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HEAVY WATER ORGANIC COOLED REACTOR

STATUS AND POTENTIAL

JUNE 1967

AEC RESEARCH AND DEVELOPMENT REPORT

ATOMICS INTERNATIONAL—COMBUSTION ENGINEERING

A Joint Venture for Heavy Water Organic Cooled Reactors

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ATOMICS INTERNATIONAL-COMBUSTION ENGINEERING
A Joint Venture for Heavy Water Organic Cooled Reactors

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ABSTRACT

This report summarizes the technical status of the Heavy Water Organic Cooled Reactor (HWOCR) concept at the conclusion of the development program conducted for the USAEC by Atomics International - Combustion Engineering from January 1965 until March 1967.

The objective of this program was to develop the HWOCR concept to the point of commercial feasibility for central station power generation. The HWOCR was to be an advanced converter reactor with the following capabilities:

- 1) Generation of electric power at competitive cost.
- 2) High utilization of natural resources of fissile materials. This was to be accomplished by minimizing the requirements for uranium consumption and maximizing the plutonium production for fast breeder use. The HWOCR would also be flexible enough to operate economically on the thorium- U^{233} fuel cycle if that fuel cycle were to become of major national interest.
- 3) Relatively simple extrapolation to the very large plant sizes required for both the economic generation of electric power and the economic production of water for domestic and agricultural use.

While additional development work remains to be performed at the conclusion of the program, all the information available indicates that the HWOCR concept is technically feasible and capable of meeting its objectives.

This report is divided into three sections describing (1) the concept, (2) the plant design, economics, and fuel utilization, and (3) the technical status and results of the development program. It is intended to provide a concise and convenient summary of the work accomplished, without including the detailed information presented in the quarterly progress and topical reports and in the extensive technical information documents published at the conclusion of the program.

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I. HWO CR CONCEPT

A. ROLE OF THE HWR IN AN EXPANDING NUCLEAR ECONOMY

1. Introduction

The Heavy Water Reactor (HWR) is an advanced converter eminently well suited to complement and support the fast breeder reactor during the transition period from its inception to the time, almost half a century later, when a fully self-sustaining fast breeder nuclear power economy can be established. This period will be one of tremendous growth of our electric power requirements and even greater relative growth of our nuclear power generating capacity. It will strain to the utmost the availability at reasonable cost of our natural resources of fissile and fertile fuel materials. Unless requirements can be met economically, the utilities will turn back to the use of fossil fuels, and the full potential for the development of nuclear power in this country will not be realized.

The current growth trend of our nuclear power capacity leads to projections of installed capacity in 1980 which almost double the predictions made in 1962. The USAEC has estimated domestic uranium ore resources at 200,000 to 525,000 tons of U_3O_8 available at less than \$10/lb.⁽¹⁾ Current projections of the United States nuclear power generating capacity by 1980 show a cumulative consumption of over 200,000 tons of ore by that year. It is therefore evident that maximum utilization of our resources of fissile materials must be an immediate and urgent goal of the U.S. nuclear power development program.

On a long-term basis this goal can only be achieved through the development and widespread use of reliable and economically competitive high-gain breeder reactors which can sustain the rapidly growing nuclear economy without relying on an external supply of fissile material. These reactors will produce enough fissile material beyond their own requirements to provide fuel inventories for new plants. However, there are some uncertainties connected with the effectiveness of breeder reactors in limiting the uranium requirements of the expanding nuclear industry to an economically acceptable level. These uncertainties involve:

1) The performance (specific inventory and breeding gain) of the breeders which will be offered to utilities. Breeders having high inventories and low breeding gains have long doubling times and, therefore, cannot have a significant effect on the uranium supply problem.

2) The characteristics of the converters which initially will supply the bulk of the plutonium fuel for breeder inventories. These will determine, to a large degree, the rate at which new breeder capacity can be introduced based on the availability of converter-produced plutonium fuel.

3) The schedule on which the engineering problems connected with the breeder can be resolved; i. e., when it will be adequately demonstrated with respect to safety, reliability, and costs, so as to induce full commercial exploitation.

The effect of these uncertainties on the total uranium requirement to supply a rapidly growing nuclear industry is quite significant. At one extreme, they threaten the viability of the industry within the lifetime of plants now under construction. If resolved favorably, they permit the use of a virtually inexhaustible supply of low-cost energy.

The motivation for undertaking the development of the class of reactors known as "advanced converters" has been the desire to conserve the available uranium resources through the widespread use of more efficient nuclear systems, thereby reducing the effect of delays in high-performance breeder development and possibly minimizing the effect of the other uncertainties as well.

Among the several advanced converter concepts which have been seriously considered for development, only the heavy water reactor has the potential performance characteristics and inherent flexibility to accomplish these goals effectively. This is largely due to its unique ability to complement a breeder economy by providing the required supply of plutonium.

This section of the report presents the results of studies comparing the effectiveness of the several feasible alternative solutions of the ore depletion problem. These studies quantitatively establish the advantages of the heavy water concept in meeting a wide variety of potentialities.

2. Ore Conservation in Mixed Reactor Systems

The most meaningful comparison of the effectiveness of various nuclear reactor systems in reducing the requirements for mined uranium must be made on the basis of ore requirements as a function of time in a variety of potential

future situations. Comparison on the simple basis of fuel requirements for a single plant can be misleading since it does not take into account the important effect of inventory requirements in an expanding industry nor the interaction between the various reactor types.

The studies reported on in this section calculated the uranium requirements for a variety of assumed mixed systems involving several reactor concepts. While the simplified economies studied are not intended to represent actual situations, they do give quantitative insight into limiting situations and thereby allow meaningful comparison of the potential effects of various reactor systems on the economy.

It has been pointed out that only the widespread application of breeder reactors to fill the need for new generating capacity can eventually place an upper limit on the total requirement for mined uranium. The quantity of uranium ultimately required would continue to rise indefinitely without breeders and, with breeders, would be largely determined by breeder performance, timing, rate of introduction, and by the type of converter reactors in widespread use during the 30- to 40-year transition period to a self-sustaining breeder economy.

If economically attractive high-performance breeders could not be developed, or if their development were delayed by many years, the depletion problem could be minimized by extensive use of advanced converters based on nearer term technology than the breeders and higher efficiency in their utilization of fissile material than the current light water reactors.

Of the many proposed advanced converter concepts, three have received serious attention by the AEC, the utility industry, and the reactor suppliers. These are:

- 1) The Heavy Water Reactor (HWR) involving a pressure tube design to allow the use of low-temperature heavy water for efficient neutron moderation along with a variety of coolants. This concept, using organic coolant, has been the subject of development in the U.S., Canada, and Europe. It offers a high degree of flexibility with respect to the type of fuel cycle used, as it can perform well with either enriched or natural uranium, or with thorium.

- 2) The High Temperature Gas-Cooled Reactor (HTGR) is a graphite-moderated, helium-cooled system using the thorium- U^{233} fuel cycle. It is initially fueled with enriched U^{235} but once it has reached equilibrium (about 7 years after startup), the primary fuel is recycled U^{233} with small additions of U^{235} .

3) The Light Water Breeder (LWB), formerly the Seed-and-Blanket Reactor (SBR), is also a thorium- U^{233} system having a very high conversion ratio which, however, falls short of actual breeding in economic designs. It is a concept based on light water reactor technology and, therefore, its development represents a minimum departure from proven technology.

Both of the thorium cycle converters (HTGR and LWB) are characterized by high conversion ratios and require much less uranium for burnup than the current light water reactors. However, they fall short of breeding and thus require some uranium makeup and significant amounts of uranium for new plant inventory. Extensive use of these systems would delay the depletion of low-cost uranium reserves, but ultimately would require much more uranium than any system other than the one based on the sole use of light water reactors. This is because the thorium cycle plants produce no plutonium or net surplus of fissionable material to fuel the breeders, which are the only reactors that can ultimately limit the requirement for uranium.

In the unlikely event that economic high-performance breeders could not be developed at an early date and national emphasis should shift to the thorium cycle, the heavy water reactor can achieve as efficient a fuel utilization using thorium fuels as its solid-fuel advanced converter competitors. The inherent flexibility with regard to choice of fuels in a heavy water system allows it to progress to thorium utilization by way of an interim uranium-plutonium cycle based on natural or low enrichment uranium, whereas the other converters must be supplied with highly enriched U^{235} as the starting fuel. This initial inventory may represent a severe economic penalty once low-cost uranium reserves have been depleted.

A comparison of cumulative uranium requirements for several mixed reactor economies is presented in Figure 1. These curves are based on analyses which assume the growth of the nuclear industry projected by the AEC in the 1967 Supplement to the 1962 Report to the President⁽¹⁾ (80,000 to 110,000-Mwe capacity by 1980), commercial acceptance of advanced converters by 1980 and high-performance breeders by 1985, and breeder construction limited by plutonium availability. The data on projected ore prices indicated in this figure are taken from the AEC's estimates in Reference 1. The curve showing the uranium ore requirements if all new nuclear capacity were made up of light

Figure 1. Ore Requirements of Typical Reactor Systems

water reactors of current design clearly shows the need for advanced systems in order to maintain a viable nuclear industry through the turn of the century. The other curves show the ore requirements in the limiting cases in which all new converter capacity after 1980 is assumed to be of the type indicated and high-performance breeders are introduced as rapidly as possible using the available plutonium. The effect on uranium requirements as a result of the type of converter used is obvious. Widespread use after 1980 of heavy water reactors fueled with natural uranium can significantly reduce the requirement for mined uranium. If heavy water reactors are used in place of light water reactors, the potential savings are about 500,000 tons of U_3O_8 , representing a value of approximately \$24 billion. In association with a lower performance breeder (doubling time of 10 instead of 7 years), these savings would amount to 1.6 million tons of U_3O_8 , representing a value of over \$100 billion.⁽²⁾

Compared to the light water reactor economy, the thorium cycle converter (HTGR, LWB, and also HWR) economy tends to stretch out uranium reserves 5 to 10 years due to its low make-up requirements and high conversion ratio,

but ultimately leads to unlimited ore requirements because it produces no plutonium for breeder startup. This is the fundamental difference between the heavy water converter and all the other types which are being considered.

There is one thorium cycle system, however, which, if used extensively, can effectively limit future maximum ore requirements. This is the Molten Salt Breeder Reactor (MSBR). The MSBR uses the thorium- U^{233} fuel cycle, but by virtue of continuously recirculating molten fuel is able to achieve a small breeding gain while operating in the thermal spectrum. This is possible through the elimination of core structural material and the continuous removal of fission product contamination from the fuel stream. The breeding gain of the MSBR is adequate to achieve short doubling times because of the very low specific inventory requirements of this fluid fuel system. The MSBR, however, has as yet received relatively little support from the utility industry, presumably because of its requirement for remote maintenance of all parts of its primary loop.

3. Plutonium Availability Constraint upon Breeder Operation

The long delay between the time breeders are first introduced and the time at which ore requirements peak out (30 to 40 years in Figure 1) results from the limitation upon the rate of breeder capacity growth due to the availability of fissile plutonium for inventory. Uranium-235 can be used to fuel fast reactors, but plutonium is a much better fuel because it requires a lower fissile inventory and improves the breeding gain due to its higher fission neutron yield in a fast spectrum. The lower fissile inventory and improved breeding gain associated with the use of plutonium are reflected in significantly lower fuel cycle costs. This economic advantage permits the plutonium-fueled system to achieve competitive energy costs and utility acceptance at an earlier date. The cost penalty associated with the use of U^{235} to fuel the breeder has been estimated to be of the order of 0.5 mill/kwh. This would discourage and delay widespread use of this approach.

A comparison of the effectiveness of the possible alternative routes for supplying fissile inventory for breeder reactors is summarized in Table 1. The data in this table clearly indicate the improvement offered by the heavy water reactor on the basis of natural uranium consumed. The basic reason for the economic disadvantage of using U^{235} in a fast reactor as a source of plutonium is evident from the data shown in the gaseous diffusion plant column of the table.

TABLE 1
COMPARISON OF PLUTONIUM PRODUCING REACTORS WITH
ISOTOPE SEPARATION AS A SOURCE OF INVENTORY
FOR FAST BREEDER REACTORS

System	LWR	HWR*	Gaseous Diffusion Plant
Feed Material	Enriched U	Natural U	Natural U
Fissile Product	$\text{Pu}^{239} + \text{Pu}^{241}$	$\text{Pu}^{239} + \text{Pu}^{241}$	U^{235}
kg Fissile Product/ Metric Ton of Natural U Feed	1.3	2.8 - 3.5	4.6
Fast Breeder Capacity Inventoried/kg of Fissile Product (kwe)	330	330	240
Fast Breeder Capacity Inventoried/Metric Ton of Natural U Feed (kwe)	430	930 - 1160	1100
Energy Produced/Metric Ton of Natural U Feed (kwhe)	$\sim 50 \times 10^6$	$\sim 70 \times 10^6$	$(\sim 3 \times 10^6)$ (consumed)

*The higher values are those obtained with the natural uranium metal fueled HWO CR

The path from thermal converters using uranium fuel to the plutonium-fueled fast breeder (as opposed to investment in the thorium cycle) is the obvious one, and the one upon which the U. S. and most other countries with nuclear power programs are now embarked. Only in the case of the HWR can a shift be made from light water reactors to advanced converters without impeding the introduction of breeders. The only source of plutonium for breeder inventories are converters utilizing the uranium-plutonium fuel cycle, i. e., light water reactors and heavy water reactors. The thorium cycle concepts, as mentioned earlier, do not produce plutonium and, therefore, to the extent to which they are used to fill requirements for new capacity, will actually restrain the breeder growth rate. It is for this reason that the HWR alone, among the current advanced converter candidates, can be useful for several decades beyond the time breeders become available.

4. Importance of Breeder Performance

The effect on uranium requirements of the several uncertainties with respect to the breeder performance ultimately attained and of the characteristics of the converters used to complement the breeders is dramatically illustrated in Figure 2. Here the *total* ore requirements for several economies involving converters and breeders are plotted as a function of the breeder introduction time with breeder doubling time as a parameter. This analysis assumes that all new converter capacity, after 1975, is either LWR or HWR and that breeder additions are limited by plutonium availability.

An interesting point brought out by analysis of balanced economies is the fact that it is not very important, in terms of total ore requirements, *when* breeders become available. What *is* important is the type of converter in use with the breeder and the characteristics of the breeder itself, as illustrated in Figure 2. In each case the increase in ore requirement caused by delayed introduction of the breeder is seen to be rather small before 1985. Significant savings are noted, however, in the cases in which a better converter (HWR) or a better breeder is assumed. The insurance provided by the HWR against

Figure 2. Total Ore Requirement as a Function of Year of Breeder Introduction

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uncertainties in breeder performance and timing is obvious. The total uncertainty band with the HWR is well below the region of very high ore prices while with LWR's alone, the low cost reserves could easily be depleted if the best breeder performance were limited to doubling times of 10 years or more.

Another point worth noting in conjunction with Figure 2 is the effect of plutonium storage in the case of the HWR as opposed to plutonium recycle in the case of the LWR. The fact that plutonium from the HWR can be stored with negligible economic penalty, until the demand for breeder inventory raises its value enough to justify recovery, results in a significant backlog in the supply — a plutonium "mine" in a manner of speaking. This buildup in plutonium inventory in later years allows large additions of breeder capacity and reduces the effect of breeder delays beyond 1985. In other words, the HWR economy is less sensitive to breeder timing than the light water reactor economy. The natural uranium version of the HWR actually produces more plutonium for export per unit energy generated than does a good breeder ($\sim 45 \text{ kg}/10^9 \text{ kwh}$ vs $\sim 40 \text{ kg}/10^9 \text{ kwh}$). If the economy were dominated by HWR plants fueled with natural uranium, it would therefore be more expedient, in terms of ore depletion, to postpone large-scale breeder construction until rather short (< 7 years) doubling times are assured.

5. Summary

Continued dependence on light water reactors alone to fill the projected demand for nuclear power additions into the next century can be expected to seriously deplete the reserves of low-cost uranium ore based on prudent assumptions regarding the extent of available reserves. Such depletion will naturally lead to significant increases in the cost of nuclear fuel and threaten the viability of the nuclear industry.

Examination of the available alternatives has shown that:

- 1) The light water reactors are relatively wasteful with respect to the utilization of uranium.
- 2) The projected growth of the nuclear industry is much too rapid to allow even the most optimistic breeder reactor economy to be self-sustaining in terms of inventory requirements during this century.

3) Because of (2) above, converter reactors will have to be built and operated for many years to satisfy the industry's projected growth. During this period they cannot possibly be replaced completely by breeders no matter how soon the breeders become available nor how optimistic their performance.

4) The use of thorium cycle converters can stretch out but not guarantee the adequacy of a given supply of uranium. This is due to the fact that these systems do not generate plutonium for breeder inventories.

5) The use of heavy water reactors can significantly reduce the drain on our resources while facilitating the growth of breeder capacity.

6) Heavy water reactors, due to their more efficient utilization of uranium, can sustain relatively low energy costs in the face of increasing uranium ore prices.

7) The use of heavy water reactors will assure for the breeders a relatively cheap source of plutonium.

8) The inherent flexibility of the heavy water reactor with respect to fuel choice can assure the applicability of this system even in the unlikely event that fast breeders could not be developed and our national interest should shift to the thorium cycle and thermal breeders.

B. SELECTION OF ORGANIC COOLING (HWOCR)

Most of the heavy water moderated reactor concepts have been based on the use of a pressure tube reactor design in which the coolant and moderator are kept separate from each other and operate under different conditions of temperature and pressure. This separation allows the utilization of the thermal advantages of a high-temperature coolant and the good neutron economy of a low-temperature moderator. It lends itself to the use of a variety of coolants without interfering with the advantages of heavy water moderation.

Extensive studies have been conducted in many countries to optimize the coolant selection for the pressure tube heavy water moderated reactor. The coolants considered have been pressurized and boiling D_2O , pressurized and boiling H_2O , steam, organic, and gases. The most recent evaluations were performed by AECL^(3,4) and the USAEC^(5,6) in 1963 and 1964. Both concluded that of the various coolants studied, boiling light water and organic showed the most attractive potential economic performance with a high probability of successful technical development. There appeared to be no overriding reason for selecting one or the other of these coolants, with both requiring an appreciable amount of technical development work and showing approximately equal potential

energy generation costs. AECL based its decision to go ahead with boiling light water cooling on its ability to relate the required development program closely to the technology already existing on heavy water. They also proceeded with the construction and operation of the organic cooled WR-1 test reactor at Whiteshell, Manitoba, thus maintaining their capability in organic coolant technology. The USAEC in 1964 decided to proceed with organic cooling, partly on the basis of existing U.S. organic technology, and partly to complement AECL's work on the boiling light water concept and thus provide for the establishment of the technology required for the commercial utilization of the two heavy water reactor concepts showing the greatest potential of technical and economic success. As stated by AEC Commissioner J. T. Ramey to the Canadian Nuclear Association in May 1965, "Both the advanced boiling light water and organic concepts are being developed vigorously, and both technologies will be available to both countries without a costly duplication of effort."⁽⁷⁾

The organic coolant selected is a commercially available mixture of terphenyls and their decomposition products (AECL is using a mixture of partially hydrogenated terphenyls in the WR-1 because of its being liquid at ambient temperature). The most important characteristics of such a coolant are its low vapor pressure at operating temperature, its chemical compatibility with most fuel and structural materials, and its lack of activation under exposure to neutron irradiation. Its drawbacks are modest heat transfer capability, decomposition under exposure to heat and radiation, tendency to form deposits on fuel element surfaces if not adequately purified, and flammability. It is to be noted that the requirement for maintenance of high coolant purity to prevent deposit formation at the surface of the fuel elements is a characteristic of the light water cooled reactors as well as of the organic cooled reactors; the principal difference resides in the techniques used for the purification of the coolant.

The use of the organic coolant in a heavy water moderated reactor provides for:

- 1) Flexibility in the selection and use of fuel materials. Absence of chemical interaction allows the use of oxide, carbide, or metal uranium or thorium fuel without risk of release of chemical energy in case of a cladding failure. The high density carbide and metal fuels are especially attractive in a reactor concept emphasizing high neutron economy.

2) Generation of superheated steam. The high operating temperature of the organic allows superheating of the steam produced by approximately 90°F at a pressure of 900 psig.

3) Use of a low pressure carbon steel primary coolant system. Corrosion of such a system by the organic coolant is negligible.

4) Accessibility to the primary coolant system of the reactor during reactor operation as long as the coolant does not contain appreciable amounts of fission products. The activity of the coolant is limited to that produced by activation of the inorganic impurities it contains.

5) Minimization of the D₂O inventory and losses since the use of the D₂O is limited to its function as moderator under conditions of low temperature and low pressure.

The effect of the decomposition and consequent make-up requirements of the organic on energy generation costs is minimized in an HWR since the amount of organic in the core of the reactor is quite small. The make-up cost can be reduced still further through recovery of usable coolant by hydrocracking of the decomposition products. Film formation can be kept at satisfactorily acceptable levels by control of the purity of the coolant. Flammability is taken care of through normal industrial safety precautions implemented both in the design and in the operation of the plant.

The selection of the organic cooled heavy water moderated reactor (HWOCR) concept for development in the U.S. was therefore based on its anticipated technical performance and economic potential subject to the success of a vigorously pursued development program.

C. DUAL PURPOSE APPLICATION

The application of nuclear energy to dual-purpose power generation and water desalting plants has recently become of major national and international interest. There are several important considerations which differentiate the most desirable energy source characteristics for dual-purpose applications from those for power-only applications. The most important of these can be listed as follows:

- 1) Feasibility of large size plants
- 2) Requirement for high plant factor
- 3) Effect of financing conditions (favoring high capital cost - low operating cost plants)

Analysis and evaluation of the HWOCR concept have shown an excellent potential of satisfying all three of these considerations. It therefore appears to be well suited for dual-purpose applications.^(8,9)

1. Feasibility of Large Size Plants

The combination of water and power in a single installation permits the consideration of much larger energy sources than would otherwise be required for power-only applications. This consideration favors those energy sources which lend themselves to relatively straightforward extrapolation to very large plant sizes and provide appreciable reductions in capital cost per unit of energy generated with increased size.

