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STUDY OF 40 MW PRESSURIZED WATER,
BOILING WATER, AND ORGANIC MODERATED REACTORS
FOR PRODUCTION OF PROCESS STEAM

by

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1. INTRODUCTION

This report covers the work by the Argonne National Laboratory in connection with a request by the AEC for a feasibility study of nuclear reactors for process steam. The following four points, taken from a letter M. B. Powers (COO) to Norman Hilberry (ANL), January 29, 1959, were used as the main guide points:

- 1) An evaluation of those reactor concepts which are fairly well proven such as the PWR, BWR, and OMR, to determine if any concept is particularly suitable for generation of 380°F saturated steam and which, if any, are unsuitable.
- 2) A study of materials of construction and fuel element types to determine if the low operating temperature could result in significant cost reductions. This should include recommendations of materials and fuel element types which are optimum for both the concept selected and the conditions of steam generation.
- 3) Shielding and containment should be considered as that required in an industrial site.
- 4) Recommendations of the R & D program which could be accomplished in the reactor specified (40 Mw thermal, 380°F outlet steam) with estimates of the upper operating limits for the reactor concept determined from 1) and 2) above.

2. SUMMARY

The study efforts were toward the creation of a reactor design for each of the three types and the development of each concept to a point where the range of costs could be established. In order to expedite the study, Sargent and Lundy performed most of the conventional system and the containment design. In addition, they made most of the cost estimates.

Table 2.1 lists the summary data for the three reactor types studied. The studies presented have high burnup cores of low enrichment in an effort to minimize fuel costs. Of the three designs presented in Table 2.1, the PWR core is the most favorable. The difference in core loadings between a PWR and BWR is not a characteristic difference. With redesign, a more favorable BWR core could be made although it probably would not be quite as favorable as the PWR. The OMR core design is as favorable as the PWR only because of a current situation which has an appreciable fabrication cost differential between the aluminum clad metallic fuels and zirconium clad oxide fuels. The reactor system designs are quite comparable.

Table 2.1

SUMMARY DATA

	<u>PWR</u>	<u>BWR</u>	<u>OMR</u>
Power, Mwt	40.6	40.6	40.6
Core diameter, in.	42	56.3	48
Core length, in.	42	60	48
Core volume, liters	953	2,442	1,475
Coolant	Water	Water	Diphenyl
Primary flow rate, lb/hr	4.22×10^6	1.39×10^5	7.9×10^6
Inlet temperature, °F	400	239*	421
Outlet temperature, °F	430	427	450
Pressure, psia	900	333	75
Enrichment, %	1.8	1.5	1.5
Initial loading of uranium, kg	3,380	10,300	7,650
Burnup, Mwd/M.T.	17,200	12,300	8,900
Process steam conditions			
Flow rate, lb/hr	1.25×10^5	1.25×10^5	1.25×10^5
Temperature, °F	380	380	380
Pressure, psia	196	196	196

*Inlet temperature into reactor vessel - not into core

Carbon and low alloy steels are acceptable materials of construction for the primary systems of both the PWR and BWR. While the corrosion rates will be higher than with stainless steel, adequate water quality control can be maintained by means of available techniques. The OMR has no materials of construction problems.

The oxide rod fuels for the water-cooled reactors are good from the standpoints of reliability and long life. However, their current fabrication cost is high. The aluminum cladding is acceptable for use in the OMR. The irradiation damage limits or the high percentage of alloying elements is a penalty for the OMR.

Aluminum corrosion rates or the means of controlling them is too uncertain to recommend the use of aluminum as fuel cladding material in water-cooled reactors where the aluminum temperatures reach as high as 500°F and a service life in the reactor is a matter of years.

The range of steam production costs for the designs given in this report are as follows:

PWR	64-150	¢/1000 lbs. steam
BWR	83-172	¢/1000 lbs. steam
OMR	75-154	¢/1000 lbs. steam

3. CONCLUSIONS AND RECOMMENDATIONS

3.1 Conclusions

1. The factors influencing the choice of reactor type and the characteristics of the design and of the core are too broad to allow a generalized choice of either type or design for process steam application. The quantity of process steam required, the pressure of the process steam, and the economic ground rules applied are so broad as to make generalized studies of small value. Fortunately in this study the quantity and pressure of the steam were set, and the effects of variations in ground rules are illustrated by the factor of two variations in the steam costs in this limited study.

2. Of the three reactor types studied for this report, none have dominant characteristics which show it to be a clear-cut choice over the other two for producing 125,000 lbs. of 380°F saturated steam per hour. All three types have sufficient operational experience to show that they have a potential for competitive production of process steam. The choice of any one type is more reasonably based on design and operating characteristics such as control, and fuel cycle.

3. The comparison between the cores of the two water-cooled reactors in this report is not a characteristic difference. The BWR could be made operable with a core having the same enrichment and total loading of uranium as the PWR. To do so would require the BWR to have higher velocity flow either by major increase in riser height for a natural circulation reactor or by adding pumps for forced circulation reactor. In either case, a steam separating drum or internal separators would be needed. In addition, the metal to water ratio might have to decrease. The former of these two factors increases capital cost; the latter factor, if required, decreases burnup with a resultant increase in fuel cycle cost.

4. Carbon or low alloy steels should be acceptable materials of construction for water-cooled reactors having operating temperatures of 500°F or lower providing the pH is maintained at 10-11, hydrogen is added to the water and dissolved and suspended solids are maintained at low levels. The carbon steel systems will corrode at a more rapid rate than the stainless steel ones. However the increased quantity of corrosion products can be controlled by a reasonably sized addition to the water quality control system. Also, the increased corrosion rate should not excessively affect either the heat transfer properties of the core, the operability of the reactor system, or the service life of the equipment. Under these alkaline conditions, aluminum cannot be used for any purpose.

If process steam of a few tens of pounds pressure were of interest, it may be that carbon or low alloy systems can have acceptable corrosion rates in high purity water of pH-7. Under these conditions, aluminum clad fuels may be acceptable for long periods in a reactor.

Aluminum clad fuels may be acceptable in stainless steel systems where the pH of the water can be 5-6. The aluminum surface temperatures as high as 500°F may be acceptable. The operating history of aluminum under these conditions shows erratic results. To recommend use of aluminum clad fuels for a few years service in a reactor would require a significant corrosion allowance of the cladding metal. The increased cladding needed for aluminum over zirconium displaces an appreciable amount of fuel from the core. The aluminum picture can well lose these uncertainties as some of the intangible factors in the present corrosion picture are understood and as water conditions to control corrosion rates are devised. Aluminum can be economically beneficial so long as a major price differential exists in the fabrication cost between aluminum and zirconium clad fuels. As this differential shrinks, and it should, the better corrosion characteristics of zirconium in high temperature water will be more important for long fuel service such as is proposed in this report.

The uranium oxide fuels are quite acceptable for the water-cooled reactors. Their major drawback is high price. This should be reduced with growing demand and with the new techniques such as oxides swaged in the tubes.

The OMR has no structural materials problems. Carbon steels are acceptable for the reactor primary system and aluminum for the fuel cladding metal. The OMR materials problems are in the coolant itself and in the fuels. With an expected reduction in the unit price of an organic coolant and a decrease in decomposition rate by some means such as inhibitors or by improved coolants, the influence of the cost of the organic coolant on the price of the steam can decrease from 10% to 5% of the total steam price, based on the most favorable total price.

The fuels for the organic reactor are not so well established. The metallic fuels, which have the best over-all performance, suffer from low radiation damage limits or, if the uranium has sufficient alloying constituents to tolerate high burnup, then the enrichment increases significantly.

5. The particular size reactor and the steam pressure chosen are quite acceptable as the first demonstration unit. The 40 Mwt reactor size is relatively small for an economically favorable nuclear system. However, it is not outside of the total competitive price range.

3.2 Recommendations

1. For any comparative reactor evaluations, the study rules should be thoroughly established prior to the start. The rules should include reactor size, operating characteristics such as steam conditions, condensate quantity and quality, plant factor and load conditions. In addition, all economic rules as to capital charges, operating costs, and method of handling inventory should be set. The optimum condition would be to set the reactor site and the reactor use prior to the evaluation.

2. For the conditions studied, a PWR system is somewhat favored over the BWR and the OMR. The ease of reactor and system load control due to the negative temperature coefficient and the better core characteristics give this advantage. While in this report, the PWR shows little economic advantage over the OMR, the PWR should have more potential. This potential comes from the larger fuel fabrication cost reduction probable with zirconium clad oxide than with aluminum clad metal. With these probable reductions, the combination of integrity and cost with a carbon steel primary system-zirconium clad fuel reactor is favored over a stainless steel primary system-aluminum clad fuel.

The carbon steel system is recommended because data indicate that the material is adequate. The reluctance to use it in reactors thus far can be attributed to the fact that stainless steel has been used successfully in reactors and to the attitude that it does not cost much more. A process steam reactor of the proposed operating conditions is an excellent place to prove out carbon or low alloy steels without seriously affecting the system operability.

3. Monel tubes are recommended for the boilers in all studies because of the unknown quality of the feedwater in the secondary system. Since chlorides are commonly found in water and since chloride stress corrosion is a problem with stainless steels, Monel is proposed as a substitute for a general study, where long tube life is considered important. When a site is selected and the secondary system water quality determined, the boiler tube material should be re-evaluated. It may be that the corrosion characteristics of the water will allow use of carbon or low alloy steel tubes.

4. The specifications for the final design of such a reactor system should be reviewed thoroughly to be certain that the high integrity type specifications for a reactor vessel are not generally applied to all reactor systems components.

5. While research and development programs directed toward the economic improvement of any of these reactor types is worthwhile, the following three areas can be particularly beneficial to the favored PWR concept:

- a) Increased efforts on the reduction of fuel fabricating costs with especial emphasis on the zirconium clad oxide rods. The current fabricating cost is the largest unit cost in the fuel cycle.
- b) Increase study of containment to reduce the required volume and pressure. This can result from more understanding of the need for containment volume and from study of methods of energy absorption rather than containment.
- c) Increased understanding of corrosion of materials and water quality control. In the case of water quality control, more effort should be applied to collect data and provide an understanding of the cooling systems of reactors now operating. The BWR system can profit equally from these same three areas; the OMR system requires more development in new fuel and improved economics of the coolant.

4. GENERAL

The most controlling factor in any such a reactor study is the cost of the steam produced. This study was held to well-proven reactor types. This was interpreted to also mean reasonable well-proven features of reactor design. In order to understand the economics, reactor designs were made. This work was carried out in the framework of holding costs down while using materials and adhering to designs having some proven background.

