REQUIRE MORE INTER- CONNECTING PIPING AND ULTRA HIGH PURITY FEEDWATER (~10 ppb)
ONCE-THROUGH	(3 OR 4) UNITS 9-11' DIA x 125' LONG (VERTICAL ORIENTATION PREFERRED)	

* FOR 1000 MWe HWO CR

pressure steam than recirculating generators, unless the latter are equipped with preheaters or economizers.

c. The once-through type of steam generator requires ultra-high purity (~ 10 ppb) feedwater to minimize fouling of heat transfer surfaces. This is generally accomplished by providing a condensate polishing system consisting principally of demineralizers and filters.

d. Modular once-through steam generator units require a significant amount of manifold or interconnecting piping.

e. Once-through generator instrumentation, control, and startup system requirements are greater than for recirculating generators.

2. Main Steam and Feedwater System

a. Steam Conditions

Steam cycle heat balances were calculated to determine the variation of the heat rate with turbine inlet steam conditions, the steam generator and reactor thermal requirements, and the cycle arrangement. This analysis considers the use of two presently available 544-Mwe turbine generators in a non-reheat regenerative cycle with five feedwater heaters. The turbine type used is a tandem-compound, 1800 rpm, non-reheat, four flow with 43-in. last-stage blades. The generator is a 640-Mva, 22-kv hydrogen-cooled unit. Additional analyses based on the use of a single 1000 Mwe (nominal) turbine will be conducted.

Figure 0950-7 shows the steam generator thermal requirements for a gross output of 1088 Mwe for turbine inlet pressures ranging from 500 to 1500 psig, inlet temperatures of 675, 725, and 775°F, and a condenser pressure of 1.5 in. Hga. Figure 0950-8 shows the gross and net steam cycle heat rates in the

stated range of steam conditions. The steam plant cost data shown in Figure 0950-9 are based on the engineering studies conducted by Sargent and Lundy and reported in their report SL-2008, dated June 30, 1963. The equipment costs include the turbine generator and accessories, condensers and accessories, circulating water system, feedwater heaters, boiler feed pumps, condensate pumps, piping and insulation, and demineralizing and electrical equipment.

The foregoing data are being used in the plant system parameter optimization program, and for sizing the reactor, coolant system, and steam generators. Process and instrumentation diagrams have been prepared based on the selected reference plant which operates with steam conditions of 900 psig and 725°F at the turbine inlet.

b. D₂O Moderator for Feedwater Heating

Steam cycle heat balances were calculated to determine the improvement in the steam cycle heat rate or efficiency by utilization of the D₂O heat for feedwater heating.

Figure 0950-10 shows the variation of the net and gross steam cycle heat rates with pressure, at a constant turbine inlet temperature of 725°F, when the moderator heat is used for feedwater heating. Figure 0950-11 shows the improvement in the net cycle efficiency and heat rate when compared with the normal feedwater cycle. The heat from the D₂O was assumed to be 5% of the coolant thermal power and is utilized in the first stage of feedwater heating after the condenser.

Figure 0950-12 shows the justified incremental investment resulting from fuel, coolant makeup, and capital cost savings when the moderator heat is used for feedwater heating. A maximum justified incremental investment of 1.6 and 1.0 million dollars, with 7% and 12% annual fixed charges respectively, occurs

at turbine inlet pressures from 800 to 940 psig. The justified incremental investment was calculated using the following basis:

Reactor ΔT ($^{\circ}\text{F}$)	100
Reactor ΔP (psig)	100
Fuel cost (\$/million Btu)	0.10
Capacity factor	0.80
Coolant cost (\$/lb)	0.15
Radiolytic decomposition rate (lb/Mwt-hr)	0.495
Incremental capability charge (\$/Kwe)	75
Annual fixed charge rates (%)	12 and 7

A comparison of the D_2O heat rejection methods, included in another section of this report, shows that the use of the D_2O for feedwater heating is economically justifiable.

c. Partial Load Operation

The calculation of heat balances at 60, 40, and 20% load was initiated for the reference cycle (900 psig-725 $^{\circ}\text{F}$ -1.5 in. Hga) in order to provide data which will assist in establishing the design criteria for the reactor, coolant, plant control, and steam generator systems, and for future economic comparisons between the two-turbine and single-turbine plant.

d. Single 1000-Mwe Turbine Plant

Information on the tandem-compound 1000-Mwe turbine generators for application in the 1970's was requested from General Electric and Westinghouse.

Preliminary heat rates and plant costs were obtained for the steam conditions under consideration, based on data provided in the Sargent and Lundy report, SL-2008. To check the efficiency level and as a basis for future evaluation work, heat balances will be calculated as soon as the latest data are received from the turbine manufacturer.

3. Computer Program Development

The main heat transfer system optimization philosophy and logic were established and summarized into a specification for the optimization codes. Existing codes were examined, but none was found to be completely satisfactory for the HWOCR studies. Work was started on the preparation of new computer codes to meet the optimization requirements.

A decision was made that the primary (organic) loop optimization package should be able to accommodate both the recirculating and the once-through type heat exchanger designs. To date, one type of recirculating boiler, the kettle-type design, has been programmed. Test returns have indicated good results, and use of the package has begun. The plotting of results against selected parameters is a built-in feature of this program. The sequence and logic for the once-through type boiler programming was determined.

For the secondary (steam) loop optimization package, effort was limited to the writing of a steam cycle analysis routine. Following the practice currently recommended by manufacturers and in the literature, the treatment of a regenerative steam cycle was generalized so that this routine will analyze steam cycle with up to 10 regenerative heating stages. The routine was written and developed to the checkout phase. Work was then started on the collection of equipment cost data.

One of the major difficulties experienced in the steam cycle analyses was the generalization of steam expansion. The path of steam expansion is a function of its initial and end conditions. An approximation was made for initial conditions of 665 to 1265 psi and 625 to 775°F, and an end condition of 1.5 in. Hg. Several tests indicated that this arrangement is satisfactory. However, the validity of the approximation must be re-established if the steam conditions are revised.

Development of a digital computer program to calculate the steam cycle heat balances in the HWOCR Program was initiated also. Among the inputs to the program are the turbine type, number and type of the feedwater heaters, turbine inlet steam conditions, condenser pressure, heater terminal differences and drain cooling, feedwater pump pressure rise, steam generator feedwater temperature and pressure, steam flow to the turbine, reheater conditions, and generator characteristics.

C. COOLANT PURIFICATION

Based on the PNPf type of design and currently available coolant decomposition rate data, the preliminary conceptual design of a 1000-Mwe plant coolant distillation system for decomposition product (HB) control was completed. It was calculated that the maintenance of an HB level of 10% would require a distillation rate of 12,000 lb/hr. This rate corresponds to the processing of the equivalent of one coolant inventory (assumed to be 400,000 gal) in ~ 11 days. A cost estimate, based on this preliminary design, indicated a capital and operating distillation cost of ~ 0.05 ¢/lb of coolant distilled. Due to the low coolant through-put rate, however, it was concluded that this type of distillation system would require an additional coolant clean-up process, such as adsorption, in order to maintain the concentration of impurities in the coolant at an acceptably low level.