As described in Section II-A of this report, the HWOCR reactor consists of an array of pressure tubes containing the fuel. The organic coolant flows in parallel through these tubes. This arrangement is easily adaptable to subdivision of the reactor into repetitive portions or modules. The thermal power rating of the reactor can, therefore, be varied within a wide range by simply adding or deleting modules, and changing the coolant system external to the reactor accordingly. There would be no major design differences between a very large dual-purpose plant and a smaller power-only plant, and the reproduction of identical modules should provide for significant reductions with increased size in capital cost per unit of energy generated.

2. Requirement for High Plant Factor

In a dual-purpose plant the energy-consuming portion of the plant represents a much larger fraction of the total investment than in a power-only plant. This places a special premium on an energy source with high availability and low incremental energy cost, the latter being an economic requirement to justify base-load operation of the heat source at its maximum availability.

The efficient operation of a flash-type evaporator is very sensitive to slight changes in operating conditions. Consequently, the number of startups should be minimized since considerable time could be spent in balancing the evaporator plant for optimum operation. The energy source should therefore have high reliability so as to provide a steady steam supply to the water plant. Such reliability can be achieved through a preventive maintenance program

which in turn requires ready access to all critical components while the plant is on-line. The low activity of the organic coolant of the HWOCR provides this access under most operating conditions.

The pressure tube design and the low coolant vapor pressure make the HWOCR particularly adaptable to on-line refueling. With a strong maintenance program and on-line refueling, the overall availability factor of the plant should be limited only by the requirements of the energy-consuming portions of the plant such as turbine overhaul and evaporator maintenance. In addition to providing for high availability, on-line refueling minimizes the reactivity requirement and therefore the inventory of fuel in the reactor core, thus lowering the fuel cycle costs.

In addition to the high availability requirement of the plant over the short-term period, the high capital investment associated with a water plant requires the energy plant to be extensively base-loaded over its full lifetime. To justify loading the plant in this manner, its incremental energy cost must be at least as low as that of any other power plant on the grid. In this regard, it is desirable that the energy source have a high degree of fueling flexibility with regard to fuel type and form, to take full advantage of improved technology and changing market prices for uranium and plutonium.

With its pressure tube design, on-line refueling capability, excellent neutron economy, simplicity of fuel geometry, and compatibility of the organic coolant with most fuel materials, the HWOCR offers high flexibility of fuel choices and optimization of fuel management. Its fuel cycle can be adapted to meet the economic conditions of any time period during its operation and thus provide for base-load operation over a major part of its lifetime.

3. Effect of Financing Conditions

The portion of the energy plant related to water production will normally carry much lower fixed charge rates than large power-only plants. The investor-owned utility power plants have fixed charge rates which range from 10 to 14%. The fixed charge rates for water plants range from 4 to 7% in the large public water systems. This favors a high capital cost - low fuel cycle cost energy source such as the HWOCR.

On the basis of adaptability to extrapolation to very large plant sizes, high anticipated plant factor over operating lifetime, low fuel costs and excellent fuel selection flexibility, the HWO CR shows an excellent potential for dual-purpose applications.

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II. HWOCR PLANT DESCRIPTION AND EVALUATION

A. DESIGN DESCRIPTION

1. Introduction

The preliminary conceptual design of the reference 1000-Mwe HWOCR central station plant, AI-CE-Memo-6,⁽¹⁰⁾ was completed in October 1965. This design is based on the use of rod-type, SAP-clad, enriched hyperstoichiometric uranium carbide fuel assemblies and SAP pressure tubes. A conceptual design study of a SAP-clad natural uranium carbide fueled reactor, TI-015-10-001,⁽¹¹⁾ was issued in March 1967, based on the same 1965 technology as the enriched design. Most recently, "improved" natural and enriched SAP-clad uranium carbide conceptual design studies, TI-015-10-003,⁽¹²⁾ were completed, incorporating the information obtained from the HWOCR design and development work since 1965. In addition, a brief conceptual design study, TI-015-10-002,⁽¹³⁾ was performed in 1967 on a natural uranium reactor design fueled with uranium metal fuel (alloyed with trace impurities), clad with a zirconium alloy in an annular configuration.

A more extensive evaluation study was carried out on the feasibility of extrapolation of the HWOCR to very large plant sizes (10,000 Mwt) for the dual purpose of power generation and water desalination.^(8,9) This study also incorporated a fuel system optimization which resulted in the selection of the rod-type SAP-clad hyperstoichiometric uranium carbide as the reference fuel system for this application.⁽¹⁴⁾

The reference coolant in all these designs is a mixture of ortho- and meta-terphenyl (Santowax OM) containing 10% high boiler decomposition products, with recovery of useful coolant from the decomposition products through hydrocracking.

Section II-A presents:

- 1) A general description of a uranium carbide fueled HWOCR central station plant, with parameter data for the reference design, the natural uranium design, and the two improved designs.
- 2) A discussion of the technical bases and criteria used in these designs.
- 3) A description of the uranium metal fueled natural uranium design.

The description and evaluation of a dual-purpose plant are presented in Section II-D of this report.

2. General Description of Uranium Carbide Fueled 1000-Mwe HWO CR

An artist's sketch of the reference HWO CR plant is shown in Figure 3. Some of the reactor and plant parameters for this design, the natural uranium design, and the two improved designs are presented in Table 2.

Figures 4 and 5 show a general arrangement of the reactor components. The fuel and organic coolant are contained in vertical pressure tubes. These tubes are located on a square lattice spacing of approximately 11 in. They are about 4 in. in diameter and 20 ft long. They are made of SAP for the enriched uranium designs and a zirconium alloy for the natural uranium designs. These pressure tubes penetrate the calandria tank containing the heavy water moderator. The calandria tubes are made of Zircaloy-2, and are thermally insulated from the pressure tubes by a gas gap. The moderator is circulated through the calandria tank and maintained at low temperature and pressure. The reactor is regulated by horizontal control rods which penetrate the sides of the calandria tank. The actuators are outside the shield and are accessible during operation. Emergency shutdown is achieved by the addition of boron to the moderator. The modular construction of the pressure tubes and control devices shown in Figure 5 makes it feasible to build reactors of different sizes, including the very large ones, without redesigning these components. This standardization also precludes the requirement for increases in reactor assembly height due to pig-tail stacking with increased reactor size.

Each pressure tube contains five fuel assemblies approximately 44 in. long. Figure 6 shows a view of a fuel assembly. Each assembly consists of finned SAP-clad hyperstoichiometric uranium carbide fuel rods (37 in the reference design).

The design incorporates on-power refueling which minimizes reactivity requirements and is expected to allow high plant availability. A pair of refueling machines are located above and below the reactor. The refueling scheme is to move the fuel through the reactor in opposite directions in alternate pressure tubes. This procedure results in each nearly spent fuel assembly being surrounded by four recently inserted fresh fuel assemblies. Coolant flow is in the

TABLE 2
URANIUM-CARBIDE-FUELED HWO CR PLANT
CHARACTERISTICS

	Enriched (AI-CE-Memo-6) (Reference Design)	Natural (TI-015-10-001)	Improved Designs	
			Enriched	Natural
			(TI-015-10-003)	(TI-015-10-003)
Plant				
Net power generation (Mwe)	1076	1064	1076	1076
Net plant efficiency (%)	34.8	32.6	36.4	33.4
Steam conditions (psig/°F-°F)	900/725/725	600/675/675	1200/755/755	600/710/710
Reactor outlet temperature (°F)	750	700	780	735
Reactor ΔT (°F)	155	170	195	211
Reactor ΔP (psi)	184	201	200	195
Number of coolant loops	3	3	2	2
Primary coolant flow rate (lb/hr)	110 x 10 ⁶	108 x 10 ⁶	84 x 10 ⁶	86 x 10 ⁶
Reactor				
Calandria height/diameter (ft/ft)	20/25	20/37	20/20	20/30
Number of pressure tubes	492	927	187	626
Pressure tube material	SAP	Zr Alloy	SAP	Zr Alloy
Pressure tube ID/OD (in./in.)	4.32/4.55	3.75/3.89	5.30/5.44	3.75/3.87
Lattice pitch (in.)	10.5	11	12.9	11
D/U new fuel atom ratio	30.3	36	30.3	36
Total D ₂ O inventory (lb)	654,000	1,170,000	410,000	820,000
Fuel				
Geometry	37 Rod	19 Rod	55 Rod	37 Rod
Average specific power (kwt/kgU)	25.1	14.1	37.5	20.9
Fuel inventory (MTU)	123	232	79	157
Feed enrichment (wt % U ²³⁵)	1.16	Nat.	1.16 Avg	Nat.
Discharge enrichment (wt % U ²³⁵)	0.21	0.22	0.20	0.18
Discharge fissile plutonium (gm/kgU)	3.16	2.75	3.30	2.85
Average discharge burnup (Mwd/MTU)	15,000	7,830	18,000	8,800
Maximum discharge burnup (Mwd/MTU)	20,000	11,000	20,000	9,800

Figure 3. Artist's Conception of HWO CR Reference Plant

Figure 4. HWOCR Reactor Assembly

7680-5186A

Figure 5. HWOCR Pressure Tube Module

00-154368A

Figure 6. HWOCR Fuel Assembly

7680-15266-1

Figure 7. HWO CR Reactor Complex

7-1167 UNC

00-154115E

Figure 8. Flow Schematic of HWO CR Reference Plant

same direction as fuel movement, so that the fresh fuel is cooled by the lowest temperature coolant. The resulting combination of axial flux flattening and high average temperature differential between fuel cladding and coolant maximizes the amount of power which can be generated in each pressure tube. At equilibrium conditions, all the fresh fuel is identical in enrichment; radial power flattening is achieved by allowing the central fuel elements to go to a higher burnup. An overall elevation view of the reactor complex is shown in Figure 7.

A simplified flow schematic of the plant is shown in Figure 8. A modular approach has been taken for the coolant loops. Three loops are used for the 1000-Mwe reference plant. Each loop consists of a pump, steam generator, superheater, reheater, and the associated valves and piping, and is designed to handle about 1000 Mwt. The pumps have a capacity of 83,000 gpm and, in the reference design, were located at the reactor outlet in order to minimize the pressure in the reactor. The steam generated in the three coolant loops goes to a single 1000-Mwe turbine-generator. The steam cycle conditions are 900 psig, 725°F, with reheat to 725°F. For a demonstration plant either one, or preferably two, coolant loops would be designed for use so that the development of reliable large organic components would not have to be repeated for subsequent applications.

The present containment arrangement places the reactor and its emergency cooling system inside the containment building, with the main coolant pumps and steam generators located outside. Isolation valves in tandem are placed where the coolant lines penetrate the building. These will close rapidly and automatically in case of an accident. By placing the steam generators outside the reactor building, the necessity of containing the large volume of steam resulting from a massive steam generator failure is avoided, as are the numerous penetrations of the structure for the steam lines. The result is an effective safeguard system at a moderate cost.⁽¹⁵⁾

3. Technical Bases and Criteria of the UC Fueled HWO CR Designs

A brief discussion of the technical bases and criteria used in the uranium carbide fueled HWO CR designs and design studies is presented below. A more detailed evaluation is presented in References 10, 11, and 12. Table 3 summarizes the key bases and criteria used in establishing the design parameters presented in Table 2. The associated economic and fuel utilization evaluations are presented in Sections II-B and II-C.

TABLE 3
URANIUM-CARBIDE-FUELED HWO CR DESIGN
BASES AND CRITERIA

	Enriched (AI-CE-6) (Reference)	Natural (TI-015-10-001)	Improved Designs	
			Enriched (TI-015-10-003)	Natural (TI-015-10-003)
Discharge Burnup (Mwd/MTU)				
Maximum	20,000	11,000	20,000	9,800
Average	15,000	7,830	18,000	8,800
Maximum Fuel Element Heat Release (kw/ft)	27	25	33	33
Number of Fuel Rods Per Fuel Assembly	37	19	55	37
Maximum Fuel Cladding Circumferential Strain (%)	0.3	0	0.3	0
Maximum Fuel Cladding Temperature (including film formation allowance) (°F)	870	870	880	880
Coolant Heat Transfer Un- certainty Factor	1.15	1.15	1.05	1.05
Coolant Film Formation Allowance				
Increased Thermal Resist- ance (Btu/hr-ft ² -°F) ⁻¹	1.0×10^{-4}	1.0×10^{-4}	0.5×10^{-4}	0.5×10^{-4}
Pressure Drop Factor	1.10	1.10	1.10	1.10
Fuel Assembly Coolant Mixing (%)	64	64	92	92
Maximum Coolant Velocity (ft/sec)	30	30	36	34
Form of Fuel Assembly Enrichment	Uniform	Uniform	Two-Zone	Uniform
Fuel Assembly Local Power Peaking Uncertainty Factor	1.22	1.22	1.06	1.06
Fuel Assembly End Plate Material	Zr Alloy	Zr Alloy	Zr Alloy	Zr Alloy
Reactor Exit Coolant Tem- perature (°F)	750	700	780	735
Pressure Tube Material	SAP	Zr Alloy	SAP	Zr Alloy
Pressurized Pressure Tube- Calandria Tube Annulus	No	No	Yes	Yes
Location of Main Coolant Pump	Hot Leg	Cold Leg	Cold Leg	Cold Leg

It is to be noted that the "improved" uranium carbide plant designs are based on improvements of a near-term category. They represent gains which can reasonably be expected as a result of the recently concluded HWOCR development program. Further gains which could be anticipated as a result of a more extensive development program would include such advanced features as roughened fuel element cladding surfaces, controlled mixing, corrugated pressure tube - calandria tube assemblies, once-through steam generators, and a slightly higher temperature moderator system. The improved designs discussed in this report take no credit for any of these features.

a. Burnup

The burnups for these designs are based on economic considerations and do not represent metallurgical or engineering limits. All of these burnups are lower than the maximum burnup ($>25,000$ Mwd/MTU) which had been planned for demonstration under the HWOCR development program. Maximum-to-average burnup ratios are decreased in the improved designs because of reductions in local power peaking factors. Higher average burnups for the improved concepts are obtained by reductions in parasitic absorption while the average feed enrichment is kept the same as in the reference cases. The lower parasitic absorption can be achieved through reductions in pressure tube and calandria tube thicknesses, which are made possible by pressurizing the pressure tube - calandria tube annulus. A small decrease in fuel cladding thickness is incorporated through improved design of end plates and end plug attachments. The major portion of the fuel cycle cost improvement is attributable to the higher average burnups. Higher plant efficiencies also help reduce fuel costs.

b. Maximum Heat Release Rate and Cladding Strain

A maximum heat release rate of 33 kw/ft is used in the improved concepts. This is based on keeping the maximum fuel temperature below 2400°F and the maximum circumferential cladding strain to less than 0.3%. It is presently expected that a strain of 0.5% would still be acceptable.

c. Maximum Fuel Cladding Temperature

The maximum HWOCR cladding temperature is presently limited by fuel-cladding reaction considerations. Present information indicates satisfactory compatibility between hyperstoichiometric UC and SAP at temperatures of up to

900°F. The maximum cladding temperature of the improved designs is 880°F, which is 10°F greater than that of the reference designs. This requires an increase in cladding thickness of only 0.001 in. to offset the small loss in strength when going from 870 to 880°F. The use of end plug attachments to cladding with finning over the entire length of the rods reduces the cladding thickness requirement. This has been incorporated into the improved designs, thereby reducing cladding thickness by a few mils.

d. Coolant Heat Transfer Uncertainty Factor

The heat transfer correlations and associated physical property values used for the reference designs are based on those which result in the lowest, rather than the most likely, heat transfer coefficients. This conservatism, in the form of an uncertainty factor, amounts to about 1.15. For the improved designs, this factor is reduced to 1.05 to reflect the most likely situation which would evolve from testing specific fuel assembly geometries in Santowax OM at HWO CR temperatures, pressures, and velocities.

e. Coolant Film Formation Allowance

Early data from the U-3 loop irradiation with the U-305 experiment have indicated that maximum film formation rates might cause a thermal resistance at the surface of the cladding of the fuel elements as high as 1.7×10^{-4} (Btu/hr-ft²-°F)⁻¹ at the point of maximum cladding surface temperature, and that maximum fuel assembly pressure drop might increase by as much as a factor of 1.3 (though a ΔP instrumentation problem existed throughout the U-305 irradiation). However, recent experience at the U-3 loop (U-305 Alternate 1) with a new instrumentation system and at the WR-1 reactor indicates that proper coolant purity control (including use of approximately 300 ppm of water) reduces the film formation rate appreciably. The corresponding maximum thermal resistance equivalent at the cladding hot spot is expected to be less than 0.5×10^{-4} (Btu/hr-ft²-°F)⁻¹ and the fuel assembly pressure drop increase less than 10%. The improved design concepts are based on these most recent data.

f. Fuel Assembly Coolant Mixing

Before HWO CR mixing tests had been run, a conservative interpolation of existing Canadian⁽¹⁶⁾ and Westinghouse⁽¹⁷⁾ water mixing data were used in the initial HWO CR reference design. This interpolation indicated that more than

70% mixing would be expected to take place within the fuel assembly. The design value of 64% mixing for the reference designs is based on this information. Subsequent HWO CR mixing tests and separate noise analysis flow studies have indicated coolant mixing in the HWO CR fuel geometry to be greater than 94%.⁽¹⁸⁾ The improved designs are based on a mixing factor of 92%.

g. Maximum Coolant Velocity

Maximum coolant velocity in the U-305 irradiation test series has been limited to 30 ft/sec, as has that of the reference designs. Subsequent out-of-pile loop tests have been run at velocities of over 35 ft/sec without indication of a basic difference in vibration problems between 30 and 35 ft/sec. Because of this, the maximum coolant velocity in the improved designs has been increased to 35 ft/sec.

h. Form of Fuel Assembly Enrichment

Fuel assemblies with about six square inches of fuel cross-sectional area are used for the reference designs. The feed enrichment is uniform throughout each fuel assembly. The fuel area for the improved enriched uranium design has been increased to approximately nine square inches to allow a reduction in the number of pressure tubes. To prevent the local radial power peaking factor from increasing above that of the reference enriched uranium design, two-zone enrichment is used in each fuel assembly. This change has not been made on the improved natural uranium design because of its greater sensitivity to concomitant reactivity losses.

i. Local Power Peaking

The reference designs are based on combined local fuel assembly end peaking factors of 1.22. The nuclear analysis conducted subsequent to that performed for the reference designs indicates that end peaking can be reduced by careful selection of low enrichment fuel slugs at the ends of the fuel columns in the outer row of rods so that the steady-state local peaking factor can be reduced to 1.06. Both improved designs have taken advantage of this lower local peaking factor.

j. Fuel Assembly End Plate Material

Zircaloy end plates are used in all of these designs for neutron economy. In order to minimize hydriding, a concentration of approximately 300 ppm of water is maintained in the coolant.

k. Reactor Exit Coolant Temperature

The 750°F exit temperature of the enriched uranium reference design is based on an early economic analysis. The exit temperature of 780°F of the improved enriched uranium design was established on the basis of a more recent economic evaluation and the use of a higher allowable cladding temperature. The reference natural uranium design exit temperature of 700°F is based on the heat transfer requirements of the low organic volume fraction of this design and earlier Zircaloy pressure tube hydriding uncertainties. The increase in exit temperature of the improved natural uranium design to 735°F reflects the results of recent AECL studies⁽¹⁹⁾ which show increased optimism for zirconium alloy pressure tube performance at these temperatures.

l. Pressure Tube

SAP is used as the pressure tube material in the enriched designs because of its high strength at high temperature characteristics. Zirconium alloy pressure tubes are used in the natural uranium designs for neutron economy. The SAP pressure tube is limited by its low strain rate strength. Hydriding is a limiting condition for the zirconium alloy pressure tubes.

The reference designs are based on conventional pressure tube - calandria tube arrangements with essentially atmospheric pressure in the annulus between the two tubes. In the improved designs, this annulus is pressurized, thereby reducing the required thicknesses of both tubes.

m. Location of Main Coolant Pump

Although the preliminary design optimization indicated the desirability of locating the main coolant circulating pumps in the hot leg of each coolant loop, subsequent studies of a more detailed nature have indicated some cost savings associated with locating the pumps in the cold leg. For this reason, all the designs following AI-CE-Memo-6 have incorporated the cold leg location for the main coolant pumps.

4. Natural Uranium Metal Fueled 1000-Mwe HWO CR

Fuel utilization is one of the most important of the characteristics which will determine the long-range economics and fuel requirements of nuclear power plants. The ability of a reactor to produce a large amount of plutonium per unit of energy generated is the special aspect of fuel utilization which can

be expected to outweigh all others in a nuclear industry oriented toward fast breeder reactors. The incentives of high plutonium production rate and efficient fuel utilization lead naturally to the consideration of uranium metal fuel for the natural uranium fueled HWOCR. This fuel offers the greatest promise of achieving a high plutonium production rate at a reasonable cost due to the potential low unit fabrication costs for coextruded, zircaloy-clad, annular fuel shapes, and its high density which improves the neutron economy of the reactor. Low fabrication costs are a requirement for competitive fuel costs at the low burnups which yield high plutonium production rates.

The use of the uranium-metal fuel system represents a departure from the mainline of HWOCR development undertaken for the nearer-term plants described previously. An increased scope development program would have been required to confirm the bases and criteria used in the uranium metal conceptual design evaluation study.