Since the economics are adversely affected by going in the direction of high enrichment and/or low burnup, the physics efforts were directed toward low enrichment, high burnup cores.

In order to prepare the preliminary designs, it was necessary to establish certain design criteria. These criteria are as follows:

1. The process steam flow requirement is 125,000 lb/hr at 196 psia and 380°F.
2. 93,750 lb/hr of condensate returns from the process system at 150°F.
3. Treated make-up water requiring deaeration is furnished to the system at 60°F.
4. Service water at 60°F and electrical power are available in adequate quantities at the plant site.
5. The plant is located next to an existing process plant on a site in the Middle West which has average soil conditions. Permissible load bearings for foundation design is considered as 4,500 lb/sq. ft. and the assumed site considerations are such that a minimum amount of dewatering is considered necessary in preparing foundations. Figure 4.1, while it is a plot plan of a water-cooled reactor, is typical in size for any of the three reactor systems.
6. The reactor, primary system, and auxiliaries handling primary fluid are housed in a steel containment vessel.
7. The control room, the service building, and the fuel handling building are constructed of insulated protected metal siding with a minimum amount of architectural treatment.

8. Access roads are adjacent to the plant site.
9. The reactors can have limited or no access during operation. Repairs and maintenance can be considered on a planned shut-down schedule.
10. The reactors can assume to run steadily at or near capacity. Major load swings need not be considered. A single boiler is acceptable.
11. Fuel management may consider both radial and axial reshuffling of fuel.

FIG. W.1.
PLOT LAYOUT

5. PRESSURIZED WATER SYSTEM

5.1 General Plant Description

The pressured water reactor plant proposed is housed in a steel containment vessel 56 ft diameter and 70 ft over-all height. This is illustrated in Figs. 5.1 - 5.3. The containment vessel has all the primary operating equipment. The process steam condensate deaeration, and feed-water treatment plus the demineralized water makeup for the primary circuit are located outside the containment vessel.

Adjacent to the containment vessel is a structure containing all other operating equipment. A control room is at a grade elevation and is located next to the containment vessel air lock doors for convenience in passing from the control room to the reactor building. Switch gear and other electrical apparatus are located beneath the control room. Water treating and fuel handling areas are located at the opposite end of the building from the control room.

Table 5.1 presents the summary data for the proposed system.

Table 5.1

PWR SUMMARY DATA

<u>Power, Mwt</u>	40.6
<u>Core Size</u>	
Diameter (nominal) in.	42
Length, in.	42
Volume, liters	953
<u>Coolant and Moderator</u>	Water
Total flow rate, lb/hr	4.22×10^6
Total flow area, ft ²	3.37
Flow area per subassembly, ft ²	0.065
Number of passes through core	2
Velocity	
First pass, up through center, ft/sec	13.8
Second pass, down through outside, ft/sec	10.3
Reflector, second pass control rods and thermal shield flow rate, lb/hr	6.36×10^5
Pressure at core inlet, psia	900
Pressure drop	
Total, psi	7.4
First pass, psi	4.8
Second pass, psi	2.6

Table 5.1 (Con't.)

Temperatures

First pass	
Inlet, °F	400
Outlet, °F	418.8
Second pass	
Inlet, °F	418.8
Outlet, °F	430
Total rise, °F	30
Average in core, °F	415
Maximum surface, both passes, °F	530
Maximum internal, °F	4250
Log mean temperature difference, °F	33.4

Heat Transfer

Total surface in core, ft ²	1525
Surface per subassembly, ft ²	29.3
Average heat flux, Btu/(hr)(ft ²)	9.08×10^4
Maximum heat flux, Btu/(hr)(ft ²)	4.74×10^5
Burnout heat flux, Btu/(hr)(ft ²)	2.2×10^6
Hot channel factors	
First pass	
Bulk temperature rise, $F_{\Delta t}$	2.25
Film temperature rise, F_{θ}	7.4
Internal temperature rise, $F_{\Delta T}$	5.2
Second pass	
Bulk temperature rise, $F_{\Delta t}$	3.1
Film temperature rise, F_{θ}	7.4
Internal temperature rise, $F_{\Delta T}$	5.2

Fuel

Rod diameter, in.	0.5
Cladding	Zr-2
Cladding thickness, in.	0.020
Radial air space (pellet to clad), in.	0.0015
UO ₂ pellet density, gm/cm ³	10.2
Pellet diameter, in.	0.457
Lattice spacing of rods, square pitch, in.	0.585
Number of rods per subassembly	64
Over-all dimensions of subassembly, in.	4.60 x 4.60 x 49.38
Number of subassemblies in core	52
Total number of rods in core	3328

Physics

Non-water to water ratio in fuel region	1.35:1
Over-all maximum to average flux	2.8

Table 5.1 (Con't.)

Physics (Con't.)

Cycle length, 100% operating, yrs.	3.8
Burnup, 90% of theoretical, Mwd/M.T.	17,200
Fuel enrichment, % U ²³⁵	1.8
Total loading of uranium, kg	3380
End of cycle	
Plutonium present, kg	28
U ²³⁵ present, kg	22
Void coefficient of reactivity, $\frac{\Delta k}{k} / \frac{\Delta \text{void}}{\text{void}}$	-0.2284
Temperature coefficient of reactivity, %/°F	-0.02164

Control Rods

Number in core	12
Shape	Cruciform
Thickness, in.	0.250
Length, in.	42
Material	2% boron stainless steel
Drive mechanism	Magnetic jack
Rod control, cold clean, % k	19.2
Cold clean core, % k	19.2
Hot clean core, % k	15.8

The more important components of the PWR system and the system operation and characteristics are discussed in sections that follow. Figures 5.4 and 5.5 are flow sheets and illustrated flow conditions. Figures 5.6 - 5.9 illustrate the reactor vessel and core.

The PWR has a well established background. The proposed reactor plant can be built without a prior research and development program. Since a carbon steel system is proposed, certain operating aspects represent a research and development program. An example of this is the water quality control system for the reactor. Were it desirable to have a system quite free of research and development phases, the primary system would be stainless steel except for the boiler tubes. These might also be stainless, if chloride stress corrosion were no problem. However, while a carbon steel reactor system has not operated, many tests show that the corrosion rate can be acceptable by using standard methods of water quality control. Thus, the operating data from such a reactor can be sufficiently valuable to warrant considering this operating phase a research program.

5.2 Design

5.2.1 Reactor and Primary System

Fuel (Figs. 5.8, 5.9)

The basic fuel element is a Zircalloy tube 0.5 in. O.D. by 0.02 in. wall by 43.875 in. long. There are 84 UO_2 pellets which make up the 42 in. active length. The pellet diameter is 0.457 in. A 0.5 in. fission gas and thermal expansion gap is placed at one end. Each end of an element contains a Zircalloy closure plug which is swaged and brazed in place. The 0.02 in. cladding thickness includes a 0.003 in. corrosion allowance.

A fuel subassembly consists of 64 fuel rods contained at each end in a Zircalloy tube sheet; the core loading is 52 of these subassemblies. The fuel rod and plugs extend beyond the Zircalloy tubes thus providing a pilot plug at each end which is inserted into the tube sheet cavities. At the upper end of the fuel assembly, a combination clamp and fuel removal bridge is assembled to the corner fuel end plugs by four hex head cap screws and tab lock washers, thus containing the entire assembly in the tube sheet. A hole is provided in the center of this bridge to accommodate the fuel removal pin if one is used, or to allow the insertion of some other type of fuel removal device such as the ball lock pin type. A similar bridge is assembled to the lower end of the assembly; however, in place of the removal hole, a pilot tube is inserted into the lower grid thus positioning the assembly. A removable upper hold down device confines the core in position during reactor operation.

Internal Structure and Pressure Vessel (Figs. 5.6, 5.7, 5.9)

The internal structure consists of upper and side thermal shields, a lower grid and a fuel containment frame. The upper thermal shields are made of 1% boron steel and are hung from the inside of the closure. The side thermal shields are also 1% boron steel and are designed, as are the upper ones, so that they may be replaced. The side thermal shields are supported by braces welded to the inside walls of the pressure vessel. At the upper end they are held by bolting to screw plates, also welded to the inside of the pressure vessel. The fuel containment frame is made of 0.125 in. thick Zircalloy formed and welded into an assembly of 52 fuel cavities and 12 control rod cavities. The bottom end of the partitions are boxed in except for a round coolant opening. Each unit bottom flange contains four rivet holes which mate with four equivalent holes in the carbon steel lower grid. The lower grid and the Zircalloy fuel containment frame are then riveted together. The lower grid is assembled at the lower end to the inner thermal shield. The upper end of the fuel containment frame contains struts which are also assembled to the inner thermal shield. The thermal shields, lower grid, and fuel containment frame may be assembled outside the reactor and lowered into position.

The pressure vessel is 56 in. inside diameter by 168 in. deep by 2.5 in. thick and is made of SA212B carbon steel. The lower and upper heads are true ellipsoidal shapes with a diameter to depth ratio of 2 to 1. A 16-in. inlet duct is positioned centrally in the lower head and a 16-in. outlet duct is located adjacent, but in the lower cylindrical end of the pressure vessel. The lower ellipsoidal head contains the mounting pads which are similar to those used on the ETR reactor.

Control Rods and Drive Mechanisms

The reference control rod mechanisms are similar to the ANL-29L⁽¹⁾ type except for a shorter stroke. These mechanisms are rugged and are designed for a maximum amount of usage with a minimum amount of servicing. A latching device on each rod permits quick disconnection of the control blade when required. The control blade is a cruciform in shape and is made of 2% boron stainless steel. A follower of the same cross section is comprised of a UO₂ matrix clad with Zircalloy. By widening the control rod slots to 0.625 in., it is possible to use standard 0.5 in. diameter fuel rods as followers. A second alternative is to reduce the thickness of the control blade cruciform and omit the follower. These possible alternate methods are contingent on a physics analysis of the core control.

Fuel Handling and Storage (Fig. 5.1)

New fuel enters the containment shell via the personnel access hatch and is loaded into the reactor by use of a small fuel winch mounted on the main crane carriage.