As an alternative (which would not require an auxiliary coolant clean-up process) to the above type of distillation system, the preliminary conceptual design of a high through-put "deep distillation" system was initiated. Based on a corrosion rate of $0.05 \text{ mg/cm}^2\text{-month}$ and an anticipated carbon steel surface area of $630,000 \text{ ft}^2$, it was calculated that a coolant distillation rate of $\sim 43,000 \text{ lb/hr}$ would be required to maintain the corrosion products concen-

trations in the coolant at ~ 3 ppm. (This is the general level at the PNPf.) This processing rate corresponds to the distillation of the equivalent of one coolant inventory in ~ 3 days.

The results of the adsorption development program previously conducted at AI and of the subsequent adsorption tests at OMRE were reviewed. In addition, discussions were held with AECL personnel relative to their experience with adsorption systems for the purification of organic coolant. From this background, a preliminary conceptual design for a 1000-Mwe HWO CR plant adsorption system was developed. The system consists of three adsorption columns, each $\sim 4\frac{1}{2}$ ft in diameter by 15 ft high. The operating plan involves the use of two columns in series while the third is replenished and prepared for return to operation. Since the time required for replenishment is far less than the estimated on-stream life, the third column will usually be in a stand-by condition. However, in the event more rapid coolant purification is required temporarily, all three columns may be used concurrently. Component requirements auxiliary to the three adsorption columns are slight and involve principally a downstream guard filter and a flow control valve. The concept thus far developed does not include regeneration of the spent adsorbent material.

Thorough literature searches and meetings with Phillips Petroleum Company personnel provided the bases for evaluations of the economic and technical aspects of the catalytic hydrocracking of irradiated organic coolant in HWO CR plants. Utilizing this information, an economic analysis of the hydrocracking process was made, based on two plant sizes and HB levels of 10 and 40%, and on assumed hydrocracking system capital costs. The resulting coolant makeup costs are listed below.

<u>Plant Size (Mwe)</u>	<u>HB Level (%)</u>	<u>Assumed System Cost (\$000)</u>	<u>Approximate Total Coolant Makeup Cost (\$/lb)</u>
1000	10	3,000	13
1000	10	2,000	11
1000	10	1,000	8
1000	40	2,000	19
1000	40	1,000	13
1000	40	500	10
300	10	1,000	17
300	10	500	12
300	10	250	10
300	40	500	23
300	40	250	18

Based on a coolant cost of 15¢/lb, it can be seen that the economics of a hydrocracker installation are questionable in the case of the smaller plants, depending on the operating concentration of HB in the circulating coolant. However, a significant savings (~\$500,000/yr) can be realized for the 1000-Mwe plant with a 10% HB level and an initial system investment of \$1,000,000.

The technical evaluation of the hydrocracking process was based on the preliminary design and safety analysis of the hydrocracker being developed by Phillips Petroleum Company for PNPF. This evaluation covered the following areas: (1) catalytic reactor design; (2) catalyst technology; (3) energy and hydrogen requirements; (4) safety considerations; and (5) composition, properties, and stability of hydrocracked products. The evaluation indicated general technical feasibility but also revealed uncertainties in process parameters.

D. D₂O MODERATOR SYSTEM

1. General

During the report period, considerable research was done on literature devoted to the development of heavy water technology. Particular attention is being given to the design and operating experience of reactor plants, such as the CVTR and the Canadian NPD and WR-1, which represent the latest in technology and also contain systems similar to those required for the HWO CR. In addition to the general review of current technology, which will continue throughout the program, a trip to Savannah River, the CVTR, and WR-1 to gain firsthand knowledge of heavy water systems was made by personnel involved in this portion of the HWO CR design.

2. Heavy Water Process Systems

The heavy water portion of the HWO CR plant conceptual design has been scoped out to include the following process systems:

- a) Moderator cooling system - includes the equipment and facilities required to remove heat generated in the moderator during both normal and emergency conditions.
- b) Moderator purification system - maintains the required purity and cleanliness of the heavy water.
- c) Moderator cover gas system - includes facilities for supplying and venting the cover gas, moderator system pressure control, and recombination of gaseous products or decomposition.
- d) Moderator fill, drain, and storage system.
- e) D₂O recovery system - includes facilities for the recovery of pure or downgraded heavy water from vented gas, equipment area atmosphere, and other parts of the system from which D₂O may be lost during operation or maintenance.

f) Poison injection system - provides for an emergency shutdown system to back up the normal reactor control system.

g) Moderator sampling and leak detection system.

Other systems, such as a reactor control system utilizing heavy water (Hy-Ball system), or on-site D₂O upgrading facilities, will be designed if these systems are adopted as part of the plant design.

A preliminary conceptual design has been prepared for the first three systems listed; schematic drawings of these systems are also being made.

3. Moderator Cooling System

a. Optimization

An optimization study of this system has been made in order to determine the most economical method, consistent with safety, of removing the heat generated in the moderator during plant operation. Three basic methods of heat removal were considered:

1) Dumping the heat directly to plant cooling water, i.e., river or ocean water.

2) Dumping the heat to an intermediate, closed, cooling water system which, in turn, would transfer the heat to plant cooling water.

3) Transferring moderator heat to boiler feedwater in the first stage BFW heater.

Considering the aspect of safety, method 1 is the least desirable since a leak in the D₂O/CW heat exchanger could result in a leak of radioactive material, particularly tritium, outside the plant confines. Methods 2 and 3 do not have this disadvantage since the heat sink is essentially a closed system; thus, a leak would not result in escape of radioactivity from the plant. However, a preliminary investigation of the problem by safeguards personnel

resulted in the opinion that with proper safeguards, such as leak detectors and quick-closing isolation valves, a system to dump moderator heat directly to river or ocean water could probably be designed and still meet all existing safety regulations. Hence, an economical evaluation of all three methods was made.

To arrive at reasonably optimum conditions for any given system, the following items were considered variables (ranges of variation are in brackets): (1) pipe size and D₂O velocities [to 40 ft/sec] , (2) moderator ΔT [20 to 150°F] , (3) cooling water ΔT [20 to 60°F] , and (4) moderator outlet temperature [190 to 270°F] . Cooling water temperature was assumed to be 65°F and inlet temperature at the first stage BFW heater to be 92°F (condenser pressure $1\frac{1}{2}$ in. Hga.).