The uranium-metal fueled HWOCR design⁽¹³⁾ utilizes fuel subassemblies consisting of three concentric, zircaloy-clad, uranium alloy cylinders. The fuel material is assumed to be an advanced version of the relatively radiation-stable class of trace alloys of uranium commonly designated "adjusted uranium." The basis for the design is the assumption of a sufficient improvement in the radiation performance of this type of fuel to allow achievement of peak burnups of 8000 Mwd/MTU at a maximum fuel temperature of 950°F, with less than 5% volumetric swelling. The design assumptions made with respect to the thermal-hydraulic performance of the core are consistent with those of the nearer term uranium carbide fueled designs except for the use of zircaloy fuel cladding. The design assumes a maximum clad surface temperature (including film formation allowance) of 820°F and a clad thickness consistent with the predicted strength and hydriding characteristics of the more hydriding-resistant zirconium alloys such as Ozhennite. Table 4 presents a summary of the more important uranium metal fueled plant characteristics.

The principal differences between this uranium metal fueled design and the carbide fueled designs described previously arise from the high fuel density and the annular fuel geometry with unfinned cladding. The relatively massive fuel segments and unfinned cladding result in a low average specific power, less than one-third that of the enriched uranium reference design and one-half that

TABLE 4
NATURAL URANIUM-METAL-FUELED HWO CR PLANT
CHARACTERISTICS

Plant	
Net power generation (Mwe)	1076
Net plant efficiency (%)	34.0
Steam conditions (psig/°F/°F)	900/675/675
Reactor outlet temperature (°F)	700
Reactor ΔT (°F)	139
Reactor ΔP (psi)	216
Number of coolant loops	3
Primary coolant flow rate (lb/hr)	128×10^6
Reactor	
Calandria height/diameter (ft/ft)	18/39
Number of pressure tubes	1044
Pressure tube material	Ozhennite 0.5
Pressure tube ID/OD (in.)	3.60/3.70
Lattice pitch (in.)	12.25
D/U atom ratio	30
Total D ₂ O inventory (lb)	1,470,000
Fuel	
Geometry	annular/3 rings
Average specific power (kwt/kgU)	7.6
Fuel inventory (MTU)	415
Feed enrichment (wt % U ²³⁵)	Natural + Tailings*
Discharge enrichment (wt % U ²³⁵)	0.20
Discharge fissile plutonium (gm/kgU)	2.7
Average discharge burnup (Mwd/MTU)	6700
Maximum discharge burnup (Mwd/MTU)	8000

*76% Natural, 24% at 0.25 wt % U²³⁵

of the natural uranium carbide design. These characteristics also result in a larger lattice spacing to achieve the same D/U atom ratio as the reference design, thus requiring more than double the D₂O inventory. The neutron economy of this system is much better than that of the uranium carbide design, with an average conversion ratio approaching 0.90. The relatively low exposure goal

for the fuel allows the use of an average feed enrichment below natural. Actually, the feed fuel consists of two inner rings of natural enrichment and an outer ring (~40% of the fuel by weight) of 0.43 wt % U^{235} enrichment achieved by blending natural uranium with diffusion plant tailings. The low feed enrichment and massive fuel geometry result in very low fabrication costs which allow the economic use of short fuel burnups and consequent high rates of plutonium production (~50 kg per 10^9 kwhe as compared with ~25 kg per 10^9 kwhe for the enriched uranium reference design).

B. ECONOMICS

1. Financing Basis

The economic basis and procedures used for establishing the capital cost estimates and the components of the energy generation cost estimates are in accordance with those used by the AEC and ORNL in their evaluation of power reactor concepts. The analysis presented is based on two representative types of utility financing: investor-owned and publicly-owned plants. The economic basis for these two types of financing is shown in Table 5.

TABLE 5
REFERENCE ECONOMIC BASIS

	Investor- Owned	Publicly- Owned
Fixed Charges on Plant Capital (%/yr)	12	7
Fixed Charges on Non-Depreciated Assets — Includes D ₂ O and Fuel Investment (%/yr)	10	5
Present Worth Discount Rate (%/yr)	6	4
Annual Plant Factor	0.80	0.80

The economics presented in this section reflect technology which would be available and demonstrated through the construction and operation of an HWO CR demonstration plant by the early 1970's. The economic parameters are representative of one of several 1000-Mwe units, committed for on-line service in the 1970's and 1980's, which are the forerunners of a growing HWO CR economy. Accordingly, the economic parameters related to fuel cycle costs are based on

large-scale processing geared to service an expanding HWOOCR economy. An annual plant factor of 0.80 is used in this economic analysis in accordance with established ground rules; it is anticipated, however, that, with its on-power refueling and access to its low radioactivity main coolant systems for maintenance, the HWOOCR will have an availability which will allow an appreciably higher plant factor.

For purposes of this analysis, the energy generation cost for the HWOOCR is divided into the following components:

1) Fixed Charges on Invested Capital

- a) Plant Capital
- b) Moderator and Coolant Investment
- c) Fuel Investment

2) Operating Costs

- a) Fuel Cycle
- b) Operation and Maintenance (including fluid make-up)
- c) Nuclear Liability Insurance

2. Fixed Charges on Invested Capital

The fixed charges include carrying charges on the invested capital to cover return on investment, taxes, insurance, etc., as well as charges to provide for the recovery of capital invested in depreciable items when applicable.

a. Plant Capital

The plant is assumed to be the first unit constructed on an undeveloped hypothetical site in New England, as described in TID 7025.⁽²⁰⁾ A summary of the plant capital costs for the reference plant is given in Table 6. The total depreciable capital cost for the reference plant is \$120,505,700 as an investor-owned plant, and \$116,590,400 as a publicly-owned plant. Based on the rated net generating capacity of 1076 Mwe, the unit costs are \$112/kwe and \$109/kwe as an investor-owned plant and publicly-owned plant, respectively. The most recent ORNL evaluation⁽²¹⁾ indicates total capital costs of \$124/kwe and \$120/kwe for investor-owned and publicly-owned plants, respectively. These capital costs include the latest increase in turbine prices and are based on more conservative design criteria than those originally proposed.

TABLE 6
HWO CR CAPITAL COST ESTIMATES (1000-Mwe Plants)

Account No.	Description	Enriched UC 1076 Mwe		Natural UC 1064 Mwe		Natural U-Metal 1076 Mwe	
	Direct Construction Cost						
21	Structures and improvements	\$ 11,926,000		\$ 14,326,000		\$ 14,426,000	
22	Reactor plant equipment	38,719,900		48,814,000		50,900,000	
23	Turbine-generator units	30,925,600		30,925,600		30,925,000	
24	Accessory electric equipment	3,345,300		3,345,300		3,345,000	
25	Miscellaneous power plant equipment	785,400		785,400		785,000	
	Total Direct Construction Cost	\$ 85,702,200		\$ 98,196,300		\$100,381,000	
	Indirect Construction Cost						
	General and administrative 6%	5,142,100		5,892,000		6,023,000	
	Subtotal	\$ 90,844,300		\$104,088,300		\$106,404,000	
	Miscellaneous construction 1%	908,400		1,040,900		1,064,000	
	Subtotal	\$ 91,752,700		\$105,129,200		\$107,468,000	
	Engineering design and inspection						
	Architectural and engineering services 5%	4,587,600		5,256,500		5,373,000	
	Subtotal	\$ 96,340,300		\$110,385,700		\$112,841,000	
	Nuclear engineering 2%	1,926,800		2,207,700		2,256,000	
	Subtotal	\$ 98,267,100		\$112,593,400		\$115,097,000	
	Startup costs 35% of operation and maintenance	605,400		685,000		650,000	
	Subtotal	\$ 98,872,500		\$113,278,400		\$115,747,000	
	Contingency 10%	9,887,200		11,327,800		11,575,000	
	Total Construction Cost	\$108,759,700		\$124,606,200		\$127,322,000	
	Customer Cost						
	Interest during construction 10.8% and †7.2%	11,746,000	†7,830,700	13,457,500	†8,971,600	*13,751,000	†9,167,000
	Total Depreciable Capital Cost	120,505,700	116,590,400	138,063,700	133,577,800	141,073,000	136,489,000
20	Land and land rights	360,000	360,000	360,000	360,000	360,000	360,000
	Total Capital Cost	\$120,865,700	\$116,950,400	\$138,432,700	\$133,937,800	\$141,433,000	\$136,849,000
	\$/Net kw	112	109	130	126	131	127

*Investor-owned

†Publicly-owned

Direct construction costs are compiled according to the AEC preferred system of accounts given in Reference 20. Direct construction cost estimates are based on detailed take-offs and vendor estimates for major equipment items.

Indirect construction costs have been estimated in accordance with the guidelines established in ORNL 3686.⁽²²⁾ Based on these guidelines, general and administrative, miscellaneous-construction, engineering, startup, contingency, and interest-during-construction costs are fixed percentages applied to the direct construction costs in a cumulative manner. Startup charges are estimated at 35% of the annual operating and maintenance costs, including supplies but excluding fuel. The interest-during-construction costs depend on the type of utility ownership. The annual interest rates applicable to investor-owned and publicly-owned utilities are assumed to be 6 and 4%, respectively, in keeping with the reference bases given in Table 5. It should be noted that these estimates are based on January 1967 dollars with no allowance for escalation. For the purposes of the estimate, the plant terminates at the low voltage side of the main transformer.

Capital costs for the natural uranium carbide and uranium metal fueled plants have been estimated by accounting for the differences between these plants and the enriched uranium reference plant. For the most part, these differences occur in Account 21 (Reactor Containment) and Account 22 (Reactor Complex, Fuel Handling Equipment, Instrumentation and Control, and Steam Generating Equipment). The cost summaries for these plants are included in Table 6.

b. Moderator and Coolant Investment

The costs of the required inventories of heavy water and organic coolant have been estimated using unit costs of \$17/lb and 12¢/lb, respectively, and taking into account the sizes of the calandria, piping systems, and storage tanks in each design. Fixed charges of 10% and 5% per year for investor-owned and publicly-owned utilities, respectively, have been used since these inventories are not considered depreciable items.

c. Fuel Investment

The fuel investment required for the initial core loading of each plant has been amortized over the 30-year plant life at carrying charge ratios of 10% and

5% per year for investor-owned and publicly-owned plants, respectively, plus sinking fund deposits which take into account the scrap value of the final core.

3. Operating Costs

The operating costs include fuel replacement, operation and maintenance, make-up of heavy water and coolant, and the cost of nuclear liability insurance.

a. Fuel Cycle

The fuel cycle costs for the HWO CR plants have been estimated for equilibrium conditions. These plants will initially be loaded with graded enrichments in five zones approximating the equilibrium core and, therefore, the effect of ignoring the variable fuel costs during the first few years is felt to be very small. The reported fuel cost includes annual payments for uranium, fuel fabrication and chemical recovery based on a 15,000-Mwe HWO CR industry throughput, shipping, interest on fuel in fabrication and storage, fabrication progress payments, and credit for fissile plutonium. Fuel cost bases for all three plants are summarized in Table 7.

TABLE 7
REFERENCE FUEL COST BASIS

	Initial Fuel			Replacement Fuel		
	Enriched UC	Natural UC	Natural U-Metal	Enriched UC	Natural UC	Natural U-Metal
Uranium Price						
U ₃ O ₈ (\$/lb)	8.0	8.0	8.0	8.0	8.0	8.0
Separative work (\$/kgU)	30.0	30.0	30.0	30.0	30.0	30.0
U ₃ O ₈ to UF ₆ conversion (\$/kgU)	2.7	2.7	2.7	2.7	2.7	2.7
UF ₆ to U-billet conversion, depleted fuel (\$/kgU)			3.85			3.85
U ₃ O ₈ to UF ₄ to U-billet conversion, natural fuel (\$/kgU)			3.50			3.50
Average fuel preparation cost, natural and depleted rings (\$/kgU)			3.85			3.65
Fabrication and Shipping Cost (\$/kgU)	40.0	23.0	10.50	36.0	22.5	10.50
Preirradiation time (yr)	~0.5	~0.5	~0.5	~0.2	~0.2	~0.2
Chemical Recovery Cost						
Spent fuel shipping (\$/kgU)	2.0	1.5	1.1	2.0	1.5	1.1
Separations (\$/kgU)	18.0	11.5	9.3	18.0	11.5	9.3
Conversion of UNH to UF ₆ (\$/kgU)	3.0	3.0	3.0	3.0	3.0	3.0
Uranium and Pu losses (%)	1.0	1.0	1.0	1.0	1.0	1.0
Postirradiation time (yr)	0.67	0.67	0.67	0.67	0.67	0.67
Plutonium Credit (\$/fissile gm)	10.0	10.0	10.0	10.0	10.0	10.0
Fuel Storage in Lieu of Recovery (\$/kgU)	3.0	3.0	3.0	-	-	-

b. Operation and Maintenance

Operation and maintenance costs are based on a permanent staff of 70 persons, including allowances for cost of fringe benefits and general and administrative expenses. The cost of operating materials and supplies, including contract services, has been estimated at 0.05 mills per thermal kwh consistent with current experience in the utility industry. Organic make-up costs are based on the use of a hydrocracker with 90% coolant recovery.

c. Nuclear Liability Insurance

Nuclear insurance is based on a \$60,000,000 policy and federal indemnity up to a total coverage of \$500,000,000. The total estimated premium is \$350,000 per year.

4. Energy Generation Costs

The total energy generation costs for the three basic plant designs are summarized in Table 8. These costs are based on the stated net station generating capacity and an average plant capacity factor of 0.80. This capacity factor takes no credit for the high anticipated availability resulting from on-line refueling which removes the requirement for refueling shutdowns estimated to require about 20 days per year more than ordinary turbine maintenance with batch-loaded reactors, and from access to the low radioactivity main coolant systems for maintenance. Preliminary information received from ORNL indicates their estimates to be higher by approximately 0.2 to 0.3 mills/kwhe on the basis of most recent turbine prices and more conservative design criteria.⁽²¹⁾

The energy costs shown in Table 8 are based on near-term projections of the value of uranium ore and fissile plutonium. In order to assess the relative economic performance of these plant designs under possible conditions of rising uranium prices, estimates have been made of the total energy generation costs if the U_3O_8 price were \$20/lb instead of the \$8/lb assumed in Table 8. The associated value of fissile plutonium was increased from \$10/gm to \$18/gm to reflect this higher ore cost as well as the potential demand for breeder inventories. The results of these estimates are graphically exhibited in Figure 9, which clearly indicates the very attractive relative economic performance of the natural uranium fueled designs, particularly the one using uranium-metal fuel, under conditions of high uranium prices. This is the feature of the long-range HWOCR reactor designs which would provide unique stability of fuel costs if the supply of low-cost uranium were to become seriously depleted.

Figure 9. Comparative Effect of Increasing Ore Cost on Total Energy Cost

TABLE 8
HWO CR ENERGY GENERATION COST ESTIMATES
(1000-Mwe Plants)

	Enriched SAP-Clad UC (Reference) 1076 Mwe		Natural SAP-Clad UC 1064 Mwe		Natural U-Metal 1076 Mwe	
	Investor- Owned	Publicly- Owned	Investor- Owned	Publicly- Owned	Investor- Owned	Publicly- Owned
Investment \$10 ⁶						
Plant	121	117	138.1	133.6	141.1	136.5
D ₂ O, coolant, land and working capital	11.4	11.4	20.3	20.3	25.3	25.3
Initial fuel	9.7	9.5	8.7	8.5	9.5	9.3
\$/Net kw						
Plant	112	109	130	126	131	127
D ₂ O, coolant, land and working capital	10.6	10.6	19.5	19.5	23.8	23.8
Initial fuel	9.0	8.8	8.2	8.0	8.8	8.6
Energy Generation Cost (mills/kwh)						
Fixed charges						
Plant	1.92	1.09	2.24	1.26	2.25	1.27
D ₂ O, coolant, land and working capital	0.15	0.08	0.28	0.14	0.34	0.17
Initial fuel	0.14	0.08	0.13	0.07	0.14	0.08
Operating costs						
Fuel cycle	0.71	0.70	0.50	0.49	0.25	0.24
Operation and maintenance	0.23	0.23	0.25	0.25	0.24	0.24
D ₂ O, coolant makeup	0.03	0.03	0.03	0.03	0.03	0.03
Nuclear insurance	0.04	0.04	0.04	0.04	0.04	0.04
Total Energy Cost	3.22	2.25	3.47	2.28	3.29	2.07

Capital cost differentials for the "improved" uranium carbide designs described in Section II-A have been estimated without the benefit of detailed cost breakdowns, but the differences in plant design and equipment between these versions of the HWO CR and the reference cases (Tables 2 and 3) are simple to evaluate and their cost effects can be established fairly easily. For these improved designs, the economic effects of the proposed design and performance characteristics upon the unit energy costs are summarized on the basis of private financing in Table 9.

TABLE 9
URANIUM-CARBIDE-FUELED 1000-Mwe HWO CR ENERGY
GENERATION COST DIFFERENTIAL SUMMARY (mills/kwh)
(Investor-Owned Financing)

	Enriched (AI-CE-6) (Reference)	Natural (TI-015-10-001)	Improved Designs	
			Enriched (TI-015-10-003)	Natural (TI-015-10-003)
Reactor Plant	0	+0.23	-0.12	+0.11
D ₂ O	0	+0.12	-0.05	+0.03
Turbine Generator	0	+0.01	-0.06	-0.04
Buildings	0	+0.05	-0.01	+0.04
Subtotal	0	+0.41	-0.24	+0.14
Fuel Cycle	0	-0.22	-0.18	-0.30
Total	0	+0.19	-0.42	-0.16

C. FUEL UTILIZATION

1. Method of Analysis

In general, fuel utilization refers to nuclear reactor or nuclear power plant effectiveness in minimizing the drain on the natural resources of fissile isotopes while supplying some specified demand for nuclear energy. One of the principal objectives of the HWO CR project is the improvement of fuel utilization over that afforded by current types of conventional reactors. A quantitative method of defining fuel utilization is needed not only to measure the progress made toward a given objective, but also to serve as a means of evaluating the alternative reactor concepts being developed under the U. S. Civilian Power Reactor Program. The purpose of this section is to define such a method, and to reduce it to a figure of merit which can be applied to different reactor designs without ambiguity. The HWO CR designs described in Sections II-A can then be compared with a reference light water reactor design using this new figure of merit for fuel utilization.

Typical power reactors contain, at all times, an inventory of fissile isotope which is several times greater than the amount of fissile isotope actually destroyed by the reactor in a year of power operation. If the demand for nuclear power reactors is increasing with time, the fissile isotope required for inventory in the additional reactors built must be supplied from the natural

resources, just as surely as must the net fissile isotope consumed in the nuclear reaction. To characterize the fuel utilization of a particular reactor, one must take into account not only the fissile isotope destroyed by nuclear reaction, generally referred to as burnup, but also that tied up in inventory. This presents a problem, for the ratio of the inventory requirement to the burnup requirement depends on the rate of growth of the installed nuclear capacity; that is, on 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 utilization have resulted from failure to recognize the importance of the inventory requirement.

It is well known that, for many years past, the exponential has been the simplest mathematical function by which the growth of the electric utility industry could 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 taken here.⁽²³⁾ 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.

Assuming that the nuclear power industry is growing at an exponential rate characterized by a doubling time, T_d , one can define the following quantities:

- 1) The Exponential Investment (E.I.) of natural uranium is the ratio of the number of tons of U_3O_8 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 that time. Obviously, the value of the E.I. will depend on the doubling time, and it will be necessary to select an appropriate doubling time in order to arrive at a value of the E.I. which has a meaningful relationship to the fuel supply problem.

2) The Dynamic Utilization (D.U.) of natural uranium is the reciprocal of the E.I., in megawatts-years of electric energy produced up to the time, T, per ton of U_3O_8 fed into the industry up to that time. It is the D.U. 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 and relate them to some other quantities of importance.

Let:

$C = C_o e^{0.693 t/T_d}$ = Installed energy generating capacity at time, t,
in electric megawatts

S = Specific inventory including ex-core requirements = tons of U_3O_8
required to provide the U^{235} for 1 Mwe of installed reactor capacity

F = Plant factor

B = Specific make-up for burnup, in equivalent tons of U_3O_8 per
megawatt-year of electrical energy produced

W = Total electric energy produced from $t = 0$ to $t = T$

N = Total natural uranium, as tons of U_3O_8 , put into the system for
burnup plus inventory, from $t = 0$ to $t = T$

Then:

$$W = \int_0^T FC_o e^{0.693 t/T_d} dt = \frac{FC_o T_d}{0.693} (e^{0.693 T/T_d} - 1) \quad \dots(1)$$

$$N = BW + SC \quad \dots(2)$$

Therefore,

$$\begin{aligned} E.I. &= \frac{N}{W} = B + \frac{SC}{W} \\ &= B + \frac{SC_o e^{0.693 T/T_d}}{\frac{FC_o T_d}{0.693} (e^{0.693 T/T_d} - 1)} \\ &= B + \frac{0.693 S}{FT_d} \frac{e^{0.693 T/T_d}}{e^{0.693 T/T_d} - 1} \quad \dots(3) \end{aligned}$$

If one assumes $e^{0.693 T/T_d} \gg 1$, this can be simplified to:

$$E.I. = B + \frac{0.693S}{FT_d} \quad \dots(4)$$

2. HWO CR Fuel Utilization

The applications of the Exponential Investment and Dynamic Utilization are shown in Table 10 for several versions of the HWO CR and compared with the reference light water reactor (LWR) system. The value of T_d used for this evaluation is 7 years. This value was selected as being appropriate on the basis of the AEC projections of the growth of the nuclear power industry for the next several decades. Most projections fall within a T_d range of 5 to 10 years between now and the year 2000.