Unloading irradiated fuel is accomplished by use of a fuel handling coffin mounted on a movable carriage. To unload, the fuel is removed from the core by means of the fuel loading crane and placed in special bracket while still in the reactor pressure vessel. The coffin is moved into position over the bracket and the coffin fuel tool is used to grapple the element. The coffin is lowered into the reactor water by means of screw jacks on the carriage, until its lower end is 2 ft under water. The fuel is then drawn into the coffin, the coffin door is closed and coffin is filled with water. After raising, it is rolled to a position over the discharge chute. The blind flange covering the chute has been previously removed so that the coffin can be lowered and seated upon the chute nozzle. After seating the coffin, the water in it is drained and door opened. The fuel is then lowered into the fuel storage pit. When the fuel is in the pit, it is unlatched and the handling tool is retrieved into the coffin.

The fuel element in the storage pit is taken from under the drop tube to a position where it can be handled by a light crane operating over the pit. The crane is used to move the elements to a boron stainless steel storage rack.

Primary Cooling System

The steam generator is a two-pass shell and U-tube type heat exchanger approximately 84 in. O.D. and 25 ft long. The effective heat transfer surface is composed of 975 Monel tubes per pass, 0.75 in. O.D., #18 BWG, and 20 ft long to a square array of 1.5 in. x .94 in. The carbon steel shell is designed for the full shutoff head of the secondary feed water system pumps. The unit is designed in accordance with ASME code for a design pressure of 1100 psia at a design temperature of 450°F on the primary side and a design pressure of 350 psia at 430°F on the shell side. The only deviation from standard manufacturing practice of similar heat exchangers for non-nuclear applications is the requirement for welding the tubes into the tube sheet. Monel is used for the tubes because of its ability to stand high chloride concentrations and for its excellent corrosion characteristics.

Two vertical full capacity 10,000 gpm constant speed reactor feed pumps, one in stand-by, are incorporated into the design. In the event of a primary pump failure, the plant can continue at full load with the stand-by pump until the next scheduled shutdown. To minimize the possibility of leakage of radioactive primary fluid, demineralized water is injected into a floating ring type seal at 1000 psi with a leak off connection provided for collecting and piping shaft leakage to the retention tanks. The pumps are fitted with stainless steel impellers and low alloy steel casings for the stainless steel primary system.

The primary circuit utilizes ASTM A-106, Grade B piping material with carbon steel valves of the extended stuffing box, back-seated type, and with leak off piping connection provided.

The pressurizer serves to maintain the system pressure above saturation at all times. Electric immersion heaters maintain the required pressure by keeping the water in the pressurizer at the corresponding saturation temperature. A total of 100 kw of heater capacity is provided in the form of cartridge type elements which can be replaced without opening the system.

During periods of load change the average system temperature varies according to direction of load change. This results in a change in volume of water and produces surges in liquid level in the pressurizer. The pressurizer vessel is sized to maintain system pressure within ± 150 psi of normal operating pressure during such load changes. The magnitude of the temperature change is dependent upon rate of load change, system characteristics and the negative temperature coefficient of the reactor. A complete analysis of the system has not been made, but examination of similar systems indicates that allowance for a variation of $\pm 10^\circ\text{F}$ in average system temperature should be adequate. Based on these assumptions, the required volume of the pressurizer is 500 gallons.

The pressurizer vessel is carbon steel designed in accordance with the ASME code for a design pressure of 1100 psia at a design temperature of 560°F.

Water Quality Control

Since corrosion and erosion products from the primary system cannot be eliminated from the reactor system by blowdown, primary coolant purity is maintained by ion exchangers located in a bypass purification loop. Approximately 30 gpm of coolant are extracted from the primary loop at full system pressure and cooled in a regenerative type heat exchanger. After further cooling in a second heat exchanger, the reactor coolant passes through a mixed bed ion exchanger, a strainer, and then is pumped to the outlet side of the reactor. A flow rate of 10 gpm/ft² of bed surface is expected to maintain a reactor coolant purity of 1 ppm. In order to protect system integrity, the system has been installed in duplicate.

The water is maintained at a high pH by lithium, potassium, or ammonium cation resins. Hydrogen gas is dissolved in the reactor coolant to induce recombination of the water dissociated in the reactor core. The hydrogen gas is introduced directly into the primary system.

Primary system make-up water treatment is affected by a demineralization system located in the service building. This system consists of mixed bed demineralizers and the requisite auxiliaries for resin regeneration.

All plant wastes are collected in one of two 3,000 gallon retention tanks located in the containment building where the wastes are monitored prior to discharge to the plant sewer system. Additional on-site radioactive waste disposal facilities have not been incorporated in the present design.

Shielding

The problem of shielding the PWR is mitigated by the stipulation that access to the vapor container is not permitted during reactor operation. The reasoning behind this is that the cost of shielding inside the vapor container so entry can be made outweighs the advantages of entry into the vapor container during operation.

The shielding around the core has been estimated to require two borated thermal shields around the core to protect the pressure vessel from nuclear radiation heating. These shields are shown in Fig. 5.2. Two carbon steel cylindrical shells with 1% boron added and 0.625 in. thick extend above and below the core. Borated steel is used because additional protection is provided for the vessel since capture gammas in the vessel are substantially reduced.

Outside the vessel and the insulation, a layer of steel 3 in. thick is used to protect the concrete. Steel is chosen over lead or other materials since it requires no unusual skills to fabricate, is self-supporting, and has good heat transfer and strength characteristics. The height of the outer shield extends 13.5 ft above the bottom support plate. Vertical cooling tubes are welded to the surface in contact with the concrete. These tubes serve to cool the steel and the concrete. A 4 ft layer of ordinary concrete is placed outside the steel. Additional cooling coils are located about 10 in. into the concrete.

The inlet and outlet pipes penetrate the concrete in a straight line. Additional shielding around these pipes is required since the fuel followers are directly opposite the outlet pipe. Many design choices are available, one of which would be to curve the pipes as they leave the vessel, another would be to provide an internal shadow shield. A need exists, but the form of the shielding requires further study.

The shielding above the core is composed of about 6 ft of water, and 4 in. of 1% borated steel. The borated steel holds down the capture gammas in the vessel head and control rod components. The water provides a good fast neutron removal medium. The vessel head is surrounded with removable concrete sections which are fabricated into quadrants. The inside and outside radius of the sections are 28 and 75 in. respectively. The height of the blocks are about 4 ft. Additional space is available for more shielding if it is found to be necessary. Also shown in Fig. 5.6 directly above the vessel head fitting around the control rods is an alternate steel-masonite shield which, if needed, could be used.

Below the core, the shielding is used to support the vessel. Here a 3 in. thick cast steel disc with cooling passages cast in is shown. The steel provides a firm bearing for the vessel and protects the concrete. Since there is no working space below the vessel, the shielding required is dependent upon the maximum allowable dose for the steel and concrete and the possibility of activation of the ground water below the vessel. The necessary steel structural supports and the concrete base inside and outside the vessel are more than adequate for the shielding requirements.

Since the primary system is activated because of the N^{16} reaction and crud activation, it is necessary to provide about 2 ft of ordinary concrete around the steam generator and any other primary system components which because of their location can cause a dose of greater than 7.5 mr/hr outside the containment vessel. The control room instrumentation is shielded to a degree that holds dose rate to about 1 mr/hr.

Instrumentation

The function of the instrumentation is to provide the operator with enough information to operate safely the plant and to maintain an uninterrupted steam flow. The control scheme is somewhat less complex than an electric power producing plant since the turbine and its auxiliaries are not present. The primary system also has been simplified by using only one loop, a minimum of motor operated valves and natural convection for the removal of decay heat.

The instrumentation for the reactor system includes the usual nuclear indicators such as flux level monitors and period indicators as well as the usual flow, temperature and pressure controllers, recorders, and indicators.

The instrumentation and the control philosophy affect the exact method and extent of controlling the operation of the plant. It is clear that a control philosophy must be developed along the line of providing only the essential instruments for safe operation. More emphasis should be placed on annunciators which alert the operator to a condition requiring correction since complete automatic control is expensive. This change in philosophy can be accepted because of the demonstrated self-controlling ability of reactors like APPR and PWR in the event of complete loss of load.

The total instrument system is assumed to be of a magnitude comparable to that of the EBWR.^(2,3)

Miscellaneous

The heat exchangers installed as a part of the reactor water purification loop serve as shutdown coolers under normal conditions. Under emergency conditions, the reactor decay heat can be removed by a natural circulation circuit established between the overhead storage tank and the shell side of secondary heat exchangers of the water purification system. Under these conditions, the reactor water flow is maintained by the water purification system booster pumps which are served by the emergency power source.

In the event of a control rod malfunction, liquid poison may be injected into the reactor by means of a hand pump located in the control room. The 300-gallon boric acid supply tank is located adjacent to the control room.

The shield cooling system consists of 0.75 in. O.D. cooling tubes at a distance of approximately 1 ft. from a 3 in. thick steel shield embedded in the concrete biological shielding of the reactor to prevent overheating the shield by the absorption of radiation. A shell and tube heat

exchanger capable of dissipating 0.2% of the gross reactor heat, two 30 gpm circulating pumps, and an expansion tank comprise the shield cooling system.

The pressurized water reactor does not require preheating prior to start-up. The reactor is brought up to operating conditions at a rate of 100 to 150°F per hour.

5.2.2 Secondary System (Process Steam) (Figs. 5.4, 5.5)

The design of the secondary system is conventional and based on a return condensate temperature of 150°F and 25% makeup (35,000 lb/hr) at 60°F. The major equipment in the secondary system consists of a deaerating makeup water heater, two deaerating heater drain pumps, and a steam generator blowdown flash tank. Prior to mixing with the plant returns, the makeup water is deaerated and heated with flash steam from the blowdown and a process steam supplement.

The vertical direct-contact deaerating tray type heater incorporates provisions for collecting and venting noncondensable vapors for maintaining a feedwater oxygen control of 0.005 cc/liter. A heater hot water storage capacity of 5,000 gallons is sufficient for approximately one hour of normal operation.

Two full capacity deaerating heat drain pumps, one standby, are of standard materials and design.

A carbon steel blowdown flash tank operating at 3 psig flashes approximately 17% of the reboiler blowdown. The blowdown rate of 4,500 lb/hr is based on a secondary feedwater analysis of 103 ppm total dissolved solids and the shell side of the reboiler operating with 3,000 ppm total dissolved solids.

The plant service water, control air, and station air requirements are met by the requisite equipment located in the service building.