The first part of the study assumed a total heat generation in the moderator of 6% of thermal power to the coolant. The cost of each system was estimated at a given D₂O outlet temperature. Moderator ΔT and other applicable items were varied to arrive at an optimum cost for each outlet temperature. The results are shown in Figure 0950-13. Each point plotted represents the optimum, or bottom, of a curve which could be plotted for a particular moderator outlet temperature. The curves show that for the temperatures considered, heat rejection to a closed cooling water system is the most expensive method considered. The initial capital cost of the method using boiler feedwater is greater than that for directly dumping the heat; however, the annual return, or savings due to the increased plant efficiency, makes the net annual cost of the boiler feedwater heating method the lowest. For this case, the use of moderator for BFW heating resulted in a decrease in the net cycle heat rate of 62 Btu/kw-hr and an increase in cycle efficiency of 0.24%.

The study was then extended and refined to compare the system economics at the three principal steam cycles being considered: 600, 900, and 120 psig (all

at 725°F). For these cases, a total moderator heat generation rate of 5% was used. Since the method using a closed cooling water loop was seen to be most expensive and provided little, if any, advantage from a safety standpoint over the BFW method, this scheme was not considered further. The results of this part of the study are shown in Figures 0950-14, -15, and -16. The sharp decrease in annual return shown for the 1200-psig steam cycle (Figure 0950-16) is due primarily to the increased heat rejection to the main condenser from the external turbine moisture separation equipment. External moisture separators were not considered for the 600 and 900-psig steam cycle.

The annual charges include the cost of the moderator cooling system, with its heavy water inventory (excluding the calandria); the cost of the cooling water systems; all associated power costs; and, for the BFW heating method, the costs of adjustments to the feedwater system, such as an increase in main condenser size, elimination of the first extraction steam feedwater heater, and the additional condensate booster pumps required.

In all cases, the evaluation of the D₂O system was based on four 1/3 capacity loops; however, due primarily to the high heat exchanger costs, this is more excess capacity than would be provided in an actual design. Reduction of the excess capacity and its costs would tend to favor, cost-wise, the boiler feedwater heating method, since this is a larger system.

b. Conceptual Design

Based on the results of the optimization study, the conceptual design of the moderator cooling system has been started, using boiler feedwater as a heat sink. A reference D₂O outlet temperature of 200°F has been chosen to avoid pressurization of the calandria which would thicken the calandria tubes excessively. The addition of reactor capital costs and extra fuel costs for the pressurized case would detract from the savings indicated in Figures 0950-13 through -16. The system is being designed for a minimum heat rejection capabi-

lity of 527×10^6 Btu/hr (110% of nominal full-power capacity). The system will consist of four moderator coolers, each pair of coolers representing the first-stage feedwater heater for one of the two 500-Mwe turbine-condensate cycles, and four pumps located upstream of the coolers. The configuration of the system, i.e., the manifolding and number and location of calandria connections, will be determined by the requirements of removing decay heat, particularly during such postulated accident conditions as system ruptures or a calandria leak. Investigation of these problems, as they affect system design, is currently under way.

4. Moderator Purification System

The preliminary conceptual design of the moderator purification system consists of a small, separate loop circulating approximately 50 gpm of D_2O through a cation bed and mixed-bed demineralizers. A pre-filter is provided for general system clean-up during plant startup and early operation, and an after-filter is included to protect against resin carryover from the demineralizers. A cooler is also included to decrease the D_2O temperature to approximately 100° prior to entering the demineralizers.

5. Moderator Cover Gas System

The major portion of the cover gas system consists of a closed loop through which the cover gas (helium) is continuously circulated from the calandria overflow (or head) tank, through the expansion tank, and to a recombiner. The recombiner (palladium on alumina pellets) is used to recombine the products of gaseous decomposition occurring within the calandria. The system is designed to circulate approximately 100 scfm and to maintain a maximum concentration of excess D_2 in the gas stream of 0.5%. Circulation is provided by two full-size, heavy-water-sealed, rotary blowers. Heat exchangers to preheat the gas are not provided; the recombiner will be electrically heated to prevent catalyst poisoning by moisture at system startup.

E. AUXILIARY PROCESS SYSTEMS

Based on a study of past experience, current technology, and HWOGR plant requirements, a coolant degasification and pressurization concept basically similar to the PNPf was selected. A shutdown cooling system was developed in which cooling immediately following shutdown is accomplished by dumping steam to the main condensers via a turbine bypass; subsequent cooling (approximately 3 hr after shutdown) utilizes a shutdown heat exchanger which generates steam for auxiliary steam services. Based on an investigation of coolant shipping methods, a new or makeup coolant system which includes insulated steam-coiled railroad tank cars is envisioned.

Preliminary studies of the radioactive waste systems indicate that these will conform generally with the arrangements in effect at PNPf. Utilization of mechanical pumps instead of steam jet ejectors for the vacuum source in the waste gas system is a contemplated minor departure.

Design studies were also applied to other miscellaneous auxiliary process systems. In conjunction with these studies, preliminary flow and P&I diagrams were prepared and systems descriptions initiated.

F. INSTRUMENTATION AND CONTROL

In the area of instrumentation and control, effort was applied principally to obtaining background information on the application of devices to previously designed and constructed organic process systems.

In coordination with the development of the process systems, preliminary work was also done in initiating P&I diagrams for these systems; process parameter measurement and control requirements were investigated and applicable types of equipment outlined.

G. PLANT STRUCTURES, EQUIPMENT, PIPING, AND ELECTRICAL

The following drawings are presented as Figures 0950-17 through 0950-20:

1) Figure 0950-17 (Drawing Number R-HWR3-GA001) shows the 8300-Mwt HWOCR fully contained in a 250-ft diameter containment sphere. This sphere can contain a heat transfer system consisting of up to twelve loops, each with one pump, one steam generator, and one filter. The D₂O moderator cooling equipment and the spent fuel storage pool are also within the containment. This drawing was prepared in order to ascertain that subsequent 1000-Mwe plant arrangements could serve as prototypes and be extrapolated to the 8300-Mwt plant size.

2) Figure 0950-18 (Drawing Number R-HWR2-GA002) shows the 1000-Mwe HWOCR fully contained in a 175-ft diameter containment sphere. The heat transfer system consists of three loops. The equipment contained is as listed for Figure 0950-17.

3) Figure 0950-19 (Drawing Number R-HWR2-GA003) also shows the 1000-Mwe HWOCR fully contained in a 175-ft diameter sphere. This drawing differs from that shown in Figure 0950-18 in that the heat transfer loops may include any of five different arrangements of the modular or once-through steam generator.

4) Figure 0950-20 (Drawing Number R-HWR2-GA004) shows the 1000-Mwe HWOCR partially contained in a cylinder 100 ft in diameter and 170 ft high. The reactor complex and the D₂O moderator cooling system are within the containment. The heat transfer loops are outside the containment. The heat transfer system consists of a distribution header feeding four loops, each with one pump, one filter, and two vertical once-through steam generators. This drawing also shows the feasible application of a cylindrical housing to the complete containment of the primary system of this reactor plant.