It is evident from Table 10 that, in terms of the dynamic utilization, the HWO CR, particularly when fueled with natural uranium, is much more effective in utilizing nuclear resources than is the light water reactor. The advantage of the HWO CR stems from both the low make-up requirement for burnup and from the low fissile fuel inventory requirement. Consequently, a growing nuclear power economy based primarily upon HWO CR advanced-converters would substantially increase the energy obtainable from the available uranium ore reserves. Furthermore, any increase in the price of uranium would have less influence on the energy generation cost of the HWO CR since its higher fuel utilization makes the fuel cost a smaller fraction of the total energy generation cost. This latter point is illustrated in Figure 9 in Section II-B.

Table 11 presents the same results in the form of actual inventory and burnup requirements for a 1000-Mwe plant. This table shows that:

1) The inventory requirement of the HWO CR is approximately half that of an LWR.

2) The burnup requirement of the enriched uranium HWO CR with or without plutonium recycle amounts to about 70% that of the LWR. The burnup requirement of the natural uranium fueled HWO CR amounts to about 70% that of the LWR without plutonium recycle, and to about 20 to 40% that of the LWR with plutonium recycle.

TABLE 10
DYNAMIC FUEL UTILIZATION

	LWR	HWO CR		
		Enriched UC (Reference)	Natural UC	Natural U-Metal
Basic Data				
Thermal efficiency (%)	32.0	34.8	32.6	34.0
Average specific power (kwt/kgU)	38.0	25.1	14.1	7.6
Discharge burnup (Mwd/kgU)	32.0	15.0	7.8	6.7
Feed enrichment (wt % U235)	3.26	1.16	0.71	0.60
Discharge enrichment (wt % U235)	0.92	0.21	0.22	0.20
Discharge fissile Pu: (gm/kgU)	6.44	3.16	2.75	2.70
(kg/Mwe-yr)	0.230	0.221	0.393	0.433
Utilization Factors				
Specific inventory* (tons U3O8/Mwe)	0.672	0.286	0.330	0.411
Specific burnup (tons U3O8/ Mwe-yr)				
with Pu recycle	0.176	0.118	0.075	0.035
without Pu recycle	0.241	0.180	0.186	0.158
Exponential investment† (tons U3O8/Mwe-yr)				
with Pu recycle	0.259	0.153	0.115	0.086
without Pu recycle	0.324	0.215	0.227	0.209
Dynamic fuel utilization† (Mwe-yr/ton U3O8)				
with Pu recycle	3.86	6.52	8.70	11.60
without Pu recycle	3.09	4.65	4.40	4.79

*Based on average fissile enrichment and a 1-yr excore residence time

†Based on a plant factor of 0.8 and a nuclear capacity doubling time of 7 yr

TABLE 11
FUEL UTILIZATION SUMMARY FOR A 1000-Mwe PLANT

Plant Factor = 0.80

Annual Energy Generation = 7×10^9 kwhe

	LWR	HWOCR		
		Enriched UC (Reference)	Natural	
			UC	U-Metal
Inventory, Including 1 yr Excore (tons U_3O_8)	672	286	330	411
Burnup (tons U_3O_8 /yr)				
With Pu recycle	141	94	59	28
Without Pu recycle	193	144	148	126
Fissile Pu Production (kg fissile Pu/yr)				
No Pu recycle	183	176	314	345

3) The fissile plutonium production is approximately the same for the enriched uranium fueled HWOCR and the LWR. It is 70 to 90% greater than that of the LWR in the case of the natural uranium fueled HWOCR.

The impact of the differences in fuel utilization indicated in Tables 10 and 11 can be more readily appreciated using the graphical illustration in Figure 10. This figure compares the ore requirements and fissile plutonium production per seven-billion kilowatt-hours of electricity (one year's operation of a 1000-Mwe central station at a plant factor of 0.80) generated in an LWR and a natural uranium fueled HWOCR.

D. DUAL-PURPOSE PLANT

The application of the HWOCR to dual-purpose power generation and water desalting was briefly discussed in Section I-C with respect to its special requirements, such as feasibility of very large plants, need for high plant factor, and effects of financing conditions. The HWOCR with its modular construction, on-power refueling, ease of access to the main coolant system for maintenance, low fuel cycle costs, and excellent flexibility in fuel selection is well suited for this application. Studies were therefore conducted to establish the feasibility of

A. PLUTONIUM USED TO INVENTORY FAST BREEDER REACTORS

B. PLUTONIUM RECYCLE IN HWOCR AND LWR

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Figure 10. Annual Uranium Consumption and Plutonium Production of HWOCR and LWR

extrapolation from the HWOOCR reference design to a very large plant design⁽⁸⁾ and optimize the fuel selection for this application.⁽¹⁴⁾ The conceptual design of a one-billion gallon per day, 3400-Mwe net, dual-purpose plant utilizing two 8250-Mwt HWOOCR reactors has been described in Reference 9.

1. Size Extrapolation and Fuel Optimization Studies

The initial phase of the HWOOCR study for dual-purpose application was directed toward determining the feasibility and practicability of extrapolating the reference power-only HWOOCR plant design to a 10,000-Mwt capability. The results of this study have been reported in AI-CE-Memo-28.⁽⁸⁾ It was concluded that it would be practicable to build a 10,000-Mwt HWOOCR nuclear steam supply plant and that its capital cost per unit of thermal energy generated would be significantly lower than that of a 3000-Mwt plant. There were no technical problems uncovered which were peculiar to the large size extrapolation and could not be solved by standard engineering design techniques.

A detailed study, AI-CE-Memo-59,⁽¹⁴⁾ was conducted to optimize the fuel selection and operating parameter range for the HWOOCR dual-purpose application. The fuel materials investigated consisted of hyperstoichiometric uranium carbide, uranium dioxide, and uranium metal, clad in zircaloy, SAP, and aluminum. Rodded, annular, and inverse-cluster geometries were considered in this evaluation. To evaluate these various (27) types of fuel elements, the cost of the steam produced was estimated for the most promising combinations of materials and geometries. The cost of available heat was determined by dividing the cost of the steam by the thermal efficiency associated with its temperature.

Under the prescribed ground rules, the SAP clad-rodded-uranium carbide fuel element provided available heat at the least cost per million Btu. The efficiency associated with the higher steam temperature attainable with the UC fuel outweighed the decreased fuel fabrication cost anticipated with annular uranium metal fuel elements.

2. Plant Description

The one-billion gallons per day, 3400-Mwe net, dual-purpose plant described in Reference 9 utilizes two 8250-Mwt HWOOCR units in parallel, each sharing in common the reactor fuel facilities, the heavy water moderator

Figure 11. Reactor Arrangement and Support, Dual Purpose Plant

process plant, and the waste disposal facilities. The steam which is developed in this nuclear heat source powers five 800-Mwe backpressure turbines which exhaust steam at 25 psia (240°F) to the brine heaters of ten 100-mgd multistage flash-type evaporators.

Typical characteristics of the reactor segment of this plant are presented in Table 12. The reactor arrangement is shown in Figure 11. The calandria tank is of the parallelepiped shape with semicircular ends. It is built up from standard modules which have been figuratively inserted between the cleaved halves of a circular 3000-Mwt reactor vessel. There are 76 (38 rows) complete modules with 52 shortened modules making up the semicircular ends. The pressure tube, fuel, and refueling machines are identical with those described for the power-only plants.

TABLE 12
HWO CR DUAL-PURPOSE PLANT REACTOR CHARACTERISTICS*

Thermal Rating (Mwt)	8,250
Number of Pressure Tubes	1,424
Calandria Length (ft)	61
Calandria Width (ft)	24
Pressure Tube Length (ft)	47
Reactor Inlet Temperature (°F)	574
Reactor Outlet Temperature (°F)	750
Number of Coolant Loops	4
Main Coolant Pump Capacity, Each of Four (gpm)	160,000
Main Coolant Pump Drive Horsepower	31,000

*For each reactor. Data from Reference 9.

The calandria is 61 ft long by 24 ft wide. A reinforced concrete biological shield, 6 ft thick, is placed around its periphery. A reactor of this power level exclusive of headering occupies an overall space of 73 by 36 ft. The containment for each of these reactors is 190 ft in diameter. The steam generators, superheaters, and main coolant pumps are located outside the containment in the steam generator building.

Extrapolation of the HWO CR reactor concept to the 8250-Mwt size can be done with relatively modest engineering development because of the modular design approach. Special design of the calandria is required in that it becomes a flat-sided pressure vessel which must be adequately cooled without increase in heavy water inventory. This is accomplished by providing light water filled partitions on the exterior flat sides of the calandria tank to give it adequate support. Control of the reactor by a computer which utilizes approximately 75 zones of control is practicable.

Each of the reactors is provided with eight headers which distribute the coolant through the core. These headers are connected in parallel to a reactor inlet mix tank and reactor outlet mix tank (Figure 12). Four individual heat transfer loops are arranged in parallel for each reactor. Each loop consists of a main coolant centrifugal pump, a steam generator, and a superheater. The superheated steam is manifolded from all eight heat transfer loops and distributed to the five back-pressure turbines for maximum plant flexibility and availability.

A heat balance for this plant is shown in Figure 13. The calculated on-site auxiliary power requirements are approximately 600 Mwe which, when deducted from the gross output of 4000 Mwe, leaves 3400 Mwe as salable power. A portion of this power would have to be used for transporting the product water to its point of use. The exhaust steam from the five back-pressure turbines provides the heat required by the desalting plant. The overall plant arrangement is shown in Figure 14.

3. Plant Economics

A summary of the economic analysis for the one-billion gallon per day, 3400-Mwe net, dual-purpose plant is shown in Table 13. The two products of a dual-purpose plant are fresh water and electricity, with the cost of water depending on the market value for power. Since this value depends on many factors relating to the nature and location of the power market, it is impossible to predict a specific cost for water at this time. The cost relationship between these two products which provides the necessary revenue to cover all production costs is shown in Figure 15. This figure shows, for example, that if power can be sold at the busbar for 2.0 mills/kwh, which is approximately the cost of power from a power-only plant of equivalent net capacity, then the cost of water

Figure 12. Flow Sheet, One Billion Gallons per Day, 3400-Mwe (net)
HWO CR Dual-Purpose Plant

3-28-66C

00-154943

Figure 13. Heat Balance, One Billion Gallon Per Day, 3400-Mwe (net) HWO CR Dual-Purpose Plant

00-154942A

Figure 14. Plot Plan, One Billion Gallons Per Day, 3400-Mwe (net) Dual-Purpose HWO CR Plant

AI-CE-87

TABLE 13
HWO CR DUAL-PURPOSE PLANT ECONOMICS*

Plant Output	
Reactor power (Mwt)	2 x 8250
Gross electrical power (Mwe)	5 x 800
Net electrical power (Mwe)	3400
Product water (MGD)	1000
Annual Production	
Electricity at 0.90 plant factor (10^9 kwh)	27
Product water at 0.94 plant factor (10^9 gal)	344
Investment ($\$10^6$)	
Nuclear steam supply	260
Power conversion	170
Water plant	<u>500</u>
Total depreciable plant	930
D ₂ O at \$20/lb and initial core	<u>120</u>
Total	1050
Annual Cost ($\$10^3$)	
Depreciable capital at 5.5% [†]	51,200
Fuel and D ₂ O inventory at 3.5% [†]	4,200
Plant operation and maintenance [§]	19,300
Fuel expense at \$8/lb U ₃ O ₈	<u>24,900</u>
Total	99,600

* Data taken directly from Reference 9, with no attempt at normalization of economic and plant factor bases.

† Based on the rates used by Bechtel in the MWD Study TID 22330.

§ Includes nuclear insurance; heavy water and organic makeup; maintenance supplies and labor for the reactor plant, turbine plant, and water plant; chemical treatment of water plant feed.

Figure 15. Cost of Water as a Function of Value of Power

at the site boundary will be about 13 cents per 1000 gallons. Conversely, if water is valued at 10 cents per 1000 gallons, a marketable cost for agriculture, then the cost of power is about 2.4 mills/kwh. It appears from this analysis that with further anticipated improvements in desalting technology and with the depletion of low-cost water sources, water produced from future large HWO CR dual-purpose plants could well be justified, not only for domestic consumption, but also for agricultural use.

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III. HWOCR RESEARCH AND DEVELOPMENT STATUS

A. COOLANT

1. Introduction

The application of organic cooling to the heavy water moderated reactor is based on the extensive experience with the use of this type of coolant derived from the development programs conducted during the past decade in the United States, Canada, and the Euratom countries. Additional work has been performed in Spain, Denmark, and the USSR. This experience has included the operation of both in-pile and out-of-pile facilities. The USAEC has sponsored the operation of the Organic Moderated Reactor Experiment (OMRE) from 1957 to 1963, of the Piqua Nuclear Power Facility (PNPF) since 1963, and of the MIT reactor in-pile loop since 1961. AECL has operated the X-7 in-pile loop in the NRX reactor since 1960, the U-3 loop in the NRU reactor since 1963, and the Whiteshell Reactor No. 1 (WR-1) since 1965. Euratom recently announced criticality of the Eссор test reactor. The Soviet Union reported on the operation of the Arbus reactor at the 1964 Geneva Conference on the Peaceful Uses of Atomic Energy.

The principal uncertainties associated with the use of organic coolant have involved its stability under exposure to heat and radiation, its deposition of films at the surface of fuel elements, and its processing requirements.

The experience obtained has shown that coolant stability does not present a problem under the operating conditions of the HWOCR. The cost of coolant make-up is practically negligible when a coolant recovery system is used to reclaim coolant from the decomposition products formed. The coolant make-up cost penalty without such a system is small enough not to detract from the economic potential of the concept.

Control of film formation requires the same attention to coolant processing and purity control as is required in water cooled reactors, though the processing methods used are different. While a detailed understanding of the mechanism of film formation in organics (or, for that matter, crud and scale formation in water) remains to be established, the experience from the operating organic facilities has demonstrated satisfactory control of film formation under HWOCR

conditions. The WR-1 has now been in operation for over a year and a half with an excellent on-line record. The operation of the U-3 loop with the two HWO CR irradiation experiments, U-305 and U-305 Alternate 1, over the past two years has shown highly satisfactory performance with no operating difficulties associated with the use of organic coolant. Though the interpretation of the data has not always been clear because of the nature of these experiments, the evidence shows that the operating exposures already achieved are within the range of acceptability for the HWO CR, with continued operation expected to provide a demonstration up to the actual design burnups.

The operation of the PNP F has shown problems which, however, appear to be exclusively associated with specific aspects of the design of that organic moderated and cooled plant, and would in no way apply to the performance of the HWO CR. Corrective measures are presently being taken to make the required modifications to the reactor and get the plant back into operation.

The presentation which follows provides a summary evaluation of the status of the technical understanding of the behavior of organic coolants in an HWO CR environment.

2. Coolant Selection

A number of organic compounds and mixtures of compounds have been evaluated for use as coolants in the organic cooled HWR concept. Considerations of radiolytic and pyrolytic stability and film formation tendencies, as well as physical properties (principally viscosity and vapor pressure), availability, and cost have generally limited the choice of organic coolants to mixtures of polyphenyls and hydrogenated polyphenyls, specifically, the terphenyl isomers and partially hydrogenated terphenyl isomers, and their decomposition products. A large body of information on the properties and behavior of these mixtures has been developed from OMRE and PNP F operating experience and the technology established through the research programs conducted in the United States, Canada, and Europe. (24, 25)

For the HWO CR application, the coolant choice can be further narrowed to low melting mixtures consisting largely of ortho- and meta-terphenyls

(Santowax OM* and Santowax WR*) and partially hydrogenated terphenyls (HB-40*) since it is highly desirable that the coolant be liquid at the operating temperature of the D₂O moderator.

3. Physical Properties

The physical properties of polyphenyl coolants have been extensively studied throughout the world and are summarized in a recent report.⁽²⁶⁾ Table 14 presents some of the typical properties of the various terphenyl mixtures. In general, physical property data are sufficiently well established for engineering design purposes.

4. Coolant Stability

Terphenyl coolants decompose by radiolysis and pyrolysis to give a wide variety of products, which have been classified according to their volatility as gases (H₂, CH₄, C₂H₆, etc.), LB (low boilers, more volatile than diphenyl), HB (high boilers, less volatile than p-terphenyl), and IB (intermediate boilers, similar to the original coolant in volatility). Gases and LB are minor components (about 2% by weight of the total decomposition products). HB is the major product; thus, the rate of formation of HB is roughly equal to the coolant make-up rate. The presence of HB has both desirable (lowers makeup requirements) and undesirable (lowers heat transfer performance) effects on reactor operation, so that there is an optimum HB concentration. Based upon comparative costs, an HB concentration of 10 to 20% appears to be optimum for an HWOCR plant.

There is general agreement that the pyrolysis data on unirradiated terphenyls can be expressed using first order kinetics. While the pyrolysis rate of unirradiated coolant is quite low at the operating temperature of an HWOCR, it has been shown that pyrolysis is enhanced by a factor of 4 to 5 by the occurrence of "radiopyrolysis" or increased thermal decomposition of irradiated terphenyl coolant. Although this phenomenon is not completely understood, the radiolysis contribution still outweighs the combined effects of pyrolysis and radiopyrolysis.

*Monsanto Company trade names. Santowax OM is a coolant containing approximately 3 wt % diphenyl, 55 to 69 wt % o-terphenyl, 23 to 37 wt % m-terphenyl, and 1 wt % p-terphenyl which has a liquidus point of 175 to 190°F. Santowax WR has a similar melting point range but has an ortho-to-meta ratio of 1 to 2. HB-40 is a 40% hydrogenated mixture of terphenyls.

TABLE 14
TYPICAL PHYSICAL PROPERTIES OF ORGANIC COOLANTS

Coolant	HB Con- tent (wt %)	Density (gm/cm ³)			Viscosity (centipoise)			Specific Heat (Btu/lb-°F)			Thermal Conductivity (Btu/hr-ft-°F)			Liquidus Tempera- ture (°F)	Flash Point (°F)	Flame Point (°F)	Auto Ignition Temperature (°F)
		600°F	700°F	800°F	600°F	700°F	800°F	600°F	700°F	800°F	600°F	700°F	800°F				
Santowax OMP	0	0.870	0.823	0.776	0.36	0.27	0.20	0.566	0.599	0.631	0.0659	0.0623	0.0588	~350	375	460	>1000
	10	0.885	0.839	0.793	0.42	0.30	0.22	-	-	-	-	-	-				
Santowax OM	0	0.859	0.813	0.765	0.31	0.23	0.17	0.568	0.604	0.640	0.066	0.064	-	~185	342	370	1072
	10	0.873	0.828	0.779	0.39	0.29	0.23	-	-	-	-	-	-				
Santowax WR	0	0.877	0.832	0.780	0.33	0.23	0.17	-	-	-	-	-	-	~195	380	420	1075
	10	0.878	0.834	0.785	0.36	0.26	0.19	-	-	-	-	-	-				
Santowax DOM	0	0.820	0.766	0.708	0.28	0.20	0.16	-	-	-	-	-	-	~100	-	-	1110
	10	0.838	0.785	0.730	0.32	0.24	0.20	-	-	-	-	-	-				
HB-40	0	0.788	0.737	0.682	0.36	0.26	0.16	0.65	0.70	-	0.065	0.065	-	< 32	367	396	706
	10	0.799	0.750	0.694	0.41	0.29	0.20	-	-	-	-	-	-				

Ref: AI-CE-15 "Physical Properties of Some Polyphenyl Coolants," H. Mandel (April 15, 1965), with exception of specific heat of HB-40 and liquidus temperature data.

Radiolysis rates are relatively well established at temperatures of up to approximately 700°F. There is general agreement on the effect of various parameters in this temperature range:

- 1) The radiolysis reaction rate can be approximated by second order kinetics.
- 2) The rate of HB formation is approximately the same for the different terphenyl mixtures. Thus, there is no significant difference in make-up rate between Santowax OM (high ortho content) and Santowax WR (high meta content).
- 3) The effect of temperature on the decomposition rate is small in the temperature range of interest, corresponding to an activation energy of ~ 1 kcal/mole.
- 4) There is a substantial "fast" neutron effect; that is, for a given amount of energy deposited in the coolant, fast neutrons cause much more damage (by a factor of 4 to 5) than do electrons or gamma rays.

Interpretation of the results at the higher temperatures ($>700^\circ\text{F}$) is complicated by the difficulty in separating radiolysis from pyrolysis and radiopyrolysis effects. In general, the following can be stated:

- 1) The order of the reaction may be somewhat different; however, second order kinetics can be used with little error.
- 2) Recent AECL work indicates an appreciable dose rate effect at the higher temperatures.⁽²⁵⁾ At 750°F, the radiolysis yield was found to decrease with increasing dose rate. The net result is that the effect of temperature on the decomposition rate is less at the high dose rate expected in the HWO CR (>1.0 w/gm) than at the lower dose rates observed in some of the other reactors.

Canadian data on the decomposition rate of HB-40 show that the total coolant degradation rate is comparable to that of Santowax OM at temperatures of up to 750°F.⁽²⁵⁾

5. Film Formation

a. Introduction

Film formation is the single most critical problem associated with the use of organic coolants. In general, the work on film formation has progressed sufficiently to permit the design of organic coolant systems with reasonable confidence in the feasibility of satisfactory control of film formation. The film on the fuel element surfaces is similar in effect to crud formation in water reactors or scale in conventional boilers. The general effect is to increase cladding

surface temperature (due to decrease in heat transfer performance) and pressure drop (due to increased surface roughness and flow restriction).

b. Experimental Results

Early work on film formation, including OMRE operating experience, has been summarized in References 24 and 25. It was concluded from the experience at the OMRE, in-pile tests at the X-7 and U-3 loops, and extensive out-of-pile experimentation, that coolant oxidation and the presence of chlorine were important causes of fouling, and that film formation rates could be controlled at acceptable levels by maintaining adequate coolant purity control, as indicated by tests such as ash, iron, chlorine, carbonyl oxygen analyses, and especially the MST (Membrane Stain Test – a measure of colloidal particle content) and SPFT (Small Probe Fouling Test). Distillation, clay bed adsorption, and glass spool filtration have been shown to be effective methods for maintaining the required coolant purity.