5.2.3 Containment

The containment vessel is cylindrical with a hemispherical dome and an ellipsoidal bottom head. A diameter of 56 ft. and a straight height of 30 ft. 6 in. were chosen in order to provide an adequately spaced plant layout. The vessel is designed for an internal pressure of 27 psig. This is the pressure resulting from an instantaneous release of the water in the primary system to the containment vessel. The atmosphere inside the containment building was assumed to be dry air at 100°F before the accident, and the 30,000 lb. of water in the primary system was considered to be saturated at 430°F. The vessel should be designed for 27 psig and be tested at 125% of design pressure. This overpressure is higher than the

pressure which would result if 25% of the zirconium in the core reacted with water simultaneously with the system rupture. No credit is taken for a reduction in pressure because of the spray system supplied by the overhead tank.

The vessel is constructed of firebox quality steel conforming with ASTM Specification A-201, Grade B. The cylindrical section of the vessel is 0.69 in. thick.

Emergency access into the containment vessel is provided by an air lock consisting of two doors in series with an interlocking arrangement which assures that one door is closed at all times. Access into the containment vessel during normal operation is not permitted.

5.3 Heat Analysis

5.3.1 Steady State

The removal of the 40.6 Mwt of heat created in the reactor core is accomplished by using a two-pass arrangement through the core. The coolant enters at the bottom of the vessel, passes through the center of the core, then flows downward through the outer fuel region, the reflector, and the thermal shields and out of the bottom of the vessel (see Fig. 5.6). No orificing of the individual fuel bundles is made. A two-pass design is used since it produces a higher performance heat removal system than a single-pass arrangement. The two-pass flow yields a temperature increase by a factor of 2.6, at a reduced flow rate of about 2.9 over a single-pass when the system pressure is 1000 psia. The core is accessible from the top since there are no baffles or flow dividers required at the top. The core is essentially split into two radial regions.

In the heat analysis used to establish the operating conditions of the core, it is assumed that the power distribution varied according to a J_0 Bessel function and a cosine function in the radial and axial directions respectively. A total of 15 cm reflector savings is assumed for both the radial and axial directions. The point of maximum surface temperature for the flow velocities considered occurs at the longitudinal mid-point.

It is realized that flux peaking occurs at the outer edge of the core. However, it is assumed that the magnitude of the peaking does not exceed the highest point in the second pass region. This assumption plus the selection of radius at which the second pass begins on the basis of an equivalent area for 28 fuel subassemblies justifies ignoring the power peaking at the outer edge of the core. The choice of 28 subassemblies in the outer region is based on a mechanical design consideration for location of the control rods.

Of the total power, 69% is produced in the first pass and 31% in the second pass. Approximately 15% of the primary flow is by-passed through the reflector, the outer control rods, and the thermal shields. Orificing is used in the top grid to distribute the correct flow through the reflector and thermal shield areas. The individual fuel elements are not orificed.

No hot spot factor analysis was made on the core; instead hot spot factors typical of this type of core and fuel subassembly design are used.⁽⁴⁾ It is believed that the hot channel factors used are reasonable values in light of current practice in the pressurized water reactor field. A hot channel factor as used in this report is defined as the over-all maximum to average power ratio multiplied by the engineering and nuclear variations (hot spot factors) which may occur in a reactor core and within the pressure vessel. Since the two-pass core has a flattening effect on the radial max/ave power ratio, the hot channel factor for the bulk temperature rise, $F_{\Delta t}$ is lower than one might find in other similar designs. For example, if the core were not divided the max/ave radial power ratio would be about 2:1 for the entire core. By using a two-pass core, the effective radial max/ave in the first pass is 1.25:1. Thus, the two-pass hot channel factor used for the fluid temperature rise, $F_{\Delta t}$, is lowered.

The values used for the hot channel factors are listed in Table 5.2.

Table 5.2

HOT CHANNEL FACTORS

Pass	Fluid Temperature Rise, $F_{\Delta t}$	Film Temperature Rise, F_{θ}	Internal Temperature Rise, $F_{\Delta T}$
First	2.25	7.4	5.2
Second	3.11	7.4	5.2

The Dittus-Boelter correlation⁽⁵⁾ for the film coefficient is used in the heat transfer analysis to establish the temperature difference between the fluid and the heat transfer surface. The Jens and Lottes correlation⁽⁵⁾ is used to determine the heat flux required for burnout based on the fluid temperature conditions.

The heat transfer analysis selected the maximum surface temperature in both passes to be equal based on the power and flow area distributions of the chosen configuration and the flow allowance for the reflector, the outer control rods, and the thermal shields. With equal maximum surface temperatures in both passes, there is no overcooling in

the highest heat output channel for either pass. The pressure for the system is dependent upon the maximum surface temperature selected. The system pressure is chosen to be the vapor pressure for the maximum surface temperature. No additional pressure allowance is made at the maximum temperature point because the pressure would have to decrease about 80 psi before fully developed local boiling could occur. This is true because local boiling on the heat transfer surface requires a driving force which is the temperature difference between the surface temperature and the saturation temperature. In this reactor, the driving temperature difference is about 12°F . This temperature difference is adequate for instrumentation and operational variances encountered in practice. This coupled with the established conservatism ensures a safe design.

The size of the pellets is based on a maximum internal pellet temperature of 4250°F at the point where maximum surface temperature occurs. A radial gap of 1.5 mils between pellet and clad is assumed in all cases. The size of the pellet is influenced by the surface temperature, the diametral clearance and the volumetric heat generation rate in the pellet. Another factor which is unknown is the degree of scaling which may occur on the surface of the fuel element. Since the surface of any scale that forms operates at the design surface temperature, local boiling does not occur. The main effect of scaling is to increase the maximum internal pellet temperature; therefore, an allowance of 250°F was made for this situation since a common⁽⁴⁾ design point for the maximum internal pellet temperature is 4500°F . The allowance of 250°F corresponds to a scale of about 0.0024 in. if one uses the conductivity of water for the scale conductivity.

The heat analysis data are given in Table 5.1.

5.3.2 Unsteady State

The primary system has been designed for natural convection in the event of a loss of pumping power and simultaneous shutdown of the reactor. In order to maintain the natural convection, it is necessary to have heat rejected in the steam generator. An auxiliary means of supplying feedwater to the steam generator is necessary for long term power failures. While natural convection is designed into the primary system, the natural convection system is not self-starting. It is felt that this is not a problem since the flow pattern is in all cases established by the pump. If the pump fails before the reactor is critical, there is no heat removal problem. If the pump fails while coming up to power or while the reactor is at full power, the flow direction is established and decay heat rejection in the core is possible without exceeding the surface temperature limitations. The exchange of heat in the steam generator keeps a temperature difference between the two vertical pipes which contain the coolant going to and from the reactor. Since the height of the steam generator above the top of the core is a determining part of the driving head available for flow, the steam

generator is placed in an elevated position. The main operating floor is chosen since the height of the floor above the top of the core provides adequate driving head for circulating the coolant. A condition of turbulent flow exists which materially helps in producing an adequate film coefficient in the core. Consequently, the fuel element surface temperature stays below the normal operating temperature of 530°F.

Auxiliary power is available through the use of a diesel electric power generator. All important instrumentation, lighting and motorized equipment vital to safe operation of the plant receives its power through this auxiliary power source. These units are capable of 15 second startup to full power operation. Additional protection can be provided through the use of batteries for some vital instruments and lighting. The incorporation of the battery supply adds to the expense of the plant. The use of a battery supply depends on the instrumentation philosophy which must be developed.

5.3.3 Startup and Shutdown

A portion of the excess reactivity of a new core is held down by boron in the primary water. The boron is not required at the operating temperature of the core and, therefore, must be removed by ion exchange resins when the reactor is at power. A scram after the boron is removed can be handled by injecting boron again or holding the primary system at temperature.

The primary coolant flow rate is maintained by the pump. The hydraulic characteristics of the primary pump and the system determine the flow rate. No flow control valve is used since the pump head is just enough to supply the proper coolant flow rate for the core.

The reactor is brought to criticality after the operating functions of the prescribed primary and secondary equipment and instrumentation have been carefully checked. The temperature of the system can be raised at a rate of about 100°F per hour with the reactor core operating at low power. This is done to avoid overstressing the plant equipment.

Once the reactor system is heated and operating satisfactorily, the steam generator, which has also been heated during the reactor startup period, is capable of delivering a steam load. The steam temperature and pressure are related to the steam load and the reactor coolant temperature. With the reactor at operating temperature, but low power and low steam load, the pressure rises in the secondary side of the steam generator. A pressure control valve leading to the process steam plant limits the outlet pressure at 196 psia to protect the process steam system. At low loads the steam is superheated slightly. At operating

load and with the reactor at full power and temperature, the steam is at the proper conditions, 196 psia and 380°F. At higher steam demands, the steam pressure decreases according to the characteristics of the systems. Through all these changes, a measure of protection is provided by using pressure regulating devices on the steam generator. The primary system experiences temperature and pressure changes. However, these are adequately accounted for by the pressurizer, the reactor control system, and the inherent safety of the pressurized water system. The general reaction of the reactor to load changes is to increase its power on power demand and to decrease its power when the steam load is decreased. These characteristics result from the negative temperature coefficient of reactivity which protects the reactor system during load changes. Reactors of the pressurized water type have demonstrated their ability to withstand complete loss of load.

During a shutdown of any kind the reactor operator has at his disposal enough information to control the situation. The normal shutdown entails a slow cooling down and lowering of the primary pressure after the control rods have been inserted. Boron injection may not be required when the core reaches an equilibrium fuel status. Natural convection can be used while the reactor is still warm. As the temperature decreases, the water purification cooling system is used to remove the remaining decay heat. This system is also used while the reactor is going through a refueling process of any kind. Natural circulation is not available during refueling because the primary water system must be drained to a point such that the water level is a little below the reactor vessel flange.

With regard to the incorporation of emergency protection, it is recognized that any reasonable safety consideration given to reactors like APPR, PWR, and Yankee can be provided in this reactor design. A careful study is required to outline the most economical method of treating such emergency. An integral part of any study of this type is to determine the reasonableness of proposed accidents and emergencies.

5.4 Physics (PWR)

If a reactor is to be designed to produce heat in economic competition with fossil fuels, the cost of the fuel cycle must be less than that for the fossil-fueled plant, since the capital costs are probably higher than in conventional plants. To this end, those designs yielding the best fuel cycle, i.e., long cycles and high fuel exposures, together with low enrichments must be considered. In reactors moderated by hydrogen, moderate enrichment of about 1.5% - 2.5% is required. The result is a low thermal conversion ratio. Thus, in order to achieve an adequate total conversion ratio for a long fuel cycle, there must be appreciable resonance absorption which necessitates the following general criterion: the fuel to water volume ratio must be as high as possible, i.e., the lattice should be undermoderated.