Numerous sketches and layouts were made of various schemes during the evolution of the above drawings. Among the factors which came under consideration during this effort were the following:

- 1) Equipment location, outline, and size, notably the steam generators
- 2) Piping arrangement and stresses
- 3) Component installation, operation, and maintenance
- 4) Extent or degree of containment
- 5) Scope of containment vessel
- 6) Containment pressure
- 7) Fuel handling

As expected, partial containment (containment of the reactor complex and the D₂O cooling system only) shows a marked cost advantage over full containment (containment of all heat transfer and fueling equipment) with no compromise in piping material or coolant volume.

Preliminary comparative cost studies were made for various containment configurations and construction techniques. Basically, the studies compared spherical containment against cylindrical, and steel-vessel construction against various types of concrete construction. The results of these studies were plotted as cost curves (Figures 0950-21 through -26) in dollars per cubic foot of containment volume against design pressures. A containment pressure range from 5 to 15 psig was studied. The curves indicate that, within the pressure range and size of containment investigated (130 ft to 250 ft diameter), spherical containment is less economical than cylindrical, and that above ~10 psig, prestressed concrete has an economic advantage over steel construction.

A study was made of the hazardous classification which will have to be given to electrical installations in certain areas of the HWOCR plant. Class I,

Division 1, Group D, as defined by the National Electrical Code, is the most rigid classification which will be required. The requirement for this classification will be minimized where possible by the use of forced ventilation and inert gas purging, and by the judicious location of electrical equipment.

H. LARGE ORGANIC COMPONENTS TEST LOOP

A detailed study was made of the areas of uncertainty which exist relative to the main organic circulating pumps and large valves. This study indicated that proof-testing of this equipment is desirable to minimize the probability and frequency of plant outages due to inadequate operation or malfunction of these components. While deficiencies could be detected and corrected during plant pre-operational testing (without prior testing in a test facility), extensive delays at this stage could prove much more costly than a test facility.

A conceptual design and preliminary cost estimate for a test loop was completed. The main piping in the loop is 42 in. in diameter. The loop is designed primarily to accommodate a 90,000-gpm pump, which is the currently estimated most probable size for the 1000-Mwe HWO CR plant, but it could be used for larger flow rate pumps within the limits of permissible fluid velocities and the pump drive. The completed conceptual design package of the test loop consists of process and instrumentation diagrams for organic flow, blanket gas, and instrument air; a single-line electrical diagram; an equipment layout drawing; and a report.

Figure 0950-27 is the organic flow diagram.

I. UC FUEL EVALUATION

1. UC-Rod-Cluster Physics Parameter Survey

A core physics parameter survey is in progress for a rod-type fuel element. Based on prior calculations for this type reactor, a 19-rod UC cluster of finned

rods was selected for the initial phase of the survey. The independent variables considered were fuel rod diameter, fin height, fuel enrichment, and deuterium-to-uranium atom ratio (lattice pitch). Three values for each variable were selected for a total of 81 cases. The following listing contains the values for the variables considered:

<u>Fuel Rod Diameter (in.)</u>	<u>Fin Height (% Fuel Dia)</u>	<u>D/U Ratio</u>	<u>Enrichment (wt % U²³⁵)</u>
0.500	5	20	0.711
0.625	10	30	1.0
0.750	15	40	1.25

There are several other core parameters which are required for the physics analysis. Many of these parameters depend on the result of heat transfer and stress analysis on the cases studied. The initial values of these parameters assumed for the physics calculations are presented below. The effects of changes in these parameters will be determined by perturbation of the cases analyzed in the initial study.

Cladding material	SAP
Cladding thickness (in.)	0.025
Number of fins/rod	14
Fin width (in.)	0.030
Gas bond thickness (in.)	0.003
Carbon content of fuel (wt %)	4.8
Process tube material	SAP
Process tube thickness (in.)	0.200
Lattice geometry	square
Average specific power (kw/kg)	6

(continued)

Average fuel temperature (°F)	1200
Average coolant temperature (°F)	650
Average moderator temperature (°F)	190

Some of the physics results of the initial survey are presented in Table 0950-3. Only data for fin heights which are 10% of the fuel rod diameter are given. The results of these survey calculations as presented in this table should be viewed only in a relative sense since no reactivity allowances were made for such items as fuel management scheme, detailed structural requirements and control margin.

2. Nested Tube Physics Parameter Study

A physics parameter survey is in progress for an annular ring geometry HWOCR fuel element shown in Figure 0950-28. This element consists of a control rod and two concentric rings of cast UC, clad in non-freestanding Zircaloy. The nuclear survey of this element consisted of four enrichments (0.7, 0.9, 1.15, and 1.45%) at each of three D/U ratios (20, 30, and 40). Values of parameters held constant for the survey are given in Table 0950-4. Some of the results of the physics parameter study are given in Table 0950-5.

3. HWOCR Physics Methods Comparison Study

A physics methods comparison study was conducted in order to ensure that resulting HWOCR core designs from each separate facility within the HWOCR project are all on an equivalent physics basis.

The basic goal of the comparison study was viewed not as a core design problem, but rather as a comparison of physics calculational methods and cross-section libraries. Therefore, test case questions requiring thermodynamic analysis were eliminated; e.g., determination of a power coefficient. The physics analysis was limited to fuel cell spectral calculations and single space point depletion calculations for an infinite array of fuel cells. The overall effect of these simplifications reduces the study to a comparison be-

TABLE 0950-3

PHYSICS RESULTS

FUEL DIA (in.)	FEED ENRICHMENT (% U ²³⁵)	D/U RATIO	DISCHARGE BURNUP (MWD/MTU)	DISCHARGE ENRICHMENT (% U ²³⁵)	DISCHARGE FISSILE Pu CONCENTRATION (gr Pu/Kg U)
0.500	0.711	20	--	--	--
0.500	0.711	30	4,900	0.32	2.63
0.500	0.711	40	6,900	0.23	2.91
0.500	1.0	20	12,600	0.22	4.26
0.500	1.0	30	15,900	0.12	4.11
0.500	1.0	40	17,000	0.09	3.90
0.500	1.25	20	20,500	0.14	5.06
0.500	1.25	30	23,800	0.07	4.65
0.500	1.25	40	24,700	0.05	4.33
0.625	0.711	20	--	--	--
0.625	0.711	30	7,080	0.22	3.06
0.625	0.711	40	8,310	0.17	3.08
0.625	1.0	20	14,700	0.16	4.37
0.625	1.0	30	17,400	0.09	4.12
0.625	1.0	40	18,100	0.07	3.90
0.625	1.25	20	22,600	0.11	5.21
0.625	1.25	30	25,300	0.06	4.60
0.625	1.25	40	25,800	0.04	4.27
0.75	0.711	20	4,900	0.33	2.76
0.75	0.711	30	8,000	0.19	3.18
0.75	0.711	40	8,900	0.16	3.14
0.75	1.0	20	16,000	0.14	4.41
0.75	1.0	30	18,300	0.08	4.12
0.75	1.0	40	18,700	0.07	3.87
0.75	1.25	20	23,800	0.09	5.05
0.75	1.25	30	26,100	0.05	4.58
0.75	1.25	40	26,300	0.04	4.24