In PNPf operation, coolant purity standards were met by distillation and filtration, but exposure of the coolant to atmospheric oxygen could not be entirely avoided during shutdowns. Film formation on fuel cladding surfaces was generally about as predicted, but there was a massive formation of coke-like material in the relatively stagnant (with internal natural convection recirculation) moderator regions.

Some additional information on film formation has been obtained from the MIT reactor organic in-pile loop, the Euratom studies, and an experiment on the effect of velocity in the E-2 Loop of the NRX Reactor. The most important new results, however, have come from the U-3 loop experiments in the NRU Reactor and the WR1-951 experiment in the WR-1 Reactor.

Two irradiations, U-305⁽²⁷⁾ and U-305 Alternate 1, have been carried out in the U-3 Loop of the NRU Reactor under the HWO CR Program. The fuel assemblies in both were substantially identical, consisting of 18-rod bundles, the lower (upstream) half consisting of two Zr-4-clad UO_2 assemblies, and the upper (downstream) half of three SAP-clad UC assemblies. Table 15 presents a description and operating characteristics of the U-305 tests. The coolant was Santowax OM with an HB content of 10 to 20%. The U-305 experiment was primarily a fuel test, and, as such, variations in coolant composition which occurred were unintentional. The Alternate-1 experiment is primarily a coolant

TABLE 15
DESCRIPTION AND OPERATING CHARACTERISTICS
OF THE U-305 EXPERIMENT

	UO ₂	UC
<u>Fuel Assembly Description</u>		
Fuel system	Zr-4-Clad UO ₂	SAP-clad UC
Fuel geometry	18-Rod bundle	18-Rod bundle
Fuel diameter (in.)	0.522 OD - 0.184 ID	0.499
Initial radial gas gap (in.)	0.0030	0.0025
Carbon content (%)	—	4.82 - 5.3
U ²³⁵ enrichment (%)	1.12 and 1.52	1.5
<u>Operating Characteristics</u>		
Coolant material	Santowax OM + 10 - 20% HB	
Coolant Velocity (ft/sec)	30	
Coolant Temperature (°F)	575 - 650	
Cladding hot spot temperature (°F)	860	850
Maximum fuel temperature (°F)	3,000	2,000
Maximum linear heat rate (kw/ft)	14.7	22.7
Maximum heat flux (Btu/hr-ft ²)	304,000	326,000

test, and deliberate changes in coolant treatment have been made. The U-305 was operated to a maximum UC burnup of 10,000 Mwd/MTU (exclusive of end peaking) over the period from July 29, 1965 to April 16, 1966 (when it was removed from the reactor because of the appearance of defects in the zircaloy cladding of several Zr-4-clad UO₂ fuel rods). The Alternate-1 experiment was inserted May 11, 1966, and as of June 1, 1967, had accumulated a maximum UC burnup of over 12,000 Mwd/MTU (exclusive of end peaking).

The principal purification method at the U-3 loop is clay bed adsorption, and its effectiveness has been further demonstrated. Prior to the insertion of the U-305 experiment into the U-3 loop, a program to improve the coolant purity was conducted. Two large clay adsorption columns were used in series. As the

first (upstream) column became saturated (break-through of Na-24 activity is a good indication of this), it was valved out, the adsorbent was changed, and the fresh column was valved in downstream of the other. In two weeks of this treatment, the Na-24 activity and MST were reduced by a factor of 100. At the time the U-305 experiment went to power (July 29, 1965), coolant quality as indicated by all tests was comparable to that at the PNPf. The same treatment was continued, though column changes became much less frequent, and further substantial improvement was observed. Levels of coolant impurities by late August 1965, were the lowest ever observed in an operating organic cooled loop or reactor. All corrosion product activities were below the detection limit of 10 dpm/g. The MST value (in units of $10^5 \times \text{absorbance/mg}$) was about 10, compared with about 80 at startup and about 8000 before the purification program began. The electrical conductivity was reduced from 2000 picomho/cm on July 15, 1965, to 5 picomho/cm in mid-August. Carbonyl oxygen ranged from 1 to 8 ppm.

The same general purification practices were continued throughout the U-305 experiment and most of the Alternate-1 experiment. Clay columns were changed about every three weeks. The MST value was usually maintained between 5 and 20. The electrical conductivity rose steadily between column changes, typically ranging from 20 to 60 picomho/cm. While some inleakage of air did occur, the carbonyl oxygen content was normally somewhat lower than at the PNPf, typically about 50 ppm, with extreme values of 1 and 174 ppm.

In the late stages of the Alternate-1 experiment, a test was made of the effect of less frequent clay column changes. During most of this test, only a single clay column was used rather than two in series. After a period of over five months without a column change, no cladding temperature rise had been observed, indicating the feasibility of reduced clay bed adsorption purification. The MST value had increased from 10 to about 70, and the electrical conductivity has continued to increase steadily.

Figures 16 and 17 present the temperature histories of two "typical" cladding thermocouples of the U-305 and U-305 Alternate-1 experiments. The interpretation of clad temperature changes observed in the U-3 tests has been complicated by some uncertainty in flow rates, some variation in power and power distribution, and by early chromel-alumel thermocouple failures leading to the use of chromel-stainless steel junctions. Within these limitations, the following conclusions can be drawn:

Figure 16. U-305 Cladding Hot Spot Temperature History
(Thermocouple No. 21)

Figure 17. U-305 Alternate 1 Cladding Hot
Spot Temperature History
(Thermocouple No. 16)

1) There was no steady cladding temperature increase which would indicate a continuous film formation process.

2) There were several periods of relatively rapid temperature increase. The two most pronounced incidents during the U-305 experiment corresponded to periods of accidental low water content (water is normally maintained at 200 to 300 ppm to prevent hydriding of the zirconium cladding and pressure tube).

3) Two deliberate tests with low water content were made in the Alternate-1 experiment. Both appeared to show a somewhat smaller effect than that observed in U-305. This seems to indicate that the effect of water is to counteract the effect of an impurity (possibly associated with oxygen compounds), the concentration of which had already been greatly reduced. It is concluded that maintenance of the water content in the range of 200 to 300 ppm plays an important role in controlling film formation.

4) Comparison of thermocouple behavior at different temperatures confirmed an expected positive temperature coefficient of film formation, but quantitative evaluation was difficult.

5) Decreases in clad temperature occurred over long periods of normal water content. This could be due to increasing thermal conductivity or surface roughness of the film, changes in power distribution, or even film removal.

Film thickness measurements were made on flakes removed from the lowest UC fuel bundle of the U-305 experiment (the bundle with the highest burnup — about 10,000 Mwd/MTU exclusive of end peaking).⁽²⁸⁾ The thickness of the flakes varied by an order of magnitude. The rod from which each flake came is known, but its position on the rod is not. A definite difference between the inner and outer ring of fuel rods can be seen from the data in Table 16 (all thicknesses are in mils). This difference is another confirmation of the positive temperature coefficient of film formation since the inner rods experience a higher fast neutron flux than the outer rods, but operate at a lower cladding surface temperature (by approximately 50°F). It is also of interest that the areas of low flow (points where bundles mated) showed no evidence of film formation. In fact, machining marks were clearly discernible.

Additional information on film formation will be obtained with the completion of the Alternate-1 experiment and the hot cell examination of the fuel assemblies.

The magnitude of the pressure drop increase due to film formation is not completely defined. The pressure drop history of the U-305 Alternate-1 experiment has been closely monitored for changes in frictional pressure drop. The data indicate that there may have been a gradual increase of approximately 10% (90 to 100 psi) over the life of the experiment.

TABLE 16
U-305 FILM THICKNESS MEASUREMENTS

Ring	Outer	Inner
Number of Measurements	19	8
Range	0.2 - 2.3	0.2 - 1.8
Median	1.2	0.5
Mean	1.22*	0.65*
Standard Deviation	0.55	0.52
Difference of Means		0.57
Standard Deviation of the Difference		0.19
Probability That Means are Different		99%

*The measured values are in good agreement with the calculated value at the hot spot of approximately 1 mil based on a thermal conductivity of 0.5 Btu/ft-°F-hr.

The WR1-951 experiment, consisting of two 19-rod fuel assemblies of SAP-clad uranium carbide, has been operated in the WR-1 for a period of approximately one year. During the first five months of operation (coolant inlet temperature of 635°F, cladding temperatures ranging from 700 to 750°F), a gradual increase in the film formation parameters was observed. Following an increase in coolant purification rate and a decrease in inlet coolant temperature to 565°F, there was an indication of a decrease in heat transfer resistance at the cladding thermocouples (the decrease being greater on the hotter thermocouples). No significant change in hydraulic performance of the element was observed during the period of increasing film formation parameters.

The film formation parameters continued to decrease when the coolant inlet temperature was raised back to its original level of 635°F early in January 1967.

When operation was resumed in March 1967 following a February shutdown, the coolant inlet temperature was raised to 650°F. A small increase in film formation parameters was observed. However, the measured cladding temperatures have remained below 775°F (peak calculated temperature of ~820°F), and as of June 1, 1967, the experiment had accumulated a peak UC burnup in excess of 8000 Mwd/MTU. During this period of high coolant temperature

operation, there has been a gradual increase in the electrical conductivity of the coolant and a large increase in MST value.

As an additional point of interest, AECL has conducted a limited examination of four of the WR-1 driver fuel assemblies. Despite the fact that the coolant quality in the WR-1 had not met the standards considered acceptable for an HWO CR plant, only thin, almost immeasurable, films were observed at the surface of the cladding of these elements.

In summary, all the experience to date has confirmed the fact that coolant purity is of prime importance in the control of fuel element film formation. Distillation, clay bed adsorption, and glass-spool filtration have been shown to be effective methods for maintaining the required coolant purity. Water appears to be a desirable impurity; however, the exact mechanism of its effect is not understood at this time.

The dependence of film deposition rate on the various key parameters is still not understood quantitatively. There is a direct dependence on radiation dose, an inverse dependence on velocity, and a positive temperature effect, but the exact form of the dependence remains to be established.

Methods of measuring the film formation potential (or "foulant" concentration) of the coolant need further development. The pyrolytic capsule fouling test (PCFT) can be made useful for clean coolants by using combustion analysis for carbon as a measure of film thickness (far more sensitive than scraping and weighing). This also applies to the small probe fouling test (SPFT), and would shorten the time required for a determination. An interesting technique now being tested at WR-1 is the direct measurement of the concentration in the coolant of particles larger than 0.2μ , using a high temperature silver membrane filter. There are some indications that this measurement correlates with the SPFT value. The electrical conductivity, which is strongly affected by the use of clay bed adsorption, may also be related to film formation potential.

c. Coolant Chemistry

Coolant radiolysis products include active compounds, such as olefins, fluorene, and free radicals, which readily react with oxygen.

The importance of oxygen compounds in film formation and probably in coking, seems well established. It also appears from experience that it is very

difficult to exclude oxygen completely (even new coolant contains some oxygen compounds). Operating coolant normally contains a variety of oxygen compounds, only a few of which have been identified and most of which may be quite harmless. Little is known of the kinetics of coolant oxidation and of the interconversion and removal of the oxygen compounds.

In AECL and Euratom loop experiments, chlorine concentrations substantially higher than normal (above 5 ppm) have been observed to cause formation of a characteristic film, largely inorganic and containing iron crystals. Crystal growth indicates that the iron is carried in solution, possibly as FeCl_2 . In systems containing zirconium, chlorine also promotes hydriding. The chlorine content of the coolant decreases with time, in part by loss of volatile chlorides such as HCl , and in part by deposition on the walls. Because of this persistent wall deposit, a single incident of high chlorine content could have serious long-range effects. It is not certain whether there is a "safe" concentration of chlorine, or whether it may have some effect even at concentrations below 1 ppm, such as usually prevail in-pile.

The chlorine content of the coolant system may be reduced by degassing, by treatment with a palladium-alumina catalyst, or by reaction with metals such as magnesium or zirconium. However, because of wall effects, none of these methods is very rapid, and strict control to prevent the entry of chlorine into an organic coolant system is the easiest way to maintain a low chlorine concentration. Little is known about the effect of other halogens, which are less probable contaminants in the organic coolant. It is expected that they would behave in a similar manner as chlorine.

6. Coking

Coking is defined here as the formation of very high molecular weight compounds resulting from extended exposure of the coolant to heat and/or radiation without continuous or periodic replacement of the decomposition products with new coolant. The most important observations on coking in organic systems include the following:

- 1) "Clinkers" were observed in the OMRE at points of low flow and high temperature.

- 2) No coke was formed in six years in the OMRE moderator region, with velocities of about 4 ft/sec, even though coolant purity was often inferior to that maintained at the PNPF.

3) In an AECL decomposition rate experiment in the E-2 loop in NRX, with velocities of about 0.005 ft/sec, the irradiation capsule was found to be nearly full of coke.

4) A solid, roughly spherical mass of coke formed between the fuel assemblies in the PNPF outer moderator region, where the calculated average nominal velocity was 0.05 ft/sec, but was in a direction opposed to convective flow, thus, probably resulting in internal recirculation and effective stagnation of coolant in this region.

5) In the PNPF inner process tubes, with a downward flow at 0.05 ft/sec, coke formed primarily at the bottom of the tubes.

6) PNPF inner process tubes, in which control rods were located and in which velocities ranged from 0.5 to 5 ft/sec, showed no coking.

7) The MIT loop in-pile sections, with velocities of 1.5 and 7 ft/sec, showed film deposition but no coke.

While very little is known about radiolytically (as opposed to pyrolytically) induced coking, the possibility of coking can easily be minimized in the HWOCR design by eliminating all flow restriction or captive coolant in the high flux region of the reactor.

7. Analytical Techniques

Analytical techniques have been developed for determining, evaluating, and controlling coolant chemistry and quality. These techniques are described in References 24 and 25. Additional development is required to firmly establish the coolant purity criteria requirements and the techniques to be used to maintain conformance to these criteria (preferably on a continuous basis).

8. Coolant Processing

Technical evaluation of the purification system for an HWOCR plant indicates that the optimum approach to maintaining coolant purity is to utilize degasification, distillation, adsorption, and filtration. It is further indicated that the coolant make-up cost can be reduced significantly by hydrocracking the high boilers to useful coolant which is returned to the reactor.

a. Degasification

Low pressure or atmospheric pressure degasification is utilized to remove gases and low boilers from the coolant. Degasification systems have been designed and operated successfully at the PNPF, the WR-1, and the X-7 and U-3 loops. Suitable design methods and procedures are available for designing the degasification system for the HWOCR plants.

b. Distillation

Side stream distillation has been employed at the OMRE and the PNPf to remove high boilers from the coolant. The low pressure distillation system consists of a simple adiabatic flash chamber followed by a packed stripping section which yields an essentially HB-free overhead product and bottoms (waste stream or hydrocracker feed) consisting of 98% HB. Suitable design methods and procedures are available for the design of this type of distillation system for the HWOcR plants. Care must be taken to maintain leak tightness of the equipment to prevent air in-leakage to the coolant stream.

c. Adsorption

Experiments have shown that adsorption is an effective process in lowering the film forming potential of "dirty" irradiated coolant. Some 40 commercial adsorbents have been evaluated under previous programs in screening tests to identify the most promising ones for coolant purification. Attapulgu clay was chosen for both economic and performance reasons. Small fixed-bed tests showed that adsorption reduced the film forming tendency of OMRE Core II coolant. Subsequently, two adsorption beds using attapulgu clay were operated as part of the impurities removal loop at the OMRE, in conjunction with Core III operation. Effluent-to-adsorber weight ratios of 8000 were achieved before the adsorber beds required replacement. Operation of the WR-1, the X-7 Loop (NRX Reactor), and the U-3 Loop (NRU Reactor) with attapulgu clay adsorber columns further substantiated that adsorption was an effective method for controlling the film forming tendency of irradiated coolant. Effluent-to-adsorbent weight ratios greater than 50,000 have recently been achieved in the U-3 Loop with no apparent adverse effect on film formation.

Thus, the technical feasibility of clay bed adsorption purification has been demonstrated, but the engineering design aspects, particularly as they pertain to large systems, have not been thoroughly established. In addition, certain performance characteristics such as adsorptive capacity of the adsorbent as a function of coolant flow rate and temperature, and regeneration of the adsorbent, as well as any required pre-treatment and quality control must be thoroughly established to provide a reliable basis for the economic evaluation of large-scale adsorption systems.

d. Filtration

Operating experience at the OMRE has shown that glass-fiber spools gave satisfactory filter performance with a reasonable operating life at a low cost. As a result, glass spool filters were installed at the PNPF. Satisfactory filter performance was obtained, although full-flow filter lifetime proved to be somewhat shorter than anticipated and replacement was required every few months. In both reactor systems the operating coolant temperature was about 600°F, well below the temperature limit of 750°F established by the filter manufacturer. Leaching tests have been run at temperatures of up to 600°F. However, there is little available experience or test data on the mechanical stability and amount of material leached from the glass spools by high temperature (750°F) organic coolant.

e. Catalytic Hydrocracking

Hydrocracking of polyphenyl materials using a number of catalysts has been studied by the Phillips Petroleum Company and the Monsanto Company utilizing bench scale equipment.⁽²⁴⁾ Except for a few runs, all experiments involving hydrocracking of high boilers employed about 50% p-xylene or n-hexane as a solvent. A few runs were made using 28% HB in Santowax DOM. The difference between these runs and those with the xylene solvent suggests that xylene may have played some part in the reaction. The best data for the solvent-free runs showed that 98% of the HB converted yielded usable coolant. This was observed with NiO-Al₂O₃ catalyst at a space velocity of 0.66 vol/vol/hr. Seventy percent of the HB charged was converted in one pass.

Tests on radiolytic and pyrolytic stability were made only with coolant reclaimed using the xylene solvent process. Electron irradiations showed that the reclaimed coolant gassed more, but polymerized less, than Santowax OMP, probably due to the presence of more alkyl groups. In-pile irradiation gave similar results. However, pyrolytic stability of this reclaimed coolant could be a problem. At 825°F, the polymer production rate was 5 to 8 times greater than that of the Santowax OMP control. The gas production rate was 30 times greater. Because of the generation of light ends, reclaimed coolant appears to exhibit a higher vapor pressure than the basic coolant.

The status of the hydrocracking process and of the associated coolant physical properties available for the HWOOCR design is as follows:

- 1) Conversion of HB to useful coolant by catalytic hydrocracking has been demonstrated with a good yield.
- 2) Tests have not been conducted on the hydrocracking of concentrated HB in reactor coolant.
- 3) Reclaimed coolant may exhibit a higher vapor pressure than the standard coolant.
- 4) Data on coolant chemistry (including film formation rate) and physical properties are presently not available for the hydrocracked coolant proposed for the HWOOCR.

It is to be emphasized that while reclamation of useful coolant through hydrocracking appears to be a highly attractive process from the economic viewpoint, it is not essential to the technical performance or economics of the HWOOCR.

9. Heat Transfer and Hydraulics

Extensive measurements have been made on the forced convection heat transfer characteristics of organic reactor coolant. The data have been generally correlated ($\pm 10\%$) by a modified form of the Dittus-Boelter Equation: (25, 29)

$$\text{Nu} = 0.023 \text{ Re}^{0.8} \text{ Pr}^{0.4}$$

The applicability of the equation for predicting the performance of special geometries such as finned fuel rods, has been confirmed by the performance of fuel assemblies under reactor operation.

The pressure drop through the fuel assemblies can be accurately predicted using conventional expressions and the equivalent diameter concept. The greatest uncertainties in determining the pressure drop in an HWOOCR fuel channel are those attributable to end plates, bundle alignment, and roughness caused by film formation. End plate losses and losses caused by bundle misalignment are determinable by experimental measurement in out-of-pile water loops. The magnitude of the pressure drop increase due to film formation is currently not completely defined. The pressure drop history of the U-305 Alternate-1 experiment has been closely monitored for changes in frictional pressure

drop. The data indicate that there has been a gradual increase of approximately 10% (90 to 100 psi) over the life of the experiment. This is well within the limits utilized in the design.

The HWO CR plant is not critical heat flux limited. Critical heat flux or DNB ratios are in excess of four using the relationship: (25, 30, 31)

$$(Q/A)_c = 129,000 + 11 (\Delta T_{sub}) G^{0.8} ,$$

where

$(Q/A)_c$ = critical heat flux (Btu/hr-ft²),

ΔT_{sub} = subcooling, $T_{sat} - T_{bulk}$ (°F), and

G = mass velocity (lb/sec-ft²).

It should be noted that while this is one of the most conservative correlations available, the majority of the burnout heat flux data were obtained in simple circular geometries. There is a lack of data on more complex geometries such as fuel element assemblies with finned surfaces. Some available data indicate that the actual critical heat flux values for finned geometries may be lower by as much as a factor of two. However, this does not present a problem in the planned reactor designs, which are in no way limited by burnout heat transfer considerations.

Calculation of coolant and cladding surface temperatures in the HWO CR fuel assemblies consider local flows and the effects of mixing in the flow channels. Based on extrapolation of the data in the literature on somewhat similar hydraulic geometries, a value of 64% mixing was used in the HWO CR reference designs. Subsequent tests conducted under the HWO CR development program on mockups of the HWO CR fuel assemblies indicate that coolant mixing actually exceeds 90%.

10. Summary

An evaluation of a number of organic coolants resulted in the selection for the HWO CR of a low melting mixture consisting largely of ortho- and meta-terphenyl and their decomposition products, such as Santowax OM or Santowax WR with 10% high boilers.