Some parametric studies were made of boiling and pressurized water lattices and the results are tabulated in Part 4 of Appendix A. Effects observable from varying certain parameters in the boiling systems are generally applicable to the pressurized cores as well.

To put the various lattices studied on a comparable basis, it was decided to set the initial reactivity and then find the fuel life for constant flux exposure (uniform burnup). To approach this situation in practice, it is necessary to shuffle fuel both axially and radially. If axial rearrangement of the fuel does not prove feasible, the life of a fuel element is determined by the portion receiving the peak exposure. The result would be a reduction of roughly 30% in the length of the fuel cycle. Planned shutdowns at approximately half-year intervals should be sufficient for replacing a few spent elements with fresh ones and rearranging the fuel in the core.

The studies showed that fuel cycle characteristics are not particularly sensitive to the fuel rod diameter. The increase in fuel enrichment necessary if stainless steel cladding is used instead of zirconium can also be noted. The recommended core uses rods of about 0.5 in. diameter, zirconium clad and structure, and a high non-water to water ratio (at least 1.2). The physics data are presented in Table 5.1.

In a pressurized core, greater fuel exposure can be obtained from a given initial U^{235} inventory than in a boiling core; however, the fuel cycle time is appreciably shortened.

5.5 Capital Costs

The capital costs of the PWR system are presented in Table 5.3. These costs were prepared by Sargent and Lundy from designs and calculations made jointly by Sargent and Lundy and ANL. At the inception of this joint effort, the preliminary studies had indicated that carbon steel might be a suitable material of construction. As a consequence, Sargent and Lundy were requested to price the proposed PWR system with both stainless steel and carbon steel construction of the primary circuit. The contrasting costs are given in Table 5.3.

The costs presented are current costs, based to a large degree on quotations. The reactor system is simplified since it is taken to be an operating facility where reliable production of low cost steam is the prime factor to be considered.

Table 5.3

PWR CAPITAL COSTS

(Prices as of 5-1-59 and Based on a 40-Hour Work Week)

	<u>Stainless Steel</u>	<u>Carbon and Low Alloy Steel</u>
<u>A-Land</u>	Not Incl.	Not Incl.
<u>B-Structures</u>		
1. Ground Improvements	\$ 25,000	\$ 25,000
2. Reactor Plant Structure	335,000	335,000
3. Service Building (Control Room Offices, etc.)	167,000	167,000
4. Lighting	13,000	13,000
5. Miscellaneous Permanent Structures	17,000	17,000
6. Electrical for Structures	4,000	4,000
Total Structures	\$ 561,000	\$ 561,000
<u>C-Equipment, Piping, Etc.</u>		
1. Equipment	1,115,950	1,007,250
2. Piping and Insulation	225,000	145,000
3. Instruments, Control, and Gauge Boards	118,500	118,500
Total Equipment, Piping, Etc.	\$ 1,489,450	\$ 1,270,750
<u>D-Electrical</u>		
1. Auxiliary Power Equipment	98,000	98,000
2. Temporary Power and Light	8,000	8,000
3. Miscellaneous Power Plant Equipment	6,000	112,000
Total Electrical	\$ 112,000	\$ 112,000
<u>E-Miscellaneous Equipment</u>		
1. Health - Physics	17,000	17,000
2. Office and Locker Rooms	3,500	3,500
3. Machine Shop and Store Rooms	Not Incl.	Not Incl.
4. Fire Fighting Equipment	1,500	1,500
Total Miscellaneous Equipment	\$ 22,000	\$ 22,000
<u>F-Personnel Training Program</u>	Not Incl.	Not Incl.
<u>G-Start-Up Supervision</u>	Not Incl.	Not Incl.
<u>H-Contractor's Overhead and Profit</u>	Incl. Above	Incl. Above
Sub-Total (A through H)	\$ 2,184,450	\$ 1,965,750
<u>I-Contingency - 10%</u>	218,550	196,250
Sub-Total	\$ 2,403,000	\$ 2,162,000
<u>J-Top Charges - 15%</u>	360,000	324,000
<u>K-Allowance for Escalation</u>	Not Incl.	Not Incl.
Grand Total	\$ 2,763,000	\$ 2,486,000

NOTE: Costs prepared by Sargent and Lundy except for reactor vessel, reactor vessel internal structures, control rods and control rod drives. These costs were prepared by ANL.

5.6 Future Status

The design of the PWR presented in this report is based on applying the latest established techniques as well as some features which are the result of the general reactor development program. There are areas requiring research and development exclusive of materials that could improve the design and decrease the cost of the plant.

The present conservatism used in establishing the hot spot factors in a pressurized water reactor are necessary because of the possibility of burnout of the fuel elements. However, capital costs can probably be reduced by applying the results of hot spot data to be obtained from the current PWR family of reactors. Reducing the conservatism reduces flowrate and core size.

Natural convection cooling of a PWR reactor offers some possibilities for producing a lower cost system. The power range where this method is applicable and economical requires study. Natural convection plus a negative temperature coefficient would result in a self-regulating (reactor power as a function of process steam demand) type of system. This type of system would have a self-starting natural circulating system with orificing of the fuel subassemblies located in the low power density areas of the core.

Within the framework of the proposed PWR design, there are a few areas where additional study and design efforts can reduce capital costs. A significant one is in containment. In a conventional sense this is illustrated in Section 6.6 of this report where reduction in containment shell volume is effected. A second profitable method of attack on containment is the use of energy absorption devices to contain the fission products. Sargent and Lundy is, at present, conducting tests along these lines. A second capital cost reduction area in the case of the PWR system is in the reactor vessel internal structure. The proposed design has a significant quantity of zirconium in the core structure. Redesign can eliminate a significant quantity of this or, in some instances, substitute less costly material.

While the proposed PWR should require no significant research and development to put into operation, the operation of a water-cooled reactor having a carbon steel primary system will yield a significant quantity of data on water quality control. A carbon steel system will have more corrosion products than a stainless steel one. However, the magnitude of this increase should not markedly effect the reliable operation of the reactor. The methods of water quality control and the measurements of crud formation, removal and formation of scale will be valuable.

References - Section 5

1. Young, J. N., "Supplementary Letter to ANL-5768," March 31, 1959.
2. "The Experimental Boiling Water Reactor," ANL-5607, May, 1957.
3. "Boiling Water Reactor Program," TID-7535, October, 1957.
4. "Preliminary Hazards Summary Report," YAECD-60, April, 1957.
5. U.S. Atomic Energy Commission, "The Reactor Handbook," Vol. 2, Sec. 1, Light and Heavy Water Cooled Systems, AECD-3646. Also McGraw-Hill Book Company, Inc. New York, 1955.

FIG. 5.1
PWR ELEVATION - SEC. A-A

FIG. 5.2
PWR MAIN FLOOR PLAN

FIG. B.3
PAR BASEMENT FLOOR PLAN

FIG. 5.4
PWR FLOW DIAGRAM

FIG. 5.6
PWR - REACTOR ELEVATION

FIG. 5.7
PWR VESSEL PLAN

FIG. 5.9
PWR CONTROL ROD

6. BOILING WATER SYSTEM

6.1 General Plant Description

The boiling water reactor plant is housed in a steel containment vessel 56 ft diameter and 77 ft over-all height. The general plant layout is illustrated in Figs. 6.2-6.6. This general layout is similar to that described in Section 5.1 of the PWR discussion.

Table 6.1 presents the summary data for the proposed system.

Figure 6.1 is the system flow diagram. Figures 6.7 - 6.12 illustrate certain reactor features.

The BWR system is a reliable proven system. As with the PWR, it can be built without prior research and development. While the particular design illustrated has a stainless steel primary system, there are good possibilities of using carbon or low alloy steels.

Table 6.1

BWR SUMMARY DATA

<u>Power</u> , Mwt	40.6
<u>Core size</u>	
Diameter (nominal), in.	56.3
Length, in.	60
Volume, liters	2442
<u>Coolant and Moderator</u>	Water
Total flow rate through core, lb/hr	7.57×10^6
Total steam rate, lb/hr	1.39×10^5
Total flow area, ft ²	6.8
Flow area per subassembly, ft ²	0.114
Number of passes through core	1
Velocity at inlet, ft/sec	5.95
Recirculation flow ratio	54.5
Reactor pressure, psia	333
Pressure drop in core, psi	0.6

Table 6.1 (Con't.)

Temperature

Inlet, °F	238.8
Outlet, °F	427
Average in core, °F	422
Maximum surface, °F	452
Maximum internal, °F	2900
Log mean temperature difference at steam generator, °F.	46

Heat Transfer

Total surface in core, ft ²	3530
Surface per subassembly, ft ²	58.8
Average heat flux, Btu/(hr)(ft ²)	43,600
Maximum heat flux, Btu/(hr)(ft ²)	130,800
Burnout heat flux, Btu/(hr)(ft ²)	500,000
Average steam volume fraction	
Core exit	0.40
Boiling zone	0.24
Total core	0.18
Slip ratio, steam velocity/water velocity	2
Ratio of boiling length to total length, average	0.76

Fuel

Rod diameter, in.	0.555
Cladding, Zr-2, in.	0.018
Radial helium space (pellet to clad), in.	0.001
UO ₂ pellet density, gm/cc	10.2
Pellet diameter, in.	0.517
Lattice spacing, square pitch, in.	0.630
Number of rods per subassembly	81
Over-all dimensions of subassembly, in.	5.75 x 5.75 x 78.25
Number of subassemblies	60
Total number of rods	4860

Fuel Followers

Tube dimensions	Identical to fuel rods
Number of rods per follower	81
Number of fuel followers	9
Total number of rods	729

Table 6.1 (Cont'd)

Physics

Non-water to water ratio in fuel region	1.2:1
Over-all maximum to average flux	3:1
Cycle length, 100% operating, yrs	6.53
Burnup, 90% of theoretical, Mwd/M.T.	12,300
Fuel enrichment, percent U^{235}	1.5
Total loading of uranium, kg	10,300
End of cycle	
Plutonium present, kg	59
U^{235} present, kg	51
Void coefficient of reactivity, $\frac{\Delta k}{k} / \frac{\Delta \text{void}}{\text{void}}$	-0.1884
Temperature coefficient of reactivity, %/°F	-0.002975

Control Rods

Number in core	9
Shape	Square tube with fuel followers
Tube dimensions, in.	5.75 x 5.75 x 0.25
Length, in.	60
Material	2% boron stainless steel
Drive Mechanism	Magnetic jack
Rod control, cold clean, % k	7.2
Cold clean core, % k	18
Hot clean core, % k	15.8

6.2 Design

6.2.1 Reactor and Primary System

Fuel (Figs. 6.7, 6.8)

The basic fuel element is a rod consisting of UO_2 pellets having a density of 10.2 gms/cm³ and clad with Zr-2 of 0.018 in. thickness. The outside diameter of a rod is 0.555 in. and the active length is 60 in. The fuel rod is closed at both ends with either slotted or threaded end fittings to fasten into either a cross spacer or a cast support grid. A 0.75 in. fission gas and thermal expansion pocket is left at the end of each fuel rod. The clad is calculated to be free standing under vessel hydrostatic testing as well as under operating conditions.