TABLE 0950-4

ANNULAR ELEMENT SURVEY

Element has central rod and 2 concentric fuel rings, cladding is unfinned Zr-2

ELEMENT DIMENSIONS

REGION	MATERIAL	OUTER RADIUS (in.)	THICKNESS (in.)	AREA (in. ²)
1	FUEL	0.350	0.350	0.3848
2	Zr	0.375	0.025	0.0569
3	ORGANIC	0.550	0.175	0.5085
4	Zr	0.575	0.025	0.0884
5	FUEL	1.075	0.500	2.5918
6	Zr	1.100	0.025	0.1708
7	ORGANIC	1.240	0.140	1.0273
8	Zr	1.265	0.025	0.1967
9	FUEL	1.650	0.385	3.5257
10	Zr	1.675	0.025	0.2611
11	ORGANIC	1.790	0.115	1.2518

TOTAL 10.0638

FLOW TUBE I D (in.) 3.580 (ZIRCALOY)

FLOW TUBE THICKNESS (in.) 0.060

CALANDRIA TUBE I D (in.) 3.900 (ZIRCALOY)

CALANDRIA TUBE THICKNESS (in.) 0.060

COOLANT SANTOWAX, 10% HB

T_{OUT} (°F) 725T_{CLAD} (MAX.) (°F) 850

RADIAL P/A 1.3

AXIAL P/A 1.5 "ROOFTOP"

TABLE 0950-5

PHYSICS RESULTS, ANNULAR UC FUEL ELEMENT SURVEY

CASE	FEED ENRICHMENT (% U ²³⁵)	D/U	DISCHARGE BURNUP (MWD/MTU)	DISCHARGE ENRICHMENT (% U ²³⁵)	DISCHARGE FISSILE Pu CONCENTRATION (gr Pu/Kg U)
1	0.7	20	4,400	0.36	2.84
2	0.7	30	8,600	0.17	3.45
3	0.7	40	9,600	0.13	3.38
4	0.9	20	12,800	0.18	4.54
5	0.9	30	15,800	0.09	4.26
6	0.9	40	16,400	0.07	3.97
7	1.15	20	20,700	0.13	5.47
8	1.15	30	24,000	0.06	4.86
9	1.15	40	24,300	0.04	4.44
10	1.45	20	29,800	0.10	6.33
11	1.45	30	33,400	0.04	5.44
12	1.45	40	33,100	0.03	4.89

tween different types of nuclear parameter survey schemes rather than a comparison of final design calculations.

A test problem consisting of a 37-rod cluster of UO_2 fuel enriched to 1.5 wt % of U^{235} with SAP cladding, organic coolant (OMP, 10% HB), SAP process tube, Zr-2 calandria tube, and D_2O moderator was conceived and mutually agreed upon by Combustion Engineering and Atomics International and accepted by representatives of ORNL and AEC. The case was identified with precision in survey methods yielding information on initial reactivity, thermal flux distribution, resonance absorption, fast effect, temperature and void coefficients of the coolant, temperature coefficient of the moderator, Doppler coefficient, and isotope depletion for fuel exposures to 20,000 Mwd/MTU.

The initial results of the AI and CE calculations are tabulated below. A discussion of these results and the methods used by each group brought to light various differences in the calculations. The calculations are currently being examined in order to better understand the discrepancies.

Physics Methods Comparison Study Results

	<u>AI</u>	<u>CE</u>
Initial k_∞	1.316	1.3255
Resonance absorption	0.2107	0.2065
Fast effect:		
Term (a)*	0.0377	0.04006
Term (b)**	0.530	0.5416
Reactivity coefficients:		
a) Doppler $10^{-6} \Delta k/k$ $^{\circ}\text{F}$	-7.95	-8.95
b) Coolant temp $10^{-6} \Delta k/k$ $^{\circ}\text{F}$	+6.43	+9.90
c) Coolant void $\Delta k/k$	+0.0467	+0.0421
d) Moderator temp $10^{-5} \Delta k/k$ $^{\circ}\text{F}$	-8.00	-2.48

* Term (a): fission neutrons produced by fissions above 0.82 Mev per total fission neutrons produced.

** Term (b): number of neutrons slowing down from above 0.82 Mev per total fission neutrons produced.

The test case was analyzed at AI using physics survey techniques currently employed in the HWOCR optimization study.

J. SAFETY ANALYSIS

A comprehensive list of potential accidents requiring safety analysis was prepared. This list will be used as a working document in design as well as safeguards to coordinate and maintain current safety analysis results and their interrelationships with design. The following major areas are covered:

- 1) Single failure accidents - coolant system
- 2) Single failure accidents - D₂O system
- 3) Single failure accidents - steam generator
- 4) Single failure accidents - auxiliary systems
- 5) Single failure accidents - general
- 6) Single failure accidents - core
- 7) Single failure accidents - fuel handling
- 8) Multiple failure accidents

Consideration of containment requirements and preliminary containment criteria was begun. The containment designs for other plants were reviewed in order to consider and apply particular features of other containment methods which are appropriate for the HWOCR. To provide additional information relative to containment criteria, an analysis was made of a major coolant line rupture. The results of this analysis indicate the range of pertinent containment criteria.

A preliminary system location study was also begun. The purpose of this study is to determine what systems and components of the plant may be safely placed outside the containment. This study is being conducted in conjunction with plant structures and equipment arrangement activities and compares possible reduction in containment size against additional engineered safeguards. A sub-

stantial reduction in containment size is realized by placing the main heat transfer loops and steam generators outside the containment. With this arrangement, double isolation valves are recommended at the containment interface.

III. EVALUATION OF EFFORT TO DATE

Substantial progress has been made on a wide front toward the attainment of the stated objectives. Systems optimizations and conceptual designs have been initiated. The bases for development of the demonstration plant (300 Mwe) conceptual design have been established.

The computer code for the optimization of the main heat transfer system has progressed to a point where detailed studies for recirculating steam generator cases can be started as soon as reactor cost data are available. The results obtained thus far have shown some general economic trends which can be summarized as follows:

- a) The steam generators are the most influential items in the optimization study.
- b) The optimum range of reactor ΔT 's for the primary heat removal system is approximately 150 to 225°F.
- c) The 900 and 1200-psig steam cycles result in lower power costs than the 600-psig case when excluding the reactor costs.
- d) The pumping power required by the primary coolant pumps warrants consideration of steam turbine drives.