The information required to assess coolant decomposition and make-up rates is presently available. Additional work is required to more clearly establish the kinetics and mechanics of radiolysis and radiopyrolysis at temperatures above 700°F.

Film formation is the most critical problem associated with the use of organic coolants. Operating experience at the OMRE, the PNPf, the U-3 Loop, and the WR-1 has shown that film formation can be satisfactorily controlled by controlling coolant purity. Distillation, clay bed adsorption, and filtration have been shown to be effective methods for maintaining the required coolant quality. In general, the extensive work conducted on film formation on heat transfer surfaces has progressed sufficiently to allow the design of organic in-pile cooling systems with reasonable confidence in the feasibility of satisfactory control of film formation. Coolant purity criteria and analytical techniques have been established to some extent but require considerably more development.

The experience with coke formation in the PNPf moderator region has emphasized the necessity for avoiding extended residence of coolant in a high flux region. This is not a problem in the HWOCR because the organic is used only as the coolant and is, therefore, rapidly circulated through the coolant channels.

The available data on the heat transfer and hydraulic characteristics of fuel assemblies in organic coolant are adequate for engineering design. Additional information is required on the effects of film formation on pressure drop and possibly on critical heat flux in actual fuel assembly geometries.

B. FUEL ASSEMBLY

1. Introduction

The HWOCR reference design is based on the use of the SAP-clad cast hyperstoichiometric uranium carbide fuel system. A typical HWOCR fuel string consists of five fuel assemblies approximately 44 in. long made up of 37 fuel rods of finned SAP-clad UC, with the UC slugs approximately 0.5 in. in diameter (Figure 6). A helium-filled gap separates the fuel from the cladding. A preliminary conceptual design study has also been conducted on the use of zircaloy-clad uranium metal fuel in the form of concentric annular rings for

application to a natural uranium fueled HWOCR. Most of the presently available information relates to the uranium carbide technology; the uranium metal concept has been considered only for future application following successful completion of the required development work on this fuel.

2. Uranium Carbide Fuel System

The uranium carbide characteristics of interest to the HWOCR are its high density, high thermal conductivity, good dimensional stability, ability to retain fission gases, and compatibility with both coolant and cladding. Because of the high thermal conductivity of the SAP cladding, an increase in effective heat transfer area is attainable through finning. This fuel system therefore allows operation at a high power density.

The technical uncertainties associated with the SAP-clad UC fuel system relate to the fuel growth, fission gas release, fuel spalling, fuel-cladding compatibility, cladding ductility, sensitivity to cladding defects, and general irradiation experience.

3. HWOCR Irradiation Experiments

The most important experimental work conducted in the fuel development area has been the irradiation, under HWOCR conditions, of two almost identical fuel strings (U-305 and U-305 Alternate 1) in the U-3 loop of the NRU reactor at Chalk River, and the irradiation of a third string (WR1-951) in the WR-1 reactor at Whiteshell. These irradiations are described from the coolant point of view in Section III-A of this report. Table 17 presents the basic data on the U-305 experiment. An isometric view of this experiment is shown in Figure 18.

The U-305 fuel string, consisting of three 18-rod SAP-clad UC and two 18-rod Zr-4-clad UO_2 fuel assemblies, was irradiated under conditions closely approximating those of the HWOCR. It reached a maximum UC burnup of 10,000 Mwd/MTU exclusive of local end peaking regions which reached approximately 12,000 Mwd/MTU. Defects in one of the Zr-4-clad UO_2 fuel assemblies required termination of the experiment after nine months of irradiation. Both the data obtained during the irradiation and the subsequent examination and measurements carried out in the hot cell clearly demonstrated that the SAP-clad UC fuel system had performed as predicted.^(27, 28)

7660-5128

Figure 18. U-305 Experiment Fuel String

TABLE 17
DESCRIPTION AND OPERATING CHARACTERISTICS
OF THE U-305 EXPERIMENT

	UO ₂	UC
<u>Fuel Assembly Description</u>		
Fuel system	Zr-4-Clad UO ₂	SAP-clad UC
Fuel geometry	18-Rod bundle	18-Rod bundle
Fuel diameter (in.)	0.522 OD - 0.184 ID	0.499
Initial radial gas gap (in.)	0.0030	0.0025
Carbon content (%)	—	4.82 - 5.3
U ²³⁵ enrichment (%)	1.12 and 1.52	1.5
<u>Operating Characteristics</u>		
Coolant material	Santowax OM + 10 - 20% HB	
Coolant velocity (ft/sec)	30	
Coolant temperature (°F)	575 - 650	
Cladding hot spot temperature (°F)	860	850
Maximum fuel temperature (°F)	3,000	2,000
Maximum linear heat rate (kw/ft)	14.7	22.7
Maximum heat flux (Btu/hr/ft ²)	304,000	326,000

The U-305 Alternate 1 experiment is presently in progress. As of June 1, 1967, it had completed over a year's irradiation, with a maximum estimated UC burnup in excess of 12,000 Mwd/MTU, exclusive of local end peaking effects, with no evidence of any operating problems.

A third experiment, U-305 Alternate 2, consisting of five SAP-clad UC fuel assemblies, has been fabricated and will be irradiated in the U-3 loop under AECL direction following completion of the U-305 Alternate 1 irradiation.

The WR1-951 experiment consists of two 19-rod SAP-clad UC fuel assemblies. It has now operated for approximately one year in the WR-1 reactor under conditions somewhat less severe than those of the HWOCR. As of June 1, 1967,

it had accumulated a maximum burnup of over 8000 Mwd/MTU with every indication of continued satisfactory performance.

An almost identical experiment, WR1-952, was installed in the WR-1 reactor in May 1967. Fabrication of another fuel string, WR1-954, consisting of seven fueled rods in each of two 19-rod fuel assemblies and designed to operate at higher power density than WR1-951 and WR1-952, has been completed. It is expected that irradiation of this experiment will be initiated some time later this year under AECL direction.

4. Dimensional Stability

In common with other fuel materials, uranium carbide experiences volumetric changes during irradiation. As the irradiation progresses, these changes result in the elimination of the radial helium gap between the fuel and the SAP cladding and a progressively increasing strain of the relatively low ductility SAP. It is expected that the limitation on the burnup capability of the SAP-clad UC fuel system is determined by the amount of strain the cladding can stand as a result of the growth of the fuel.

The dimensional changes which take place are of two basic types. At temperatures of up to approximately 2000°F, the fuel exhibits a volumetric expansion (termed "growth") attributable only to the accommodation of the fission products in the lattice. At temperatures above approximately 2000°F, the gaseous fission products coalesce and the resultant pressures lead to an additional expansion of the fuel (termed "swelling"). The experimental evidence indicates that the growth of uranium carbide is independent of temperature and directly proportional to the accumulated burnup at burnups of up to 30,000 Mwd/MTU and temperatures below those at which swelling begins. Above approximately 1400°F the fission gas release is mostly diffusion controlled and therefore highly dependent on temperature. Both the swelling and fission gas release are affected by the carbon content of the fuel, with an apparent minimum for a slightly hyperstoichiometric composition.

As mentioned above, the helium-filled radial gap between the uranium carbide fuel and the SAP cladding of the HWOCR fuel element decreases with time as the fuel grows, and finally disappears completely. The fuel is therefore at its maximum temperature near the beginning of life when the fission product

inventory in the fuel is low. The temperature decreases from then on as the thermal resistance of the radial gap goes down. Under these conditions, the existing data indicated that the most probable fuel growth rate would be approximately $2\% \Delta V/V$ per 10,000 Mwd/MTU, with a conservative value of around $2.5\% \Delta V/V$ per 10,000 Mwd/MTU.

The density and dimensional measurements conducted in the hot cell on the fuel from the highest burnup SAP-clad UC fuel assembly of the U-305 experiment have confirmed these data. Thirty-one density measurements showed a mean volumetric growth of 1.90% per 10,000 Mwd/MTU, with a standard deviation of 0.106% . One hundred and seventy-four diameter measurements showed a mean diametral growth of 0.71% per 10,000 Mwd/MTU, with a standard deviation of 0.127% . Assuming isotropic growth, this would correspond to a mean volumetric growth of 2.13% per 10,000 Mwd/MTU, with a standard deviation of 0.381% . The fuel rods on which the measurements were taken contained UC slugs with a graded carbon content ranging from approximately 4.85 to 5.2%. A very rough evaluation of the effect of carbon content appears to show a minimum growth at about 5.0%.

5. Fission Gas Release

The data available prior to the irradiation of the U-305 experiment indicated that hyperstoichiometric uranium carbide was capable of retaining more than 99% of the gaseous fission products generated at maximum fuel temperatures of up to 2400°F and burnups of up to 30,000 Mwd/MTU. Hot cell measurement of the fission gases released from the highest burnup fuel rods of the U-305 experiment confirmed that less than 0.1% of the gases generated had been released at a burnup of 10,000 Mwd/MTU (exclusive of local end peaking).

6. Fuel Spalling

It has been postulated that spalling of the uranium carbide fuel could result in chips lodging in the annular gap between the fuel and cladding and causing an excessive local cladding strain. It is known that UC does not chip as easily as UO_2 . Insufficient information, however, was available prior to the U-305 irradiation to allow an evaluation of the probability of UC spalling or chipping and its associated effects on cladding performance. While the U-305 hot cell examination has not shown any deleterious effects on the cladding attributable to such

spalling, one or more possible cases of UC chips imbedded in the cladding have been observed. Additional test data are required to provide complete assurance that UC spalling would not be a problem.

7. Chemical Reactions

a. SAP-Hyperstoichiometric UC Compatibility

A considerable amount of experimental work has been done on a study of the compatibility of hyperstoichiometric UC fuel with SAP cladding⁽³²⁾ (early screening tests had shown that hypostoichiometric and stoichiometric UC were incompatible with SAP at the temperatures of interest to the HWOOCR). The results, based on tests in excess of 12,000 hours, have in general shown that:

- 1) Hyperstoichiometric UC is compatible with SAP except for very slight local reaction at temperatures of up to 950°F.
- 2) At temperatures above 1050°F, hyperstoichiometric UC reacts extensively with SAP.
- 3) Increasing the thickness of the naturally occurring Al_2O_3 layer at the surface of the SAP by anodizing inhibits the high temperature reaction between SAP and hyperstoichiometric UC. The anodized layer, however, offers a delaying action only, as it is slowly destroyed with time.

All the available evidence indicates that there should be no problem of hyperstoichiometric UC interaction with SAP as long as the contact temperature is maintained below 950°F. Hot cell examination of the U-305 experiment has confirmed the absence of deleterious effects, though a small reaction zone (1 to 1.5 mils thick) was observed following over 5000 hours of operation at a contact temperature in excess of 850°F.

It does not appear that SAP-UC reactions will in any way limit the performance of HWOOCR fuel elements, since these elements are designed to operate at a maximum cladding temperature of less than 900°F which will exist only locally and for less than 2000 hours.

b. Uranium Carbide – Organic Compatibility

The fuel comes into contact with the coolant only in case of a cladding failure. Uranium carbide appears to be fairly inert when exposed to water-free organic coolant. Low density sintered uranium carbide showed no reaction in diphenyl and terphenyl over a period of five hours at 660°F.⁽³³⁾ Corrosion rates of between 6 and 110 mg/cm^2 -day have been reported for as-cast and annealed

hyperstoichiometric UC in Santowax R at 660°F. While the reaction of uranium carbide with water-free organic coolant appears to be negligible, it is not known how this would be affected by the presence of small concentrations (300 to 500 ppm) of water in the organic. Some bench-scale tests are required to determine the interaction between uranium carbide and organic containing up to about 1000 ppm, since the HWOOCR design is based on the presence of approximately 300 ppm of water in the coolant.

c. Uranium Carbide – D₂O Reaction

The use of uranium carbide fuel in a heavy water moderated reactor represents a potential source of energy release in the event of a major core accident. Under normal operation, the uranium carbide is separated from the D₂O by its cladding, the coolant, the pressure tube, and the calandria tube. The reaction rate for this material combination and the resultant energy release are not known well enough to allow a prediction of the consequences of a major core accident with any confidence. Some experimental data indicate that a UC-D₂O reaction would proceed rapidly with unirradiated UC at D₂O temperatures near 100°C (the reaction products were UO₂, hydrogen, and methane).⁽³³⁾ Irradiated UC, however, was found to be fairly passive with water at the same temperature.⁽³⁴⁾ Preliminary laboratory scale tests were conducted under the HWOOCR program with molten, unirradiated UC and water. While chemical reaction was noted, explosive-type behavior did not occur. Further testing with measurements of reaction rate, energy release, and reaction products for various UC-D₂O combinations is required.

8. Cladding Defects

The HWOOCR fuel assembly incorporates an initial radial gap between the UC and the SAP cladding to accommodate fuel growth. If a defect should develop in the SAP cladding or end plug, the organic coolant may leak into the gap, decompose, allow absorption of additional organic causing further decomposition under thermal cycling, and consequently swell in such a manner as to cause a cladding rupture.

Early experiments conducted in an out-of-pile loop with electrically heated UO₂ fuel rods with large (1/16 x 1/2 in.) defects indicated the probability of

coolant filling the gap between the fuel and cladding during 30 to 40 thermal cycles. A 175-cycle test was carried out with extensive damage to the cladding.⁽³⁵⁾

An in-pile experiment was conducted in the X-7 loop of the NRX reactor to test the behavior under irradiation of a UO_2 fueled specimen clad in a defective SAP sheath (0.020-in. hole drilled through the 0.020-in. -thick cladding) and cooled by recirculating Santowax OM.⁽³⁶⁾ This specimen was removed after an irradiation period of 19 days and 13 major thermal cycles at a nominal surface temperature of 860°F , a coolant temperature of 710°F , a coolant velocity of 30 ft/sec, and a surface heat flux of $360,000 \text{ Btu/hr-ft}^2$ - $^\circ\text{F}$. During the test, gaseous fission products were continuously released from the fuel to the coolant. The radionuclides introduced into the coolant were only those of appreciable volatility at the operating temperature of the fuel. The rate of release was of the same order of magnitude as that observed in cladding defect tests in water. The release rate of radioiodine, however, was considerably lower than that observed in water tests. The lower rate of radioiodine release to the coolant was attributed to absorption in the organic film around the fuel. The post-irradiation examination showed that the cladding had ruptured extensively.

AECL's SAP-clad hypostoichiometric UC fuel irradiation experiment X-721 in the X-7 in-pile loop at the NRX reactor defected at a peak burnup of 11,100 Mwd/MTU (as a result of high cladding strain, later measured as 0.55% over the fins, but estimated as 1.4% in the web between the fins). This experiment was removed from the reactor shortly after the first detection of the presence of fission products in the coolant. The hot cell examination showed that the organic coolant had entered the element through cracks in the cladding, and that decomposed organic had filled all the axial and diametral gaps and had bonded the individual fuel slugs together and to the SAP.⁽³⁷⁾

Operation of the U-305 experiment was continued for a period of 8 days following the initial sharp increase in radioactivity of the coolant. The subsequent hot cell examination showed that the failure had occurred in the cladding of the Zr-4-clad UO_2 fuel rods. Samples of coolant and cover gas taken during this period of operation showed the presence of xenon and krypton and their associated daughter products; the presence of radioiodine was not detected. With the exception of several additional radioactivity spikes, this final period of

operation of the experiment was normal from both the point of view of cladding temperature and pressure drop behavior. It included six series of reactor power cycles down to approximately 65% of full power and one reactor shutdown and startup. The hot cell examination showed cracks in the cladding of several of the UO_2 fuel rods, including two bulges on one rod; there was no evidence of rupture of any of the rods.

Further in-pile defect tests of SAP-clad UC fuel rods are required to establish the operating characteristics and capabilities of defected HWO CR fuel rods under conditions of thermal cycling.

9. Physical and Mechanical Properties

a. Uranium Carbide

Information on the physical and mechanical properties of hyperstoichiometric uranium carbide is relatively well established.⁽³³⁾ Adequate physical and mechanical property data are available for the analysis of expected reactor performance under HWO CR conditions.

b. SAP Cladding

SAP is a dispersion-hardened material consisting of a matrix of commercially pure aluminum with particles of aluminum oxide (Al_2O_3) dispersed within the matrix.⁽³²⁾ In the HWO CR design, a 10% Al_2O_3 dispersion has been specified for both the fuel cladding and the pressure tube material. The physical properties which make SAP attractive as a cladding material are its low neutron absorption cross section, high thermal conductivity, relatively high strength at elevated temperature, and excellent corrosion resistance to the organic coolant. Both the high thermal conductivity and low neutron absorption cross section make it possible to use an extended heat transfer surface (fins) most effectively. The disadvantages of SAP are its low strain capability under long-term high-temperature load conditions, the limited experience with its performance, manufacture, and fabricability (extrusion and welding).

The effects of irradiation on the short-term mechanical properties of SAP have been obtained from specimens which have been irradiated to about 1.2×10^{21} nvt ($E > 1.0$ Mev) at temperatures of 104, 527, and 788°F. No significant changes in properties have been observed at the higher temperatures between the irradiated and unirradiated specimens.⁽³²⁾ This exposure, however,

is somewhat lower than that anticipated for HWOOCR fuel cladding (about 3×10^{21} nvt). It is not expected that there will be any significant cladding property changes due to HWOOCR irradiation because none of the postulated damage mechanisms appears to be of significant consequence.

The limiting condition for HWOOCR fuel cladding is associated with the long-term strain capability of SAP. Substantial data on unirradiated uniaxial stress-rupture properties exist. These data have been correlated by a Larson-Miller correlation⁽³⁸⁾ with rather good results.⁽³²⁾ The longest presently available test period is 18,000 hours, which has been extrapolated by use of the Larson-Miller parameter to a period of 40,000 hours for cladding design. There are presently no data on the stress-rupture properties of irradiated SAP or SAP stressed under biaxial conditions.

Additional work is required to better define basic mechanical property data and to reduce the uncertainty levels associated with the use of SAP. Data are needed on the effects of biaxial stress fields and notches. Both out-of-pile and in-pile testing is required to obtain more information on strain-to-rupture capability at elevated temperatures and under low strain rate conditions.

10. SAP Cladding Manufacture and Quality Control

SAP is made by ball milling aluminum powder which is oxidized, compacted, sintered, and extruded into billet form. The billets are then extruded into tubes by direct extrusion. SAP can be satisfactorily joined by such processes as eutectic bonding, flash welding, magnetic force welding, and cold pressure welding. Silver eutectic bonding and magnetic force welding have been the principal methods used for making helium-leak-tight end closures in finned SAP fuel cladding. SAP cannot be joined by conventional fusion welding or brazing processes. During welding, the oxide segregates in the fusion zone and the weld strength approaches that of unalloyed aluminum. Conventional brazed joints are difficult, if not impossible, to produce and such joints are mechanically weak and generally not practical for elevated temperature applications.

The present methods of magnetic force welding and eutectic bonding still have their limitations. Developed techniques limit magnetic force welding to unfinned sections. This reduces the cross-sectional area of the SAP where it

is needed most. The eutectic bonding techniques developed thus far are applicable only to the bonding of aluminum end plugs to the SAP cladding. SAP end plugs are highly desirable for the HWOOCR load conditions.

The quality assurance required in the fabrication of SAP billets depends largely on a standard ultrasonic inspection for internal defects and a dye penetrant technique for surface-type defects. The problems associated with the inspection for variations in oxide distribution have not been entirely resolved. Macroetching the ends of billets can detect gross areas of oxide segregation, and work with eddy current methods appears to be promising for 100% inspection of billets on a continuous basis.

Finned cladding tubes have been fabricated by hydraulic extrusion and by impact extrusion in a variety of shapes and sizes with both 7 and 10% SAP material. Cladding tube diameters ranged from 0.3 to 0.5-in. ID, with wall thicknesses varying between 0.025 and 0.030 in. Tubing has been extruded in lengths ranging from 1-1/2 to 14 ft, with 6, 8, 10, 12, 14, and 16 fins. Finned tubes are normally twisted to 90°/ft of length during extrusion or in a post-extrusion operation. Good tolerances have been obtained.

The quality control measures required to assure the integrity of clad tubing include a hydrostatic test, a helium leak test, and dye penetrant inspections. In addition, there are eddy-current methods which have demonstrated their ability to detect surface and subsurface defects of the order of 10% of the wall thickness. Further refinements in the eddy-current procedures are desired to better define adequate inspection standards and associated acceptance criteria.

11. Uranium Metal Fuel System

The primary interest in uranium metal fuel for the HWOOCR is directed toward long-term future designs using natural uranium. The fuel is attractive for use at the low burnups achievable with natural enrichment because of its potentially low fabrication cost when coextruded with zircaloy cladding in an annular geometry.

The brief evaluation of uranium metal fuel assemblies for the HWOOCR has been limited to an advanced concept using concentric annular ring fuel assemblies with either zirconium alloy or finned aluminum cladding. Primary emphasis has been given to the higher temperature system in which zirconium alloy cladding looks best.

a fuel material which can be used in properly designed fuel assemblies to burn-ups of the order of 10,000 Mwd/MTU at temperatures of approximately 950°F.

C. PRESSURE TUBE

1. Introduction

Pressure tube assemblies for the HWOCR must use a material in the core area which has a low neutron absorptivity, is compatible with the organic coolant, and has good strength properties at temperatures of up to 765°F. Two materials were considered: SAP (a dispersion-hardened alloy of aluminum and particles of alumina) and zirconium alloys. The pressure tube assembly must also have end-seals which are easily removed remotely and provide a leak-tight seal against pressurized high temperature organic coolant. In general, the development work was advanced to the extent of demonstrating feasibility. In most areas of the pressure tube development, the work remaining is that of further qualification to establish confidence and reliability levels, and that of establishing production processes from laboratory processes.