The fuel element consists of a 9 x 9 array of fuel rods with a tip for centering in the support grid and an upper end fitting designed to permit spacing (by springs) and handling of the fuel element.

Sixty fuel elements are required to fill the core under normal full power operation. There are also nine control rod spaces which permit the fueled follower control rods to be inserted and withdrawn from the core.

Internal Structure and Pressure Vessel (Figs. 6.7, 6.10)

The reactor vessel presented follows the Boiler Safety Act of 1952, which states that the design of nuclear boilers shall follow the procedures outlined in Section 1, Power Boilers of the latest revisions of the ASME Boiler and Pressure Vessel Code.

The pressure vessel, illustrated in Fig. 6.10, consists of a cylindrical shell, lower head, ring flange, and closure head; all of which are fabricated from carbon steel and internally clad with a minimum of 0.1875 in. stainless steel. The 2-in. thick, 2:1 elliptical head closure is bolted to the ring flange against two concentric sealing gaskets. The ring flange reduces the vessel nominal inside diameter, 84 in., to a maximum vessel opening of 68 in. This particular opening is sized to accept the core support grid. The 1.625 in. thickness, including clad, of the cylindrical shell and lower head is determined using the maximum allowable stress (17,500 psi at 500°F) of SA-212 B carbon steel, and a weld efficiency of 90%. The height of the vessel, from the bottom to the gasket surface is 25 ft 3 in. with an additional 2 ft 1 in. for the closure head.

The core support grid is a 6-in. thick grid fabricated into an octagonal shape with a minimum distance across flats of 56.5 in. The top surface of the grid is located 6 ft above the bottom of the vessel to accommodate the fueled and guide sections of the control rods. The 6-in. thickness of the grid is not necessary for structural requirements but has been incorporated to act as a guide for the control rods. Support of the grid is accomplished by the 9 perforated control rod guides that extend from the grid to the bottom of the vessel.

Extending upward 10 ft 6 in. from the core support grid is the natural circulation shroud which tightly surrounds the core. The shroud is fabricated from 0.25 in. plate, and is fixed to the grid by means of a bolt and seal arrangement. Sealing of the shroud to the support grid prevents short circuiting of the fluid from the downcomer into the shroud and core due to static pressure differences inherent in the system. The downcomer area (between the shroud and the vessel) is adequate to result in downcomer velocities on the order of 1 ft/sec. Extension of the shroud one core height above the active core enhances circulation for the low power density core.

The 3-in. condensate return distribution ring is located 1 ft. 3 in. below the top of the shroud. Locating the ring in this position

promotes rapid quenching of any steam entrained in the downcomer. The ring is supported by brackets welded to the vessel wall.

The opening of the internal steam duct is 4 in. below the gasket surface of the ring forging. This gives 8 ft 5 in. of distance for two-phase mixture and steam disengagement. The inlet area is sized to be approximately 1.5 times the area of the 6 in. Sch. 80 steam outlet line. The duct is supported by welding the unit, which is fabricated from 0.125 in. plate, directly to the steam outlet line.

Figure 6.7 is a section through the core with the condensate return distribution ring, maximum vessel opening, internal steam baffle, and steam line superimposed to show the core accessibility.

Control Rods and Drive Mechanisms (Fig. 6.8)

The poison rod is a rectangular box 5.75 in. square fabricated of 0.25 in. thick 2% boron stainless steel. An attachment fitting, for connecting to the control rod drive mechanism, is provided at the top of the poison rod and a threaded portion is provided at the lower end for connecting to the fueled follower.

The fueled follower portion of the control rod is the same as the fuel elements previously described except for the end fittings. The lower end fitting is not used in this case and the upper end fitting is replaced by a threaded connector. Tack welding of the fueled follower to the poison rod allows disassembly of the unit. Disassembly of this unit provides for reuse of either the poison or replacement of the fueled section of the rod.

The control rod drives are the magnetic jack type⁽¹⁾ used in the PWR design. They are located on the reactor vessel closure to keep the core as low as possible in the containment vessel and to eliminate the possibility of radioactive crud settling in the drives. In general, the drive provides 60 in. of rod travel at 0.3 in. per minute (operating withdrawal or insertion speed). The scram time is 75% insertion in 0.5 seconds and 100% in 1.4 seconds. Position indication is $\pm .05$ in.

Drives are removed prior to head removal for refueling operations. This is not a serious drawback in that the core design calls for infrequent fuel changes.

Fuel Handling and Storage (Figs. 6.5, 6.9)

The fuel handling system is similar to that described for the PWR, Section 5.2. In addition a shipping cask, Fig. 6.9, is illustrated to accommodate 9 of the particular BWR fuel subassemblies which are 100-day cooled. A 20-ton monorail crane is used to lift the cask into the storage

pit and lower it to the loading position in the pit. The lid is removed and fuel is loaded from the storage rack using the fuel hoist. The lid is replaced, cask lifted, lid bolted and cask moved out for shipment.

Primary Cooling System

The primary system is stainless steel except for the reboiler and drain cooler tubes.

The reboiler is a horizontal shell and tube, single-pass, type with 816, 0.75 in. O.D., 18 BWG, 27 ft long, Monel tubes. The tubes are arranged in a square array with 1.5 in.x0.94 in. pitch. Primary steam is condensed inside the tubes and secondary water is boiled outside the tubes in a 58 in. diameter, 32 ft long, 0.857 in. thick, shell.

The drain cooler is a two-pass shell and U-tube type heat exchanger with 106, 0.625 in. O.D., 18 BWG, 30.3 ft long, Monel tubes. The shell is 16 in. O.D. and 33 ft long.

Two 280 gpm, horizontal, vertically split case, hot process refinery type pumps, with mechanical face seals and leak collection fittings are provided. Two pumps assure ability to operate at full load until the next scheduled shutdown in the event one pump fails. If the system were rigorously designed for natural circulation, one pump would be sufficient.

Water Quality Control

The system is filled initially with high purity water effluent from mixed bed ion exchange columns. Maintaining water purity requires removal of undissolved solids principally iron, chromium, and nickel oxides, and removal of dissolved solids. Maintaining low concentrations of undissolved solids retards ion formation.

Gaseous ions formed in the core, by neutron bombardment, are a particular problem in boiling water reactors in that gases are carried out of solution with the steam. Recent experiments in EBWR^(2, 3) have shown that hydrogen addition to the boiler water greatly enhances recombination to the degree that the oxygen content of the steam is reduced a factor of 100 by the addition of only 50 cc (STP) per liter of condensate return water.

The high pressure condenser of this system lends itself well to hydrogen overpressure in the primary system as there is no air leakage. To maintain excess hydrogen, it is necessary to collect gases vented from the condenser and pump them back to the boiler water via the condensate return and distribution system. From reference data,⁽³⁾ it is estimated that less than 0.3 cfm of gas need be handled.

The system to provide hydrogen overpressure consists of a rack of hydrogen bottles for initial injection and a loop for recirculation of hydrogen vented from the condenser. The recirculation loop takes condenser vent gases to a small heat exchanger where the remaining steam is condensed. A small compressor is used to pump the gas back to the reactor water. The gas is monitored for fission products enroute. If it is desired to operate with ruptured fuel, connections are provided in the gas cleanup to keep fission product gases from overloading the system.

A continuous 10 gpm side stream purification system is used to maintain system purity. This loop takes reactor water from the lowest point in the vessel to collect particulate precipitates. The water is cooled in a regenerative heat exchanger and run through a mixed bed ion exchanger resin column and filter that are located in series. The ion exchange resin beds have after filters but no prefilters since the resins are not regenerated.

Shielding (Figs. 6.11, 6.12)

The shield design is based on the philosophy of preventing overshielding. To accomplish this, a radiation level of 100 mr/hr (outside the unpenetrated shield) is used as the design criterion. However, levels of 10-50 mr/hr may be anticipated due to unknowns in the shield calculations. The shield is adequate for limited accessibility during operation. Space has been provided for extra shielding if it is found necessary after actual operating levels have been measured.

Radially, the entire vessel is surrounded by 3 in. of insulation and 1 in. of lead bonded to 0.5 in. carbon steel plate. The steel serves the two-fold purpose of canning the insulation and of physical support of the lead. Next, vertical cooling tubes are evenly spaced around the entire reactor. Following the lead, in the axial down direction, cooling tubes are provided. The entire reactor, with the exception of a band encompassing the core, is covered with 4 ft of ordinary concrete. In the band encompassing the core, an additional 2 in. of lead in the form of stacked bricks, 8 ft 3 in. high is symmetrically placed around the horizontal centerline of the core. Also in this region, an additional band of concrete 3 ft thick (making a total of 7 ft of ordinary concrete) is provided for a distance of 8 ft 1.5 in. on each side of the core centerline. A third set of cooling tubes is located 1 ft from the inner surface of the 7 ft concrete section. The total heat removed by the three sets of cooling tubes is approximately 40 kw.

To provide the required shielding, in the axial-up direction, a composite water and steel shield and removable concrete blocks are used. The composite shield is made of 3 in. of steel and 6 in. of water alternately spaced to a total thickness of 2 feet. This shield is permanently fixed to the closure head and weighs 6,500 lbs. Surrounding the composite shield

and magnetic jacks are four removable ordinary concrete blocks each weighing approximately 12,000 lbs. Each block is a 90° segment of a 6 ft 6 in. hollow cylinder with a 4 ft inside and 11 ft outside diameter. Provisions have been made in the design to store the concrete shield block below the operating floor during fuel handling or other reactor maintenance.

Instrumentation

This is similar to the discussion in Section 5.2.