The performance of the completed optimization code for coolant loops with recirculating steam generators is quite satisfactory. Using assumed equipment costs, the code is able to obtain general cost profiles for different parameters. It is possible that the subroutines in the package can also be used separately to develop preliminary designs for heat exchangers.

Satisfactory progress has been made in the development of the steam cycle code.

The steam cycle analysis effort has provided the basis for the plant optimization and design efforts by establishing the steam generator and reactor thermal requirements for the generation of 1000 Mwe at various steam pressure and temperature conditions. The steam cycle analysis has also provided the operating conditions for a reference plant and the cycle arrangement to be used. Cycle heat rates or efficiencies have enabled the calculation of plant overall efficiency for use in the optimization program.

The steam cycle calculations have indicated the improvement in efficiency obtained using the D_2O heat for feedwater heating and the resulting justifiable incremental investment. These data are fundamental in the economic investigation of the moderator heat rejection methods.

The steam generator evaluation studies which have been performed for the reference set of system conditions provide a background and basis for further study, refinement of results, future determinations for other than the reference design conditions, and, ultimately, the steam generator selection for the HWOCR. The information obtained from manufacturers indicates that very large units can be constructed.

The investigations thus far conducted have indicated that design and fabricating capabilities for the main heat transfer system coolant pumps exist among several pump manufacturers. Although the pump size required for the HWOCR plants is a significant extrapolation over current pump design and actual fabricating experience, no serious difficulties are anticipated; however, a proof-testing program appears to be desirable. As a further result of this effort, it has become apparent that due to pump size and shaft speed considerations, pump driver selection will be an unusually important phase of this activity.

The coolant purification methods process investigations, evaluations, and preliminary conceptual designs which have been prepared constitute a foundation upon which further effort will be expended to lead to the achievement of an optimized integrated conceptual purification system design. The effort to date on the optimization and design of the heavy water process systems has indicated that utilization of moderator heat to raise the temperature of boiler feedwater is the most economical way of rejecting this heat; information on materials of construction and practical operating experience was obtained from existing heavy water systems. The initial efforts on auxiliary process systems activities have resulted in the tentative selection of system concepts and in their initial development.

Plant structures and equipment arrangement designs and economic investigations have established initial bases which will be further developed and later utilized when decisions are made on the degree of containment.

Electrical design studies have assisted in the preliminary evaluation of main coolant pump drives. In addition, the plant electrical hazardous classification study has indicated tentative special equipment requirements and the desirability of employing certain measures to minimize these requirements.

The results of the large organic components test loop effort have indicated that an economic benefit would very probably result from proof-testing certain large organic components prior to their installation in the HWOCR plant. The test facility conceptual design package has provided the basis for a cost estimate and will serve as a foundation for detailed design and construction when these are authorized.

The physics investigations of the UC rod and annular-type fuel elements have progressed satisfactorily. The test physics case comparison has indicated that

the methods in use by the various contractors and ORNL yield substantially the same results.

The results of the initial safety studies effort will, by reference to the list of postulated accidents, provide both a basis for future safeguards analysis work and a guide for the safety aspects of plant design. The containment studies which were inaugurated will form a basis for further investigation in this area.

IV. NEXT REPORT PERIOD ACTIVITIES

The studies initiated during this period will be continued, and the conceptual design of the 1000-Mwe plant will continue to evolve. The optimization studies will be continued to establish optimum reactor plant design conditions. The conceptual design of the main heat transfer system will be developed based on these design conditions.

Effort will be continued on steam cycle analysis relating to the single 1000-Mwe turbine plant, on part-load plant operation, and on the main heat transfer loop optimization code involving the once-through steam generator. The incorporation of steam tables into the coolant loop package and the plant economics subroutine into the steam loop package are both scheduled. Plans will also be made to work the two packages together.

With respect to steam generator studies, evaluation efforts will continue, with emphasis on cost analysis. The cost penalties of instrumentation and control and startup requirements for the once-through steam generator will be investigated. The condensate polishing system costs for this type of generator will also be determined. As main heat transfer system design progresses and variables such as number of loops, allowable steam generator pressure drop,

reactor temperatures, etc. become better defined and overall costs are accumulated, selection of the optimum steam generator for the HWO CR will be made.

As the hydraulic requirements of the main heat transfer system become more firmly established, efforts will be continued to secure from qualified pump manufacturers additional recommendations for the design of the coolant circulating pumps. Investigations toward determining driver selection will be continued; the economics of electric motor drives and steam turbine drives will be further evaluated. The design of a stand for testing pump shaft seals will be started.

The investigations, evaluations, and preliminary conceptual designs of the coolant purification systems will be continued, with emphasis on obtaining firmer economic data for the hydrocracking process and on the elimination or reduction of the parameter uncertainties of this process. Effort on the moderator systems during the next report period will be devoted to further conceptual design work on the moderator cooling system, with emphasis on off-normal operation and on the D₂O recovery system.

The conceptual designs of the auxiliary process systems will be continued.

Drawings of the 1000-Mwe HWO CR reactor building outline, heat transfer system piping and equipment, fuel handling system, and general plant arrangement will be continued and developed in greater detail. They will be consistent with the requirements of the plant, as generated by the plant systems designs, and the conceptual scope of this activity. Electrical systems design will be continued also. Among the areas which will be covered are protective relaying for the main generator, transformers, station busses, and emergency power.

The conceptual design of the large organic components test loop has been completed and no further effort is planned until the detailed design of this loop is authorized.

In the UC fuel parametric evaluations, fuel cycle costs and heat transfer work will be continued for the various cases under study.

Safety studies will be continued and concentrated on containment criteria and requirements in order to enable early tentative decisions regarding required extent or degree of containment, thus allowing the affected plant designs to proceed.