The major problems associated with a pressure tube assembly utilizing SAP are: (1) the uncertainties with respect to whether or not there may be an unacceptable radiation effect on the mechanical properties; (2) the low strain capability under long-term high-temperature load conditions; (3) the inability to use conventional welding techniques for joining; (4) the low wear resistance; and (5) the lack of a commercial supplier and demonstration of commercial production. The major problem associated with zirconium pressure tube assemblies is the hydriding of zirconium in a high temperature organic coolant environment.

The emphasis of the material and pressure tube development was directed toward the use of SAP as it was felt that its feasibility for this application would be easier to demonstrate than that of zirconium. Since most of the pressure tube assembly work performed has been related to SAP pressure tube development, the technology status of SAP is the dominant feature of this section.

The limited data which have been obtained relative to the radiation effects on SAP show that there is no irradiation work hardening. Theory predicts that, under the conditions of HWOCR operation, there should be no detrimental radiation effects on the mechanical properties of SAP.

The uranium metal fueled HWOOCR design utilizes fuel assemblies consisting of three concentric uranium metal rings, with zircaloy cladding. The uranium metal is assumed to be an advanced version of the relatively radiation-stable class of trace alloys designated "adjusted uranium." In the evaluation study, it is assumed that sufficient improvement in the radiation performance of this fuel can be achieved to permit a peak burnup of 8000 Mwd/MTU at a maximum fuel temperature of 950°F with less than 5% volumetric swelling. The design is based on a maximum cladding surface temperature (including coolant film formation allowance) of 820°F because of hydriding considerations.

To minimize local radial power peaking within the fuel assembly, the fuel enrichment is varied. The maximum enrichment (natural) is used for the two inner rings, and a lower enrichment is used for the outer ring. The low average feed enrichment and short fuel exposure result in a specific plutonium production rate (kg fissile Pu per billion kwhe) approximately double that of the reference uranium carbide fueled design.

Confirmation of adequate fuel stability for the temperature and burnup range of interest does not presently exist. An increased scope development program is required to establish design adequacy. The particular combination of low alloying additions to provide sufficient stability against fuel swelling at the burnup required must necessarily be selected after careful evaluation of the existing experimental data on these materials, as well as the data which would be derived from such a development program. Indications at present lead to the conclusion that control of swelling in uranium metal fuel at a temperature of about 950°F is a feasible objective for future reactors on the basis of a continued fuel development. In some irradiation experiments⁽³⁹⁻⁴³⁾ "adjusted" uranium alloys (containing trace quantities of molybdenum, silicon, aluminum, zirconium, and iron) have exhibited little dimensional change above that normally predicted for solid fission products accommodation at burnups ranging from 0.5 to over 2 at. % at temperatures below 900°F. Exposures of 7000 Mwd/MTU are currently being demonstrated. The existing information indicates that without heat treatment the threshold temperature for swelling in uranium metal is about 850°F. Recent work has shown that this temperature can be increased by heat treatment and trace additions to produce insoluble intermetallics. It is, therefore, believed that continued development of these "adjusted" alloys can produce

Stress rupture tests on SAP pressure tubes with and without machined defects have been conducted. The results of the tests performed indicate that an acceptably conservative allowable design stress can be determined by using Section III of the ASME Code and by using a Larson-Miller correlation⁽³⁸⁾ to predict stress rupture values for the lifetime desired.⁽³²⁾

The joining of SAP to itself and to other materials poses special problems. SAP can be joined to itself by flash welding. The efficiency of a flash-welded joint is in the 85 to 90% range, based on tensile tests only.

SAP pressure tubes can be joined to other materials through a transition joint to stainless steel. A metallurgically bonded joint (SAP to SS) can be made by hot pressing; a mechanical joint can be made by rolling. Both hot pressed and rolled joints have been developed to an extent which demonstrates capability of withstanding 30 years of operation at 765°F.

The only present supplier of SAP material is the Italian firm, Montecatini. The quality of their material was found to be quite consistent and acceptable for use in the manufacture of pressure tubes. They demonstrated their ability to make pierced high quality billets suitable for large lot production. Although there is no domestic supplier of SAP material at the present time, several U. S. firms have indicated their interest and willingness to develop a SAP production capability on the basis of a SAP billet manufacturing process developed by the Oak Ridge National Laboratory.

Montecatini has indicated that they have a developed process for fabricating SAP pressure tubes up to 16 ft in length on a commercial basis. They would need to develop a process for extruding pressure tubes longer than 16 ft. Some domestic suppliers have extruded pressure tubes of the length required for the HWOCR.

Pressure tube assemblies have been fabricated and tested in out-of-pile organic loops. A SAP pressure tube assembly for insertion into the WR-1 reactor was partially completed. The pressure tube assembly tests performed have demonstrated the need for keeping the SAP fuel cladding fins from contacting and/or rubbing against the SAP pressure tube.

Pressure tube end seals have not been developed sufficiently to assure that the desired organic leakage rate can be achieved or to predict the useful life of

a seal. The B. F. Goodrich radial seal appears to possess the potential for achieving the desired goals. The effects of potential coke formation around the latching mechanism and the radial seal have not been fully evaluated.

The Canadians have irradiated zirconium alloy pressure tubes in organic coolant. Their experience indicates zirconium-2.5% niobium alloy to be a strong contender for use in HWOCR pressure tubes. Another zirconium alloy which possesses significant potential is Ozhennite 0.5 (99.5% Zr, 0.2% Sn, 0.1% Fe, 0.1% Ni, and 0.1% Nb). The precipitation of zirconium hydride and the associated effects are dependent on the temperature of the material and the concentration of hydride. The zirconium hydriding rate can be controlled by maintaining a small quantity of water in the organic (300 ppm), and by increasing the wall thickness to reduce the hydride concentration.

2. SAP Material

a. Billet Fabrication

Montecatini Edison S. p. A., Milano, Italy, is the only existing supplier of SAP billet material. U. S. industry has shown its desire to develop a SAP production capability. Montecatini's process is not as yet commercially qualified. Montecatini purchases the SAP powder from Eckart Werke, a German firm, which produces it under a licensing agreement with Swiss Aluminium, Ltd. (formerly Aluminium Industrie A. G.)

The Oak Ridge National Laboratory (ORNL) has developed a laboratory process for manufacturing SAP material of an equivalent quality to that of the Montecatini material. This laboratory process is expected to be conveniently adaptable to a production process.

The present method used for fabricating pressure tubes is to extrude 7-in. round pierced billets. Alternatives are to use 7-in. round solid billets and back-extrude, to upset the 7-in. billets to 10 or 11 in. in diameter, or to extrude tubes directly from 10 or 11-in. round compacts.* Most U. S. suppliers have indicated they would prefer to work with 10 or 11-in. billets.

*A 7-in. round billet is made by extruding a 10 or 11-in. compact

b. Billet Testing

A procedure has been established for the inspection of billets to determine if they are acceptable for extrusion into tubes. They are examined for oxide segregation, oxygen content, impurities, and inclusions. The oxygen content is established by the fast neutron activation method. Hydrogen and other impurity contents are established by standard chemical analysis procedures. Oxide segregation is determined by macrostructure studies of etched samples. Each billet is 100% ultrasonically scanned for the detection of inclusions or voids. The procedures and the extent of testing used are considered adequate for appraising whether billets should be rejected or accepted for making tubes.

c. Billet Extrusion into Pressure Tubes

Montecatini has the capability of extruding pressure tubes with 4-in. ID, 0.090 to 0.150-in. -thick walls, and up to 16 ft in length. This capability is based on the production of pressure tubes for the ORGEL Program. They will require some development work before they are in a position to furnish and to guarantee specification compliance for pressure tubes of the sizes required for the HWO CR. Table 18 shows a comparison of some of the SAP tubing dimensional tolerances between Montecatini's present capability and the HWO CR design requirements.

TABLE 18
COMPARISON OF EXTRUDED SAP TUBING DIMENSIONAL TOLERANCE
BETWEEN MONTECATINI'S PRESENT CAPABILITY AND
HWO CR DESIGN REQUIREMENTS*

	Montecatini	HWO CR Design Goal
Wall Thickness (in.)	+ 0.020 - 0.000	+ 0.015 - 0.005
Internal Diameter (in.)	+ 0.020 - 0.000	+ 0.015 - 0.005
Sag (in. / ft)	0.006	0.015
Sag (in. / 16 ft)	0.079	0.100
Ovality (in.)	0.004	0.010
Length (ft)	16.4	25

*Based on information provided by Montecatini

The U. S. manufacturers have extruded pressure tubes of approximately 4-in. ID, 0.090 to 0.200-in. wall, and up to about 30 ft in length. Good extrusions have been obtained on a given order, whereas, on subsequent orders, unacceptable extrusions were made. The reason for this is believed to be primarily a lubricating problem. However, further improvement can be made by optimizing the die design, extrusion temperature, lubricant and lubrication procedures, ram speed, and billet size.

d. Pressure Tube Evaluation Testing

Following extrusion, pressure tubes are tested to evaluate their acceptability for further use. Both destructive and nondestructive testing is performed. At the present time, samples from each extrusion are subject to stress-rupture and tensile testing. In addition, each tube is submitted to a visual examination (including dye penetrant testing), dimensional inspection, ultrasonic testing, helium leak testing, and a go, no-go gauging for checking the inside diameter of the tube.

Nondestructive testing inspection of tubing has shown that ultrasonic testing is sufficiently sensitive to detect even minor flaws. However, only a very limited amount of work on correlation between indicated ultrasonic defects and metallographic evaluation of such defects has been done. Other than this, all remaining NDT techniques are standard, although some refinements in them will be necessary before an NDT procedure can be established to provide acceptable levels of quality assurance.

e. Mechanical Properties

Sufficient data are available on the mechanical properties of SAP to assure that the material can be used in nuclear power plants. The SAP Handbook, AI-CE-Memo 24,⁽³²⁾ presents a compilation of existing data on the properties of SAP. There are, however, some uncertainties related to the lack of information on (1) the effects of radiation on the long-term stress-rupture properties at elevated temperatures, (2) the effects of low strain rate at elevated temperatures, and (3) the effects of low cycle fatigue.

The effects of radiation on the short-term mechanical properties of SAP have been determined on specimens which have been irradiated to about 1.2×10^{21} nvt ($E > 1.0$ Mev) at temperatures of 104, 527 and 788°F.⁽³²⁾ No

significant changes have been observed between irradiated and unirradiated specimens, except that at 104°F there was an increase in the strength due to irradiation work hardening. Theory⁽⁴⁴⁾ indicates that there should be no irradiation work hardening effects if the material is used at a temperature above 370°F. For other than the work hardening effect, the available data are for exposure equivalent to about 1 to 1-1/2 years of pressure tube operation in the HWO CR.

The composition of SAP changes under irradiation as a result of the transmutation of aluminum to silicon and the formation of helium from oxygen. The maximum silicon and helium contents after a 30-year exposure are expected to be about 3% and 12 ppm, respectively. The buildup of silicon should not adversely affect the mechanical properties; in fact, it might improve the strength due to the possible precipitation of aluminum-silicon intermetallics. The effect of the formation of helium on the mechanical properties is not known; however, theory^(45, 46, 47) shows that about 100 times more helium could be tolerated without reducing the ductility below the values now observed for unirradiated SAP.

Allowable design stresses were based on the interpretation of Section III of the ASME Code for Creep and Stress-Rupture Limited Designs. These stresses should not exceed the lesser of:

- 1) Two-thirds of minimum yield strength
- 2) One-third of minimum tensile strength
- 3) The stress that would cause 1% creep in the design lifetime
- 4) Eighty percent of the minimum stress that would cause rupture in the design lifetime.

Because of the mechanical anisotropy of SAP, the transverse strength differs from the longitudinal strength, which makes it difficult to determine to what the calculated stress value should be compared in the multi-axial stress situation. The appropriate failure theory to adopt, i. e., maximum shear stress or effective stress, is not apparent; in fact, neither is truly applicable where anisotropy exists.

The Larson-Miller correlation^(32, 38) is used to predict long-term stress rupture values based on the available data. Stress rupture data for both a 2:1 and a 1:1 stress condition for both smooth and notched tubes, when plotted as a function of the Larson-Miller parameter, fall above the minimum predicted

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Figure 19. SAP Pressure Tube Stress Rupture Test Data
(ID = 4.0 in. Wall Thickness = 0.092 to 0.101 in.)

transverse stress line based on literature data (Figure 19). The results from the notched specimens tested to date indicate that an apparent stress intensity factor of 10 to 12% exists when based on stress in the thickness remaining under the notch.

3. Zirconium Alloys

The absorption of hydrogen in zirconium based alloys is the major consideration associated with using these materials for HWOGR pressure tubes. When zirconium hydride is in solution, it does not produce an unacceptable effect on the mechanical properties of zirconium alloys; it is only upon precipitation of zirconium hydride platelets that a detrimental effect (embrittlement) occurs. The rate at which hydrogen is absorbed in zirconium alloys depends on (1) the condition of the tube surface (protective coating), (2) the operating temperature, (3) the alloy composition, (4) the hydrogen and water concentrations in the coolant, and possibly (5) the irradiation flux.

a. Effect of Hydrogen on Mechanical Properties

Brittleness has been shown to develop if deformation is attempted in a direction perpendicular to a preferentially oriented hydride distribution. Hydrides form in preferential directions related to prior strain or a state of stress during cooling from elevated temperatures. Randomly oriented hydrides do not have a marked effect on the short time tensile properties of zirconium alloys. The nature of the effect of high hydrogen contents on the high strain rate elevated temperature properties requires some additional work.

No definitive information exists on which to base the selection of a hydrogen concentration limit for operating zirconium alloy pressure tubes. However, an approach taken has been to limit hydrogen absorption to its solubility limit at the operating temperature. This approach is conservative; higher levels should be established.

b. Rate of Hydrogen Pick-up

In the presence of water in the coolant, hydrogen is formed by the oxidation reaction of zirconium with water. The resultant surface film of ZrO_2 serves as an inhibitor to the absorption of hydrogen. Organic coolants containing over 50 ppm of water provide sufficient water for the zirconium water reaction to

maintain a continuous film of ZrO_2 on the surface of the pressure tube. The free hydrogen in the organic coolant appears to have a negligible effect on hydriding rates if the water concentration in the coolant is maintained above 50 ppm.⁽¹⁹⁾

Experimental data⁽⁴⁸⁻⁵²⁾ indicate that zirconium-2.5% niobium and Ozhennite 0.5 have hydriding rates of about 65 and 20%, respectively, of that of low-nickel Zircaloy-2. Table 19 shows hydriding rates for four zirconium alloys in HB-40 at 750°F. Both Zr-2.5% Nb and Ozhennite 0.5 show excellent potential for use in HWOCR-type plants.

TABLE 19
HYDRIDING RATES OF FOUR ZIRCONIUM ALLOYS
IN HB-40 AT 750°F⁽¹⁹⁾
(Water content 80 - 150 ppm, test duration 8400 - 11,800 hr)

Alloy	Hydriding Rate ($\mu\text{g}/\text{cm}^2\text{-hr}$)
Low-Ni Zircaloy-2	14.5×10^{-3}
Zr-2.5 Nb	9.3×10^{-3}
Zr-0.7 Cu-0.2 Fe	9.3×10^{-3}
Ozhennite 0.5	3.1×10^{-3}

c. Effect of Irradiation on Hydriding Rate

Limited data exist on the effect of irradiation on the hydriding rate of zirconium alloys in an organic environment. Some of the data indicate that the irradiation effect decreases with increasing temperature and disappears at approximately 750°F. In the U-305 experimental fuel string, Zircaloy-4-clad UO_2 fuel rods were irradiated at a peak temperature of approximately 860°F in organic for about nine months. Post-irradiation examination showed that the zirconium cladding had failed (no failures were observed in the SAP). Preliminary evaluation of the Zr-clad rods indicates that the cladding had been hydrided, but the observed failure may well have been attributable to defective end closures. Additional Zircaloy-4-clad UO_2 fuel rods are presently being irradiated in the U-305 Alternate 1 experiment at a peak temperature of about 860°F in organic; these rods have now been under irradiation for over one year.

4. Transition Joints

A transition joint is required between the in-core section of the pressure tube and the carbon steel external piping system. Primary emphasis has been placed on developing a transition joint between SAP and stainless steel. Since SAP cannot be joined by conventional techniques, alternatives were pursued. Two alternatives showed sufficient promise for development. The two methods under development were metallurgically bonded joints (hot-pressed joining) and mechanical joints (rolled joining). The process for joining zirconium alloys to steels by rolling was developed for the NPR and the NPD reactors.

a. Hot Pressed Joints (Figure 20)

(1) Fabrication

Metallurgically bonded transition joints have been developed to the extent of demonstrating the feasibility of hot pressing joints between SAP and stainless steel. A diffusion barrier is required between these materials; a nickel-tungsten barrier is used.

The wall thickness of the SAP tubes must be increased at the ends where the hot pressed joint is made, because the apparent efficiency of this joint, based on tensile tests, ranges from 75 to 85%. Generally, a well-bonded joint will fail at a stress level somewhere between the yield strength and the ultimate tensile strength of the SAP material.

The ends of the pressure tubes can be thickened in several ways. The flash welding process was pursued more than any other. This process provides for welding machined thicker sections to the ends of the pressure tube. It has the detrimental effect of creating a weak place at the weld in a thin-wall section of the SAP tube. However, there are indications that an acceptable flash welded joint could be developed by taking advantage of the anisotropy of SAP and of the rearrangement of the anisotropic characteristics in the region of the flash weld.

The chemical milling and the standard machining processes have been demonstrated as feasible. The potential disadvantage of these processes is that of cost because of the amount of material lost in the processing. Dimensional control has been shown to be excellent for the chemical milling; for the standard machining process, it may present a problem. Of the other methods evaluated for pressure tube end thickening, only the explosive bonding process shows promise.

Figure 20. Hot Pressed Transition Joint

The process parameters required to fabricate satisfactory hot-pressed SAP-stainless steel joints have been established. These include the nickel plating and tungsten vapor coating of the stainless steel components and the time, temperature, and hot pressing pressure required to make the joint. The process has yet to be scaled for a production procedure and the production process qualified.

Nondestructive testing techniques have been demonstrated for use in qualifying the nickel and tungsten coatings prior to hot pressing the joint. The Hall effect method has been demonstrated for measuring the thickness of the nickel layer. This method has excellent sensitivity; in the range of interest, it can detect a 0.0001-in. change in nickel thickness. The beta back-scatter technique is used in measuring the tungsten thickness. Its accuracy has been demonstrated to be within $\pm 10\%$ of the thickness determined metallographically. The helium leak and hydrostatic testing techniques are used to establish joint integrity after hot pressing; however, two immersion ultrasonic techniques show considerable promise as additional methods for appraising integrity. These two techniques are delta scan and transmission ultrasonics.

A contact ultrasonic technique was established for testing SAP-to-SAP flash weld joints. A plastic acoustic lens is used to refract a shear wave at 89° normal to the weld plane. Standards have been established and the procedure qualified.

(2) Performance

Tensile test data indicate that the hot pressed joints will remain stronger* than the SAP pressure tube (on the basis of ultimate stress), both in the as-bonded condition and after 1000 hr of isothermal heat treatment at 950°F . Isothermal heat treatment was used for establishing the activation energy of the W-Al diffusion barrier system. The activation energy is in excess of 80,000 cal/mol which means that the 1000-hr test at 950°F is equivalent to at least about 250 years at 760°F (the operating temperature of an HWOCR transition joint).

*Based on a thickness of SAP at the joint approximately 1-1/2 times the required thickness for the pressure tube.

Thermal cycling 10,000 times between 650 and 750°F at a rate of about 30°F/sec has not shown any deleterious effect on the strength and leak tightness integrity of the joint. Although considerable testing is needed with respect to combined loading, the transition joint is expected to exhibit a higher joint efficiency when exposed to internal pressure as compared with simple tension. Evidence of this was observed by failure occurring in the SAP tube of double-ended transition joint test sections.

The joint has not been subjected to stress-rupture testing. The principal advantage associated with stress-rupture testing is that it would provide data for use in optimizing the transition joint wall thickness.

b. Rolled Joints (Figure 21)

(1) Fabrication

Rolled transition joints of SAP to stainless steel have been made for pressure tube applications. The rolled joint is of the sandwich type, where the SAP is rolled between inner and outer stainless steel sleeves. The fabrication techniques have been developed to assure structurally sound joints. The greatest uncertainty is whether or not the joint will remain leak-tight for 30 years of reactor operation. Leaks could be promoted primarily by either stress relaxation of the joint or by the formation of reaction products between the SAP and the stainless steel. There are several materials which could be used as diffusion barriers, should the formation of reaction products prove detrimental to the joint integrity; however, some additional development work would be necessary if something other than an oxide coating were to be used. At the present time, an oxide-nitride coating on the stainless steel appears to be an adequate diffusion barrier. The oxide-nitride coating is formed by using a standard commercially available process, consisting of treating in a salt bath.

Each end of the SAP pressure tube has to be flared in order to maintain a constant inside diameter in the pressure tube assembly. The SAP tubing can be flared by mechanical mandrels, by hydraulic expanders, and by explosive forming. All three methods are satisfactory; however, a compressive axial load must be applied to the tube during flaring to prevent splitting.

The stainless steel sleeves can be either grooved or smooth. If they are grooved, two grooves are more than adequate. The grooves are sinusoidal and

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Figure 21. Rolled Transition Joint

need not be more than about 8 mils in depth. Sharp corners have the tendency to damage diffusion barriers as well as break the SAP pressure tube wall as a result of high shear stress in the low-ductility material.

The actual rolling process for making this sandwich-type joint is a relatively standard one. After inserting the flared end of the SAP tube into the slot formed by the concentric stainless steel sleeves, an initial set in the joint is made by driving a mandrel axially through the inside diameter of the inside sleeve. Standard rolling procedures are followed after this initial set is achieved.