Miscellaneous

The miscellaneous items on the BWR are similar to those of the PWR discussed in Section 5.2. The quantity of decay heat which must be handled is given in Fig. 6.13.

6.2.2 Secondary System (Process Steam) (Fig. 6.1)

The secondary steam system is identical to the steam system of the PWR previously described with the exception that BWR uses a drain cooler to subcool the primary condensate by preheating the secondary water.

6.2.3 Containment

The containment presented for this design employs standard criteria for design as set forth by the precedent systems or other similar boiling water reactors. The method of analysis is comparable to that in Section 5.2.3. The maximum credible incident assumes rupture of the primary system with a simultaneous water-zirconium reaction. Twenty-five per cent of the zirconium fuel clad is assumed to react.

The weight of saturated water in the primary system is 37,700 lb. There is 2000 lb. of zirconium clad on the fuel, 500 lb. of which produce 1.2×10^6 Btu upon reaction with water. This heat is equivalent to an additional 17,000 lb. of saturated water. The 56 ft diameter cylindrical containment vessel affords 127,100 cu. ft of free volume. The containment vessel, thus, provides 3.37 cu. ft per lb. of primary water contained. The following are tabular results of various rupture conditions as determined from reference 3.

Table 6.2

CONTAINMENT CONDITIONS

Initial Condition

Water	Saturated 333 psia
Air	Dry 100°F

Condition of air-vapor mix in containment after burst of primary system

Pressure	44.5 psia
Temperature	240°F
Cylindrical vessel thickness	0.75 in. 56 ft dia.

Condition of air-vapor mix in containment after primary system burst plus water Zr reaction

Pressure	55 psia
Temperature	260°F
Cylindrical vessel thickness	1.0 in. 56 ft dia.

The design pressure of the containment vessel is 45 psia. A simultaneous burst and zirconium water reactions would cause 25% overpressure. Further rupture of the secondary system would cause the vessel pressure to rise to 65 psig. The rupture pressure of the vessel is 120 psig. The 25% overpressure is proven by standard pressure test. The remote possibility of all three occurring simultaneously would still result in only 54% of the rupture pressure. Fifteen thousand psi is taken as the allowable stress and 60,000 psi minimum tensile rupture stress.

The integrity of the containment vessel is further safeguarded by the fact that over 20,000 sq. ft of steel condensing surface is present in the building. The steel surface is capable of absorbing 560,000 Btu per 100°F temperature rise. The maximum rate of heat release possible through a complete failure of the 10-in. steam line is 282,000 Btu/sec. If the steel surface condensed at a rate of only 50,000 Btu/(ft²)(hr), there would be little pressure rise in the shell until the steel temperature increases 100°F (2 sec). The vessel would not reach peak pressure for 25 seconds which leaves time for concrete surfaces and the 15,000 gallon water storage to be useful as heat sinks.

6.3 Heat Analysis

6.3.1 Steady State

No steady state heat analysis is supplied since the core design results in less severe operating conditions than found in the EBWR. The reactor operating conditions are given in Table 6.1.

6.3.2 Unsteady State

In the event of total loss of power, the decay heat is removed by a heat exchanger tube bundle in the overhead storage tank which condenses primary steam and returns the condensate by natural circulation to the reactor vessel. This condenser is also used to accept minor overpressure fluctuation of the primary system. This emergency condenser can condense at the rate of 2.5 Mw. In the event of loss of normal power, a diesel generator is provided to obtain decay heat removal using the purification system coolers.

6.3.3 Startup, Shutdown and Control

Itemized startup procedure must be drawn up after detailed design study. In general the following steps will be taken.

1. Check all equipment to assure readiness for startup.
2. Check secondary steam back pressure control valves.
3. Heat up primary and secondary systems with nuclear heat, 100°F/hr to saturation temperature and operating pressure.
4. Lower primary water level to 1 ft above chimney of the shroud.
5. Reactor is brought to the desired power by withdrawal of control rods.

The control system is based upon use of the control rods as the means of matching reactor power to steam demand. Control rods are moved automatically to maintain a constant primary system pressure. The steam valve of the secondary system is operated to give the desired steam flow.

Increase in steam demand lowers secondary system pressure and temperature thus increasing the rate of condensation in the reboiler. The primary system pressure drops and the void fraction in the core is increased thereby reducing reactor power. Rods are withdrawn until reactor power output increases sufficiently to maintain primary system pressure of the preset value.

Decrease in steam demand raises secondary system pressure and temperature. Condensation of primary system steam is reduced and the primary system pressure increases. Rods are slowly inserted decreasing reactor power. Experience with the EBWR⁽⁵⁾ indicates effects of the extreme case of complete valve closure at 20 Mw operation. The pressure rises at a rate of approximately 3 psi per second. The closure of steam valves must, therefore, not exceed the rate that reactivity can be removed by control rod insertion. If a process must shut down rapidly, rods can be selectively scrammed, or secondary steam can be dumped to the stack. Rods that operate faster than the magnetic jacks, such as rack and pinion may be used. However, magnetic jacks are specified because secondary steam flow is assumed to remain essentially constant.

A method of control that should also be considered, if the process permits, is as follows. Maintain secondary steam pressure in reboiler constant by a back pressure control valve and use the control rod directly to vary the steam supplied to the process. The steam demand of the process should be well indicated by the pressure in the line downstream from the backpressure regulator. The rate of rod motion must be regulated to prevent excessive primary system pressures. If process cuts off demand at a rate too rapid for normal rod insertion, rods can be scrammed in a programmed sequence.

A degree of control can be achieved by by-passing the drain cooler. The resultant increase in primary condensate temperature reduces subcooling of the feedwater, thus reducing reactor power. The same would result with a reduction of secondary condensate return rate or makeup rate.

6.4 Physics

The physics considerations involved with the boiling cores were examined in conjunction with the PWR studies. The same goals and conclusions are applicable. These are based on high burnup core and fuel changes only after long term operation.

The size of the boiling core is about double that of the pressurized water system. This calls for a higher total inventory but a lower enrichment. The result is a long fuel cycle at some sacrifice in the maximum fuel exposure based on reactivity.

6.5 Capital Costs

The capital costs are presented in Table 6.3. The development background of these costs is the same as those discussed in Section 5.5.

Table 6.3

BWR CAPITAL COSTS

(Prices as of 5-1-59 and Based on a 40-Hour Work Week)

<u>A-Land</u>	Not Incl.
<u>B-Structures</u>	
1. Ground Improvements	\$ 25,000
2. Reactor Plant Structure	390,000
3. Service Building (Control Room, Offices, etc.)	167,000
4. Lighting	13,500
5. Miscellaneous Permanent Structures	17,000
6. Electrical for Structures	5,000
Total Structures	\$ 617,500
<u>C-Equipment, Piping, etc.</u>	
1. Equipment	759,050
2. Piping and Insulation	150,000
3. Instruments, Control, and Gauge Boards	118,500
Total Equipment, Piping, etc.	\$1,027,550
<u>D-Electrical</u>	
1. Auxiliary Power Equipment	80,000
2. Temporary Power and Light	7,000
3. Miscellaneous Power Plant Equipment	6,000
Total Electrical	\$ 93,000
<u>E-Miscellaneous Equipment</u>	
1. Health - Physics	17,000
2. Office and Locker Rooms	3,500
3. Machine Shop and Store Rooms	Not Incl.
4. Fire Fighting Equipment	1,500
Total Miscellaneous Equipment	\$ 22,000
<u>F-Personnel Training Program</u>	Not Incl.
<u>G-Start-up Supervision</u>	Not Incl.
<u>H-Contractor's Overhead and Profit</u>	Incl. Above
Sub-total (A through H)	\$1,760,050
<u>I-Contingency - 10%</u>	\$ 175,950
Sub-total	\$ 1,936,000
<u>J-Top Charges - 15%</u>	\$ 290,000
<u>K-Allowance for Escalation</u>	Not Incl.
Grand Total	\$ 2,226,000

Note: All costs prepared by Sargent and Lundy except for reactor vessel, reactor vessel internal structures, control rods, and control rod drives. These costs were prepared by ANL.

6.6 Future Status

The design of the BWR is an adaptation of present technology to a reactor core having high burnup. Since the steam demand is assumed constant, a single cycle plant is used. With widely fluctuating steam demand, a dual cycle BWR might be used to advantage.

As with the PWR system, the cost of the illustrated BWR system can be reduced by further study and design effort in the containment area. An illustration of reduction of the conventional pressure vessel type is give in Figs. 6.14 and 6.15. Consideration might also be given to relocation of the reboiler outside the containment vessel.

Improvements in containment will be possible upon accumulation of experimental data from tests of large ruptures with containment shells. Experiments are needed to permit the designer to take credit for effects of (1) condensation on shell, equipment, and shielding; (2) building spray system; (3) rupture disks to let containment exhaust to a pond of water; (4) the decontamination factor accreditable to vaporization process; (5) use of higher allowable stresses for once in a lifetime loads.

The arrangement of this system is such that the condensate could return to the reactor vessel by gravity drain. However, core reactivity is sensitive to rate of condensate return because of its affect on reactor subcooling. Any instability of this flow could perturb the core, therefore, a pump and valve arrangement is provided. It is felt that the prototype reactor should be arranged to permit natural circulation experiments. A simple surge tank with or without valve control may suffice.

The pressure vessel sizes could be reduced by studies along the following lines:

1. Use of a reduced area chimney
2. Determination of velocity effect on steam entrainment in the downcomer.
3. Determination of power effect and other parameters on the surface level of the water steam mixture above the core.
4. Toleration of wetter steam or incorporation of internal scrubbers to reduce required steam dome.

Pressure vessel costs could be reduced by use of an unclad vessel with a low alloy steel system.

References - Section 6

1. Young, J. N., "Design and Performance Characteristics of Magnetic Jack Type Control Rod Drive," ANL-5768, December, 1957.
2. "The Experimental Boiling Water Reactor," ANL-5607, May, 1957
3. Gavin, A. P., et al., "Some Chemical Investigation on Boiling Water Reactor," ANL Paper for 135th National ACA Meeting, Boston, Massachusetts, April 5-10, 1959.
4. Heap, J. C., "Equilibrium P-V-T Relation for Expanding Liquid Vapor Systems in a Containment Shell," ANL-5828, November, 1958.
5. Bergstrom, R. N., and W. A. Chittenden, "Reactor Containment," Nucleonics, Vol. 17, No. 4, Pg. 86, 1959.