FIG 0950-1. TYPICAL PRIMARY COOLANT LOOP

FIG 0950-2. EQUIPMENT COSTS vs REACTOR ΔT FOR CONSTANT REACTOR OUTLET TEMPERATURE, REACTOR PRESSURE DROP, AND STEAM PRESSURE

FIG 0950-3. TYPICAL COMPUTER CRT GRAPH

FIG 0950-4. TYPICAL TEMPERATURE - ENTHALPY DIAGRAM FOR STEAM GENERATOR

FIG 0950-5. STEAM GENERATOR PARAMETERS vs REACTOR ΔT FOR CONSTANT REACTOR OUTLET TEMPERATURE AND STEAM PRESSURE

FIG 0950-6. EFFECT OF STEAM PRESSURE ON PRIMARY HEAT TRANSFER SYSTEM

PUMP SIZE	NOZZLE SIZE		A	B	C	D	E	F	G	APPROX WGT-LBS	SHAFT SPEED RPM	DESIGN CAPACITY GPM	DESIGN HEAD FT
	SUCTION	DISCHARGE											
24 x 24 x 34	24	24	59	59	118	32	21	45	42	35,000	1800	44,500	960
26 x 28 x 40	28	26	70	70	140	38	25	53	51	56,000	1500	63,500	960
36 x 36 x 48	36	36	83	83	166	45	30	63	60	95,000	1260	89,000	960

NOTE: NPSH FOR ALL PUMPS = 110 FT

FIG 0950-6A. PRELIMINARY DESIGN DATA

FIG 0950-7. STEAM GENERATOR THERMAL POWER (100% LOAD)

FIG 0950-8. STEAM CYCLE HEAT RATES (100% LOAD)

FIG 0950-9. TOTAL TURBINE PLANT COST (500 Mwe)

FIG 0950-10. D₂O FEEDWATER HEATING - STEAM CYCLE HEAT RATES (100% LOAD)

FIG 0950-11. D₂O FEEDWATER HEATING - IMPROVEMENT IN NET CYCLE
HEAT RATE AND EFFICIENCY (100% LOAD)

FIG 0950-12. D₂O FEEDWATER HEATING - JUSTIFIED INCREMENTAL INVESTMENT

FIG 0950-13. MODERATOR OUTLET TEMPERATURE vs ANNUAL COST

FIG 0950-14. MODERATOR OUTLET TEMPERATURE vs ANNUAL COST

FIG 0950-15. MODERATOR OUTLET TEMPERATURE vs ANNUAL COST

FIG 0950-16. MODERATOR OUTLET TEMPERATURE vs ANNUAL COST

FIG 0950-17. 250 FT REACTOR CONTAINMENT SPHERE - PLANT ARRANGEMENT
PLAN AND ELEVATION (DRAWING NO. R-HWR3-GA001)

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A FULL LINE OF NORTH AMERICAN ATOMICS, INC. 10000 BAYVIEW AVENUE, SUITE 100, BAYVIEW, MICHIGAN 48064			
MAIN ORGANIC COOLANT SYSTEM - PLANT ARR-T			
STEAM GEN (LARGE, U-SHAPED) - PLAN & ELEV.			
1000 MWDC			
HWOOD			
DESIGN	REV. 2.00	DATE	11/70
CHECKED		BY	
APPROVED		BY	
R-BWR2-GA002			

FIG 0950-18. MAIN ORGANIC COOLANT SYSTEM - PLANT ARRANGEMENT - STEAM GENERATOR (LARGE, U-SHAPED) - PLAN AND ELEVATION (DRAWING No. R-HWR2-GA002)

FIG 0950-19. MAIN ORGANIC COOLANT SYSTEM - PLANT ARRANGEMENT - STEAM GENERATOR (MODULAR) - PLAN AND ELEVATION (DRAWING No. R-HWR2-GA003)

FIG 0950-20. REACTOR BUILDING - PLANT ARRANGEMENT - ORGANIC LOOP, NONCONTAINED - PLAN AND ELEVATION (R-HWR2-GA004)

FIG 0950-21. CONTAINMENT VESSEL
130' DIA x 150' HIGH
(NO WALL SHIELDING)

FIG 0950-22 CONTAINMENT VESSEL
150' DIA x 150' HIGH
(NO WALL SHIELDING)

FIG 0950-23 CONTAINMENT VESSEL
250' DIA x 150' HIGH
(NO WALL SHIELDING)

FIG 0950-24 CONTAINMENT VESSEL
130' DIA x 150' HIGH
(WALLS SHIELDED)

FIG 0950-25 CONTAINMENT VESSEL
150' DIA x 150' HIGH
(WALLS SHIELDED)

FIG 0950-26 CONTAINMENT VESSEL
250' DIA x 150' HIGH
(WALLS SHIELDED)

FIG 0950-27. ORGANIC FLOW DIAGRAM LARGE COMPONENT TEST LOOP

FIG 0950-28. HWO CR REFERENCE ANNULAR FUEL ELEMENT

300-MWE PLANT ENGINEERING

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300-Mwe Plant Engineering	1050
I. Project Objectives	1050-1
II. Progress During Report Period	1050-1
III. Evaluation of Effort to Date	1050-3
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HWO CR QUARTERLY REPORT	
TITLE: 300-Mwe Power Plant Engineering	SUBTASK NO.: 1010
PROJECT ENGINEER: G. A. Anderson	REPORTING PERIOD: Inception through June 1965

I. PROJECT OBJECTIVES

The technological and economic feasibility of advanced nuclear power plant concepts can only be determined through the construction and operation of demonstration plants. Therefore, the objective of this task is to design and construct such a plant to demonstrate the essential features of the larger 1000-Mwe plant. Design of the 300-Mwe plant will be accomplished by scaling down from the 1000-Mwe plant conceptual design.

II. PROGRESS DURING REPORT PERIOD

The primary efforts during this report period were directed to the design and analysis of the 1000-Mwe plant, and the identification of plant characteristics and construction problems for the 1000-Mwe plant is directly applicable to the 300-Mwe plant. See the task 0910 report for progress to date. A preliminary budgetary construction cost estimate was prepared for the 300-Mwe plant based on previous work reported in CEND-175.

III. EVALUATION OF EFFORT TO DATE

Efforts to date have been only nominal.

IV. NEXT REPORT PERIOD ACTIVITIES

As more detailed information concerning the 1000-Mwe plant design becomes available, efforts on the conceptual design of the 300-Mwe plant will be increased. Reactor layout studies will be initiated, including the end shields, calandria, flow tube, headers, shielding, refueling machines and reactivity control. The flow tube and fuel element designs will be prototypical of the 1000-Mwe plant. Specific safety studies will be initiated in support of the conceptual design, which is to be completed during 1966.

HWO CR QUARTERLY REPORT	
TITLE: 300-Mwe Plant Engineering	SUBTASK NO.: 1050
PROJECT ENGINEER: J. Jacobson	REPORTING PERIOD: Inception through June 1965

I. PROJECT OBJECTIVES

The objective of this task is to provide the preliminary reference design (conceptual) and budgetary cost estimate for a 300-Mwe HWO CR power plant which will serve to demonstrate the essential features of a 1000-Mwe plant and support extrapolations to larger plants.

II. PROGRESS DURING REPORT PERIOD

Task effort was initiated on a limited basis pending the availability of additional design data from the 1000-Mwe plant initial studies.

In the area of process systems, primary emphasis was placed on preliminary studies and systems optimizations for the 1000-Mwe plant. However, as the initial concepts of these systems evolved, some effort was expended in the application of these results to the 300-Mwe demonstration plant. Also, preliminary data were prepared to establish the plant thermal requirements and heat rates. Preliminary process flow diagrams and piping and instrumentation diagrams were prepared for a number of the organic, heavy water, waste, cooling water, and steam systems. In addition, preliminary write-ups describing these systems and their operation were started.