Helium leak testing is presently used as the means for evaluating the integrity of rolled joints. Joint X-ray techniques and some ultrasonic techniques were considered. However, these alternative NDT techniques do not hold much promise of being acceptable.

(2) Performance

Tensile test data show that the rolled joint is stronger than the SAP pressure tube, both in the as-bonded condition and after an equivalent stress relaxation life of better than 30 years. Thermal cycling 1000 times between 400 and 800°F, at a maximum rate of about 30°F/sec, has not shown any deleterious effect on either the strength or the leak tightness integrity of the joint.

Rolled joints subjected to a series of tests have retained their integrity as measured by helium leak testing. In this series, joints were thermally soaked at 1000°F for 500 hours. The activation energy for aluminum stainless steel diffusion is about 45,000 cal/mol, and for stress relaxation of stainless steel about 95,000 cal/mol. This soaking period was therefore equivalent to approximately a 12-year lifetime with respect to diffusion and a stress relaxation lifetime in excess of 1000 years. Following the thermal soak tests, the joints sustained pressure and temperature cycling, after which they were stress-rupture tested under a one-to-one hoop-to-axial stress ratio. The results of this series of tests indicate that the joints tested would have been adequate for use on a pressure tube. There still exists, however, the need to fully qualify the fabrication process and scale it up for both shop and field assembly usage.

5. End Fittings

The B. F. Goodrich single-flex radial seal has been tested sufficiently to establish feasibility and future development needs. This seal has been subjected to both steady state and cyclic elevated temperature conditions. Seal leakage is dependent on the thermal transient rate of the organic. Transient temperature rates in the end seal region greater than $5^{\circ}\text{F}/\text{sec}$, from 400 to 700°F , tend to cause leakage rates in excess of 5 cc/hr . Seal leakage rates of less than $1/2\text{ cc/hr}$ can be easily attained when the static temperature in the region of the seal is maintained at 400°F or less.

A test was conducted on the effect of coke formation in the latching mechanism. A CANDU end-seal mechanism was soaked in Santowax OM and then baked. The results of this test indicate that the breakaway force is greater if the organic in the area of the mechanism is removed by air boil-off rather than by draining. When the mechanism was cooled to 200°F , it could still be actuated by a force compatible with the HWOCR refueling machine requirements. Although the CANDU mechanism is similar to the one envisioned for the HWOCR, the effects of coke on the HWOCR mechanism should be evaluated.

The possible formation of coke in the area of the single-flex radial seal may prevent this seal from fully relaxing. If this occurs, difficulties could be encountered in removing the seal. It may be necessary to use a modified single-flex seal or a double-flex seal which will override this possible coking effect.

6. SAP Pressure Tube Assembly

Three in-core sections or mid-sections of the SAP pressure tube assembly have been tested in an out-of-pile organic hydraulic loop. These sections are about 20 ft long with transition joints on each end. Both rolled and hot pressed transition joints have been tested.

The feasibility of fabricating SAP pressure tube assemblies using both rolled and hot pressed joints has been demonstrated. From a fabrication viewpoint, it has been shown that the dimensional tolerances and inspection requirements are realistic and can be met with careful fabrication procedures.

The program for each of the three SAP in-core test sections tested was planned for three 500-hr out-of-pile tests. Two of these performed as designed,

and one failed. The failure occurred at one of the flash welds which joined the hot pressed joint to the SAP pressure tube. The failure occurred following completion of one 500-hr test at 650°F and 14 hr after the startup of the second 500-hr test at 550°F. The calculated axial stress, including pressure stresses, ΔP forces, weight loads, and measured bending deflections, was between 4600 and 7100 psi at the flash weld which failed. This stress level should not have caused failure of the parent material. The failure can be accounted for by adding the effects of a flash weld efficiency of 85% and a stress concentration factor at the weld of 2.5 to 3.0. Metallurgical examination indicates that failure could have been caused as a result of either poor quality SAP material* or high temperature stress-rupture effects on the flash weld under combined loading.

The major uncertainty which became apparent in these tests and has not yet been resolved is that of vibration-induced wear between the fuel bundles and the pressure tube. Almost all the fuel bundle designs tested to date had SAP fins contacting the pressure tube wall. Only one test with one fuel bundle has been conducted where the SAP fins did not contact the pressure tube wall. In all the tests performed, an unacceptable amount of wear has resulted between these components.

Data exist which show the wear rate of SAP on SAP to be about 500 times that of steel on steel.⁽⁵³⁾ The wear between fuel bundle and pressure tube has been observed for full-length (44-in.) fuel bundles, half-length fuel bundles, banded and unbanded fuel bundles, fuel rods with 6 and 12 fins, fin widths of 30 and 50 mils, and fuel bundles banded to give the equivalent of one-third length bundles. After 500-hr test runs, the wear varied from rub marks to grooves up to about 9 mils deep. Wear marks were observed at each fuel bundle location.

The following conclusions can be reached on the basis of the tests performed:

- 1) No correlation between vibration and organic temperature and flow has been established.
- 2) The frequency and amplitude of vibration of the fuel bundles appear to be random and not a function of the vertical position along the length of the fuel string.
- 3) Stainless steel bands on fuel bundles which are mounted flush with the cladding fin tips do not prevent wear at the point of fin contact with the pressure tube.

*The SAP material used for this flash weld was some of the original material manufactured and procured prior to the start of the HWOCR program.

4) Stainless steel bands on fuel bundles, which extend beyond the cladding fin tips by about 15 mils, prevent wear between the fins and the pressure tube. However, in the one test performed, wear was observed between the bands and the pressure tube.

5) Compressive loading of the fuel bundles does not prevent wear.

Insufficient testing has been performed to establish a conclusive approach for either eliminating or reducing wear to a degree acceptable for the lifetime desired. Other than wear, no additional detrimental effects (such as transition joint failures) to the pressure tube assembly have been observed from the normal vibration loadings.

Considerable data (both in-pile and out-of-pile) exist for zirconium alloy pressure tubes. Zirconium alloy pressure tubes have been operated in organic in-pile loops. After approximately two years of operation of a Zircaloy-4 pressure tube in the U-3 loop of NRU, at conditions representative of the HWO CR and containing both SAP-clad and Zircaloy-clad fuel bundles, the only observed wear on the pressure tube corresponded to the location of fuel bundle end-spacer rings. The extent of this wear is not fully known at this time. After the wear had been observed, it was appraised as not being serious enough to prevent continued operation. This in-pile test is continuing at the present time.

D. ON-POWER REFUELING MACHINE

A dual-machine concept (two refueling machines, one at each end of the pressure tubes and operating in unison) was selected as being best suited for HWO CR usage. The two-machine concept was designed to the conceptual stage. This conceptual design has been used in identifying areas in which development and test were needed to demonstrate applicability and reliability. Detailed engineering on several components of the refueling machine was performed to enable design comparisons to be made and to prepare specifications for procurement of items for testing. A description of the refueling machine is presented in Reference 54. There are five major components or subassemblies which require development. These are: (1) tool post, (2) magazine, (3) snout assembly, (4) environmental control, and (5) instrumentation and control.

1. Tool Post

Each of the three rams of the compound mechanical ram system is driven by hydraulic motors. The main ram has the two other rams located concentrically

within it. The motor for the main ram drives two large ball-screw assemblies approximately 3 in. in diameter and 30 ft long. The ball-screw assemblies will be operated in 400°F organic. The two main uncertainties with the ball-screw involve materials compatibility and manufacturing feasibility.

Manufacturers of ball-screws believe that a combination of Star "J" Stellite for the balls and wrought alloy steel, A1514150, for the screw will be satisfactory. The compatibility of these materials for HWOGR conditions remains to be demonstrated. The ball-screw is of a size larger than anything manufactured to date. However, manufacturers believe that the size is feasible and well within present fabrication capability.

In the present design, the amount of compressive load applied to the fuel elements is controlled by a torque control device located near the drive and gear train. A problem arises when fuel movement is reversed and in the upward direction. The problem is that of developing a maximum compression load on the fuel bundles in excess of 7000 lb. A loading of this magnitude produces difficulties in fuel bundle design. Insufficient design and development have been carried out to date to allow for this peak compressive load to be reduced to a more tolerable level.

2. Magazine

The design of the magazine and fuel bundle gating mechanism have been developed to the extent that indicates feasibility. No component development work has been conducted. Areas which would require development are material compatibility, gating reliability, and performance reliability.

3. Snout Assembly

The snout assembly utilizes a modified Graylock seal, the performance of which should be established. After refueling a pressure tube and after the pressure tube end seal is in place, the adequacy or leak tightness of the end seal has to be determined before the refueling machine can be disconnected. The best method for measuring the leak tightness of end seals has yet to be identified.

4. Environmental Control

Limited analysis exists and no component development work has been performed on the refueling machine environmental control system. The refueling

machine cooling system has to be developed sufficiently to demonstrate capability of removing decay heat from stored fuel bundles, of maintaining a maximum operating temperature of 400°F, and of adequately cooling fuel bundles as they are removed from the pressure tube assembly. No problems are anticipated in the performance of this work

5. Instrumentation and Control

Only conceptual designs exist on the general requirements for controlling two simultaneously and remotely operated opposing refueling machines operating in a common pressure tube. The relative merits of a special or general purpose computer for storage of logic functions must be determined. The ability of the control system to prevent damage to the pressure tubes and fuel bundles must be demonstrated. The ability of the control system to align the refueling machine and to operate reliably to prevent damage of components and leakage of coolant remains to be demonstrated.

E. COMPONENTS

Large-size equipment items used in the HWOCR facility include the reactor, steam turbine, main coolant pumps, main coolant loop isolation valves, and steam generating equipment. The design and construction of the reactor are discussed in Section II-A. Steam turbines of the size needed are being manufactured for the water cooled nuclear reactors. The HWOCR steam conditions (superheated and reheated) will result in fewer problems than will be encountered with the saturated steam cycles of the water reactors. The turbine, therefore, is not considered a particular problem of the HWOCR.

1. Main Coolant Pumps

The coolant pumps were sized on the basis of one pump per loop. The pump rating is 90,000 gpm of 750°F coolant with a 20,000-hp steam turbine drive. While there are no pumps now in operation at these service conditions, pumps for pressurized water reactors of the capacity required are being planned or are under construction. A number of manufacturers surveyed under this program have expressed confidence that pumps for HWOCR service conditions can be built utilizing existing materials and technology. A sensitive area of the pump design is the shaft seal. The seals in the HWOCR pumps will be about 10 to 12 in.

in diameter and will rotate at 1200 rpm. Larger seals have been built (for shafts up to 16 in.) for low-temperature high-pressure water service. Extrapolation of present seal experience to HWOOCR conditions will require demonstration of reliable performance.

In summary, the pump materials, size, and hydraulic performance are within present-day technology and proof-testing should result in reliable operation.

2. Main Isolation Valves

The HWOOCR system requires 36-in. gate valves which can be closed in 10 to 20 seconds. While these sizes are large, valves of this size have been built for service conditions similar to those of the HWOOCR (high pressure steam). The service temperature (750°F) and pressure (~400 psig) are well within the capabilities of present materials and construction techniques. No unusual problems are expected in this area. The most suitable valve operator is a pneumatic cylinder, since it can be made reliable and be equipped with an integral pressure source as a backup supply. Valves up to 20 in. in size have been used in organic service for several years without problems.

3. Steam Generators

There are three units in the steam generating system, namely: (1) the evaporator, (2) the superheater, and (3) the reheater. The maximum service conditions for the organic are 750°F and 400 psig, and for the steam, 725°F and 950 psig. Since the organic is non-corrosive, low-cost carbon steel can be used for all of the items. The design of the units is based on the central manifold header concept which is being used at the NPR. This concept is particularly suitable for large units since it minimizes tube sheet thickness requirements and thermal stress problems. The large surface area in the evaporator permits excellent disengagement space, and the carry-over of water to the superheater has been demonstrated to be small. The size of the units is within present manufacturing capability, and no unusual problems are foreseen.

F. CONTROL AND SAFETY

1. Introduction

The features of the HWOOCR design which are most important to the control and safety of the system are:

1) The coolant temperature and void coefficients and the power coefficient of reactivity which affect the dynamic behavior of the reactor.

2) The arrangement of the primary coolant system, namely, individual pressure tubes containing fuel and coolant, connected to the headers, mixing tanks, and main coolant loops, which affect the heat transfer characteristics of the core under loss of flow and/or coolant system rupture conditions.

For an HWOCR design based on the objectives of competitive economics and maximum fuel utilization, the coolant and void coefficients of reactivity will be positive. Based on studies of plant operation and potential transients, normal reactor control can be provided with conventional reactivity control systems, and accident conditions can be accommodated by the highly reliable shutdown system designed for the HWOCR.

The arrangement of the primary coolant loops, headers, and individual lines to each pressure tube provides the capability for isolation of ruptures and introduction of emergency coolant at a number of locations to assure transient and steady-state core cooling following line ruptures and pump outages. HWOCR safety studies have shown that the appropriate systems will assure core cooling (i. e., no fuel melting) following any credible component failure.

Thus, an overall appraisal of the safety of the reference design HWOCR, based on analog and digital computer studies, is that the design will meet present-day siting and safety criteria. In one area, namely the effect of operation with positive reactivity coefficients, in which studies show that conventional control systems are adequate, reliability of the power control system, shutdown system, and backup shutdown system is required to prevent the possibility of a divergent power transient. If the independence, redundancy, and reliability designed into these systems were not acceptable to license an HWOCR plant, detailed studies have shown the feasibility of designing less economical plants with negative coolant, void, and power coefficients. This is accomplished by reducing the lattice spacing and/or inserting displacers in the moderator to effectively dry up the lattice. Although it is believed that the design optimized on economic performance provides adequate safety characteristics, a design with negative coefficients is feasible if this were to be established as a design requirement.

2. Reactor Control

a. Reactivity Coefficients

The fuel coefficient of the reference HWOCR is negative, primarily because of Doppler broadening of the U^{238} resonance. The coolant coefficient is positive, resulting from competing effects of changes in spectrum, absorption, moderation, and leakage. The coolant time constant is about 1 to 2 seconds. The power coefficient is a combination which results from HWOCR performance factors, namely, $\Delta T_c / \Delta P$ and $\Delta T_f / \Delta P$ times their respective coefficients.* For a high-performance reactor design operating under equilibrium fuel cycle conditions, these factors cause the positive coolant temperature coefficient to outweigh the negative fuel temperature coefficient resulting in a positive power coefficient.

One of the components of the power coefficient which is unique to the HWOCR in its magnitude is the spectral effect. With the relatively cold D_2O moderator forming the neutron spectrum, the moderating properties of the organic and of the carbon bound in the UC fuel are effective in changing the neutron temperature seen by the fuel. The influence of the spectrum on reactivity coefficients changes significantly from the initial core operation to the equilibrium fuel cycle, when substantial amounts of plutonium are present. The reactivity effect of an increase of the mean thermal neutron temperature is negative with uranium as the fission source and positive with plutonium as a fission source. Therefore, during initial operation of the HWOCR, the power coefficient of reactivity is negative until substantial amounts of plutonium have been built into the core.

The void coefficient of the coolant is positive and of such a magnitude that voiding of all the channels is worth approximately 3.5% $\Delta \rho$. Because of the physical arrangement of the coolant system with each quarter of the core fed by separate headers, it is difficult to conceive of more than one quarter of the core loosing all its coolant at any one time, and the time required for this may be long compared with the response time of the shutdown system.

* T_c = temperature of the coolant; T_f = temperature of the fuel; and P = fraction of full power

The response of the moderator is slow with respect to the kinetic behavior of the core so that its coefficients do not influence the reactor dynamics appreciably. The temperature coefficient of the moderator is positive and its density (or void) coefficient is negative. The temperature effect is similar to that of the coolant coefficient, namely a positive spectrum effect. Since the density coefficient is negative, the lattice is, by definition, under-moderated.

The reactivity coefficients are dependent on fuel exposure: the negative fuel temperature coefficient decreases and the positive coolant temperature coefficient increases with burnup. The power coefficient will therefore be positive in the equilibrium cycle of a high performance design. This, coupled with on-line refueling, allows minimization of the worth of the reactivity control requirement since, when the power starts to decrease, it will continue to decrease due to the negative reactivity feedback. Thus, the fast shutdown control system need only accommodate potential reactivity addition equivalent to the voiding of all the coolant channels, or about 3.5% in reactivity, with high reliability and adequate margin.

b. Transient Analysis

As previously stated, the HWOCR can be designed to have negative reactivity coefficients with a penalty in economics. With a design optimized for performance, however, in which fuel, coolant, power, and void coefficients are $-0.58 \times 10^{-5} \Delta\rho/^{\circ}\text{F}$, $+5.8 \times 10^{-5} \Delta\rho/^{\circ}\text{F}$, $+2.3 \times 10^{-5} \Delta\rho/\%P$, and $+3.5 \times 10^{-4} \Delta\rho/\%$ void respectively, a detailed transient analysis shows the following:

- 1) The reactor response to changes of reactivity due to typical operational perturbations (step changes due to reactor noise, $1.0 \times 10^{-5} \Delta\rho$, or inlet coolant temperature changes of $6^{\circ}\text{F}/\text{min}$ maximum) allows more than 10 seconds to elapse before the power rises a few percent even for the most positive power coefficient. Thus, for normal steady-state operation, a simple automatic controller can be used to stabilize reactor power; manual control is also feasible.

- 2) A simple automatic control system having a maximum reactivity control rate of $\pm 0.05 \times 10^{-3} \Delta\rho/\text{sec}$ can compensate for inlet coolant temperature changes as high as $50^{\circ}\text{F}/\text{min}$; this is about 8 times the maximum rate of inlet temperature change expected from the external plant even under accident conditions.

- 3) There is an incentive to make the power coefficient slightly negative in order to obtain stable steady-state operation without an external controller. However, even in the case of a negative power coefficient, an

automatic controller would be required for load following. The positive power coefficient controller must necessarily be designed for very high reliability because it will be constantly cycling to maintain power constant.

4) The positive power coefficient reactor response at full power to abnormal operating conditions such as relatively large ramp reactivity changes due to control rod motion, inlet coolant temperature changes, or inlet coolant flow changes, involves a minimum of two seconds before the power level exceeds the 110% power level trip point by 30%.

5) There is relatively little difference in the inherent response of the reactor, with a slight positive or slight negative power coefficient, to large perturbations in reactivity, coolant flow, and coolant temperature because of the overriding positive reactivity feedback effect of the coolant.

6) The largest operational transients anticipated for the reactor are those associated with a loss of coolant flow. Analysis of the most severe of these transients, the total loss of coolant flow, has shown that the proposed shutdown system can safely terminate the transient even when the trip signal is obtained indirectly from an overpower trip condition instead of directly from the loss of pumping power or loss of flow.

It is interesting to note that, in the loss of coolant flow accident terminated by the overpower trip, the transient associated with a reactor having a slight negative power coefficient results in a 20°F higher fuel cladding temperature rise than the transient associated with a slightly positive power coefficient. This is attributable to the larger integrated energy output of the fuel because of the longer time required for the reactor power to reach the overpower trip point in the case of the negative power coefficient.

7) A preliminary analysis of the uncontrolled rod withdrawal startup accident shows that the reactor shutdown system can terminate the resultant excursion before the peak temperatures of the fuel, cladding, or coolant reach the 100% power design points. In this analysis, the power transient, up to ten times full power, shows no sign of becoming inherently limited when not terminated by the shutdown system.

c. Space-Time Kinetic Behavior

Studies have been conducted on the space-time kinetic behavior of the HWOCR under local perturbations and the stability problems associated with a large core. The general conclusions can be summarized as follows:

1) No local perturbations affecting the core, with negative or positive coefficients, result in a local power runaway. Local reactivity additions (up to seven dollars) were used in the dynamic analysis using the WIGL-2 code modified to accommodate HWOCR conditions.

2) Xenon spatial oscillations will occur in this core, but, because of in-core monitors and a xenon control system, do not represent a safety problem.

3. Reactor Safety

The initial studies associated with the safety of the operation of the HWOCR were directed along the following lines:

- 1) Reactor dynamics and control, as described above
- 2) System dynamics and control
- 3) Component reliability, as it affects safety
- 4) Studies of potential failures and the accidents which could subsequently arise
- 5) Studies of the radiological consequences of accidents and their influence on plant siting.

The system dynamics and control were studied by the use of an analog simulation of the reactor-coolant-steam system. These studies included investigation of transients which could arise from normal system demands and failures which could be expected during the life of the plant. The summary conclusion of these studies was that the controls to be used on the reactor and primary system and the emergency on-line shutdown cooling system were adequate to assure that fuel damage would not occur.

Reliability studies were performed to assess the requirements of vital components of the safety systems. Particular attention was given to the shutdown control system where the definition of testing requirements was established by reliability-safety demands. Fault-tree studies were initiated to determine those areas of the system which would require additional reliability or redundancy to assure safe operation.

Operational accidents were studied by a systematic evaluation of potential failures and evaluation of the further consequences of malfunction of remedial engineered safeguards. On this basis, the maximum operating accident was defined as the rupture of a lower inlet header, followed by failure of emergency cooling. This would result in a quarter-core meltdown and subsequent release of energy and fission products to the containment building. From the definition of the circumstances and results of this accident, the containment building requirements and the post-accident engineered safeguards (e.g., spray systems, atmospheric coolers and filters) were established.

The radiological consequences of an inlet header rupture (maximum operating accident), a large coolant spill with subsequent burning, and a large spill of D_2O were evaluated to identify the controlling isotopes with respect to plant siting and establishment of the siting requirements. These studies resulted in the conclusion that the HWO CR would meet present-day siting and safety criteria and that its operation, therefore, would not result in any undue hazard to the general public or operating personnel.

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