Other principal targets of the 300-Mwe plant effort during this period were in the area of facilities design. Sketches and layouts were made of the

300-Mwe HWOCR plant in the following partial containment arrangement.

a) The reactor complex and moderator cooling system were placed in a cylindrical containment vessel 100 ft in diameter by 170 ft high, having a hemispherical head and flat bottom.

b) The heat transfer system, consisting of collection manifolds feeding three loops (each containing one pump, one filter, one vertical steam generator, and one vertical superheater) was housed in a conventional building with full crane coverage. The organic service systems were placed in this building, also.

c) The moderator service facilities, spent fuel storage pool, and fuel handling complex were placed in a controlled access building.

A conceptual site arrangement study was made comparing a plant in which the various major activities are integrated into a common building surrounding the reactor containment, and a plant in which the major activities are essentially in separate structures. The latter concept appears preferable, based on greater accessibility for construction and maintenance.

An investigation was made of the requirements of the large coolant pump drive motors. Basis of the investigation was the premise that the pumps and drives would be similar in size, type, and rating to the units for the 1000-Mwe plant. The studies made for the larger plant had indicated that for power requirements in the range required for this plant (greater than approximately 10,000 hp), steam turbine drives should be used in preference to electric motors. It was tentatively decided that this philosophy would also be used in the 300-Mwe plant.

An emergency power study indicated that the amount of emergency power that must be available in a 300-Mwe plant is such that it should be delivered

at 4 KV. Also, the size of some items of equipment is such that their operation from an emergency battery source is of doubtful feasibility; operation directly from the emergency AC system is preferable. This increases the importance of the reliability of the emergency electrical generator. For these reasons, use of a gas-turbine-driven generator is being considered over the more conventional diesel-powered unit.

A study was made of the problems relating to the location of electrical equipment in areas that may be hazardous due to the presence of explosive vapors or ducts. As a result of this study, it appears prudent to devote more than normal attention to the layout of process equipment, the configuration of various plant areas, and the design of the ventilation systems in order to minimize the requirements for explosion-proof electrical equipment.

IV. EVALUATION OF EFFORT TO DATE

Utilization of the results of preliminary efforts in the 1000-Mwe plant design has provided initial process systems designs and a first representation of the size, configuration, and layout of the 300-Mwe plant.

V. NEXT REPORT PERIOD ACTIVITIES

With increasing availability of information from the 1000-Mwe plant studies, effort toward achievement of the objective of this task (conceptual design of the 300-Mwe plant) will receive greater emphasis. The design of the process systems for this plant will be continued and accompanied by the required supporting efforts in the areas of instrumentation and controls. Facilities designs will be continued on firmer bases as the process systems designs evolve further. Safety studies specifically directed toward the 300-Mwe plant will be initiated.

CANADIAN OPERATIONS

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ATOMICS INTERNATIONAL

COMBUSTION ENGINEERING, INC.

HWO CR QUARTERLY REPORT	
TITLE: Canadian Operations	SUBTASK NO.: 2050
PROJECT ENGINEER: C. A. Trilling	REPORTING PERIOD: Inception through June 1965

I. PROJECT OBJECTIVES

The objectives of this task are: (1) provide technical data feedback to the HWO CR Program on the first operational experience with an organic-cooled heavy-water-moderated power reactor (WR-1), and provide technical assistance to AECL in the commissioning and startup of this reactor; (2) provide the planning, coordination and expediting associated with the HWO CR Program fuel irradiations in the WR-1; (3) provide technical support, advice and continuous review of the HWO CR fuel irradiation experiments in the U-3 test loop at Chalk River; and (4) provide the HWO CR Program with current information on the initial testing, startup and operating experience of the CANDU nuclear power plant at Douglas Point.

II. PROGRESS DURING REPORT PERIOD

Residence assignments of HWO CR personnel at WR-1 (Pinawa) and the U-3 loop (Chalk River) were made. The initial flow of technical data of value to several tasks of the HWO CR Program began. Technical coordination and assistance were provided in the preparation, handling and field assembly of HWO CR fuel irradiations for the U-3 loop. Planning was completed for further assignments early in FY 66; at WR-1, in the area of fuel materials; at Chalk River, in connection with organic coolant chemistry of the U-3 loops; and at the CANDU reactor site at Douglas Point.

Following some initial familiarization visits, D. P. Cary assumed his duties at the WNRE site in April, providing technical assistance to AECL and information to the HWO CR project on the operations at WR-1. Information was provided on the commissioning status of various reactor systems and the test loop capabilities. Procedures were established for the flow of materials, equipment and data between AI-CE and WNRE. In late June, J. E. Mills was also assigned to WNRE to provide technical assistance in the area of plant maintenance during WR-1 commissioning, particularly the mechanical engineering aspects, including pumps and valves. Both Messrs. Cary and Mills are drawing on their experience gained at the Piqua Nuclear Power Facility in their work at WNRE. Plans were completed for the assignment of a senior materials representative to arrive on site in August.

B. J. Thomas was assigned as AI-CE Irradiation Experiments Representative at Chalk River in June, following temporary assignments there in April and May. Information was provided the HWO CR project on the Canadian fuel irradiation work (X-721 and 726), as well as data on their organic coolant technology effort in the E-2 loop of NRX. Information on the AI-CE, U-305 fuel irradiations was presented to the AECL Reactor Irradiation Committee to obtain that group's approval for the forthcoming HWO CR irradiations. Procedures and techniques for expediting the shipment of these experiments were developed. Plans were completed for the assignment of an AI-CE organic coolant chemistry representative and supporting technician to Chalk River early next quarter.

III. EVALUATION OF EFFORT TO DATE

The information and assistance to the HWOOCR Program provided by the on-site representatives in Canada has been extremely valuable during the initial phases of the HWOOCR Program. The establishment of communication channels, flow of technical documents, drawings and verbal reports has provided guidance in many project areas. Assistance in meeting the tight schedules associated with the fuel irradiations in the U-3 loops at Chalk River was particularly useful. As the WR-1 reactor approaches criticality, the cross-flow of data is expected to become increasingly valuable to HWOOCR design, development and test work.

IV. NEXT REPORT PERIOD ACTIVITIES

C. C. Woolsey will assume his duties as HWOOCR Materials Representative at WR-1 in August. Mr. R. J. Sullivan will be in residence at Chalk River in August to provide support, advice and continuous review in the area of organic coolant chemistry. He will be supported by the assignment of an AI-CE chemical technician. Mr. W. B. Wolfe will be assigned to Douglas Point in mid-September to provide informational feedback to the HWOOCR Program on the testing, startup and operation of the CANDU reactor. The above additions to the already functioning HWOOCR Program representatives in Canada should provide greater depth of technical experience and increased cross-flow of data of direct use to the project during the next quarter.