FIG. 6.1
BWR FLOW DIAGRAM

FIG. 6.2
BWR MAIN FLOOR PLAN

FIG. 6.3
BWR INTERMEDIATE FLOOR PLAN

FIG. 6.4
BWR BASEMENT FLOOR PLAN

SECTION A-A

SECTION B-B

FIG. 6 5
BWR ELEVATION - SEC. A-A & B-B

FIG. 6.6
BWR ELEVATION - SEC, C-C

NOTE:
FOR SECTION "A"- "A" SEE FIGURE 6.10

FIG. 6.7
BWR VESSEL PLAN

FIG. 6.8
BWR FUEL ELEMENT & FUELED FOLLOWER CONTROL ROD

LEAD THICKNESS (TOP & BOTTOM)	=	6 INCHES
OVERALL HEIGHT	=	80 INCHES
EMPTY WEIGHT	=	28,900 LBS.
WEIGHT (WITH 9 ELEMENTS)	=	32,070 LBS.

FIG. 6.9

BWR SPENT FUEL SHIPPING CASK

FIG. 6.10
BWR REACTOR VESSEL & INTERNALS

FIG. 6.11
BWR REACTOR SHIELDING

FIG. 6.13
DECAY HEAT RATES
45 MW REACTOR

FIG. 6.14
CONTAINMENT REDUCTION STUDY

FREE VOL. = 55,000 FT.³
 PRESSURE = 90 p.s.i.a.
 SHELL THICKNESS = 7/8 INCH

FIG. 6.15
CONTAINMENT REDUCTION STUDY

NOTE: P DENOTES PSIA

FIG. 6.16
BWR HEAT BALANCE

7. ORGANIC MODERATED SYSTEM

7.1 General Plant Description

The reactor is similar to the design used by Atomics International for the 45 Mwt reactor for the city of Piqua, Ohio.⁽¹⁾ The modifications to the Piqua design are based upon the lower temperature requirements of the process steam system and the elimination of the electrical power system. The low temperature requirements of the process steam reactor permit the use of diphenyl instead of Santowax R as the organic coolant. The higher vapor pressure of diphenyl at elevated temperature prevented its use as the Piqua coolant.

The building illustrated in Figs. 7.6 - 7.8 is an insulated sheet steel structure above grade and a reinforced concrete structure below grade. The steel portion is sealed so that the leakage rate through the wall is low. The reactor and its major operating equipment are below grade. The first floor, at grade level, is a 2 ft. thick slab of concrete. The control room and necessary space for operating personnel are located on the first floor at the opposite end of the building from the reactor area.

The system parameters and flow diagram are indicated in Figs. 7.1 and 7.2. The system, as arranged, can be built without research and development work. The system data are given in Table 7.1. Detailed description of the various portions of the systems are given later in this report with the merits which may be attributed to each item.

Table 7.1

OMR SUMMARY DATA

<u>Power, Mwt</u>	40.6
<u>Core Size</u>	
Diameter (nominal), in.	48
Length, in.	48
Volume, liters	1475
<u>Coolant and Moderator</u>	Diphenyl
Total flow rate, lbs/hr	7.9×10^6
Total flow area, ft ²	3.1
Flow area per subassembly, ft ²	0.06
Number of passes through core	1
Velocity, ft/sec	12.5
Pressure at core inlet, psia	75
Pressure drop in core, psi	35

Table 7.1 (Con't.)

Temperature

Inlet, °F	420.8
Outlet, °F	450
Total rise, F	29.2
Average in core, °F	435.4
Maximum surface, °F	550
Maximum internal, °F	600
Log mean temperature difference, °F	54

Heat Transfer

Active surface in core, ft ²	8350
Surface per subassembly, ft ²	160
Average heat flux, Btu/(hr)(ft ²)	16,500
Maximum heat flux, Btu/(hr)(ft ²)	76,000
Burnout heat flux, Btu/(hr)(ft ²)	4.7 x 10 ⁵
Hot channel factors	
Bulk temperature rise, F Δt	2.80
Film temperature rise, F θ	3.65
Internal temperature rise, F ΔT	4.60

Fuel

Meat	U metal
	W/3.5% Mo and 0.5% Si
Thickness, in.	0.150
Width, in.	5.5
Cladding, aluminum, in.	0.030
Fins	
Number per side of plate	60
Shape	Triangular
Base, in.	0.060
Height, in.	0.135
Efficiency, percent	88
Plate spacing, in.	0.5
Number of plates per subassembly	48
Over-all dimensions of subassembly, in.	6 x 6 x 72
Number of subassemblies	52
Total number of plates	2496

Physics

Non-organic to organic ratio in fuel region	1:1
Over-all maximum to average flux	3:1
Cycle length, 100% operating, yrs.	4
Burnup, 90% of theoretical, Mwd/M.T.	8900
Fuel enrichment, percent U ²³⁵	1.5

Table 7.1 (Con't.)

Physics (Con't.)

Total loading of uranium, kg	7650
End of cycle	
Plutonium present, kg	50
U ²³⁵ present, kg	39
Void coefficient of reactivity, $\frac{\Delta k}{k} / \frac{\Delta \text{void}}{\text{void}}$	negative
Temperature coefficient of reactivity, %/°F	-0.014

Control Rods

Number in core	9
Shape	Cruciform
Length, in.	72
Material	2% boron stainless steel
Drive mechanism	Rack and pinon
Rod control, cold clean, %k	17.9
Reactivity available, cold, clean, %k	15.6
Reactivity available, hot, clean, operating, %	12.8
Reactivity to be controlled	
Xe (equilibrium), %k	3.8
Sm (equilibrium), %k	2.0
Burnup, 2.5 yrs., %k	3.8
(4.0 yrs.), %k	6.7
Temperature, 450F to 250F, %k	2.8
Temperature, 250F to 80F, %k	2.4
Margin for shutdown, %k	2.0
Control required, hot, clean, operating rods out, %k	14.4
Minimum available reactivity	
hot, clean, operating, %k	9.6
Maximum control for complete shutdown, cold, clean, %k	15.6

7.2 Design

7.2.1 Reactor and Primary System

Fuel (Fig. 7.4, 7.5)

The fuel element design is a low alloy uranium plate with finned aluminum clad.⁽²⁾ The alterations to this Atomics International design modified the top fitting to include an element latching device and a spring added on the bottom to accommodate fuel expansion. A fuel sub-assembly consists of a lower locating end fitting, 48 fuel plates, a 6 in. x 6 in. hollow square stainless steel container, and a top fuel handling and hold-down

fitting. A core loading consists of 52 such fuel subassemblies. The locating end fitting is fabricated of carbon steel to form a 4 in. O.D. cylinder at the location of the core support plate and a square section at the other end to position the fuel elements and the square stainless steel sheath.

A plate type fuel element design is chosen for the process steam application. The fuel plate elements are slightly enriched U-3.5 w/o Mo-0.5 w/o Si alloy and clad with 2S aluminum. Nickel plating is used as a bonding agent between the uranium and the clad. The nickel also acts as a diffusion barrier for the uranium.

The molybdenum in the uranium alloy increases the strength of the uranium and its resistance to growth and irradiation damage. At operating temperatures, the aluminum has very little strength and, therefore, it expands and contracts with the metal fuel. The uranium metal alloy provides the structural strength for the fuel element plates. The choice of the metal fuel over an oxide fuel is based on the greater heat transfer area required by the organic coolant. The heat transfer coefficients obtained with diphenyl at these temperatures are appreciably less than those obtainable in a boiling or pressurized water system. In order to obtain sufficient heat transfer area with relatively thick plates, the finned surface is required.

The radiation damage encountered in metal alloy fuels is the limiting factor on fuel lifetime. The phenomenon which most probably limits the life of the fuel is swelling of the metal due to accumulation of gaseous fission products. At the present time, high burnup capability (approximately 2% burnup) has been demonstrated by uranium - 10 w/o Mo alloys.⁽³⁾ However, the high molybdenum content requires increased enrichment. The lower enrichment permitted by lower molybdenum content makes alloy fuels more economically competitive with the longer burnup oxide fuels. The possibility of achieving higher burnup with lower alloys has been attractive for economic reasons. Metallurgical research and development work is being performed to investigate various alloying additives which hold promise for increasing the fuel lifetime of low alloy fuels.

Internal Structure and Pressure Vessel (Figs. 7.3, 7.5)

The inner thermal shield supports the reactor core structure. The bottom core support plate is fabricated of carbon steel, 5 in. thick and 5 ft. in diameter, and welded to the thermal shield. The plate contains 52 fuel assembly location holes and 9 orificed control rod holes to regulate coolant flow to the control rods.

The upper core grid plate is a lattice structure fabricated of 0.25 in. thick carbon steel and welded to the inner thermal shield approximately 6 ft. above the core support plate. The open lattice structure acts as a guide for the fuel elements and control rods. The fuel element latch bears on the underside of the core grid plate. A duplicate of the core grid plate, the guide grid, is positioned approximately 4 ft. below the top flange of the reactor vessel and serves as a guide for the fuel handling equipment.

The reactor vessel (Fig. 7.3) is a cylinder with a hemispherical bottom head and a elliptical top head fabricated from SA-212B carbon steel. The cylindrical portion of the vessel has a wall thickness of 0.625 in. The overall vessel height is approximately 30 ft. and the dry weight (excluding any vessel internals and thermal shield) is approximately 8 tons. The vessel design, with some modification is taken from the OMR Piqua vessel design.⁽²⁾ The vessel is modified to accommodate a surge dome at the top of the vessel, top head control rod drives, and elliptical top head. The surge dome at the top of the vessel eliminates the need of a separate surge tank. The liquid level is maintained by pressure from a nitrogen gas system. This type of system is also similar to the OMRE design.⁽⁴⁾

The elliptical top head is bolted to the vessel and sealed by a spiral wound stainless steel and asbestos gasket. A reinforcing boss compensates for the 9 control rod drive penetrations through the head closure.

The two vessel outlet nozzles are located 180 degrees apart and 9 ft. below the head closure. The two inlet nozzles are located 7 ft. directly below the outlet nozzles. The coolant mixing baffle in the upper portion of the reactor assures thorough mixing and uniform temperature in the organic fluid above the reactor core.

Control Rods and Drive Mechanisms

The control rods are fabricated from 0.25 in. thick 2% boron stainless steel sheet in a 10 in. by 10 in. cruciform shape. The top of the rod terminates in a fitting which furnishes a means of latching the end to the control rod drive mechanism. A small guide encloses the edges of each control rod from the core support plate to the upper grid plate to guide the rod. Cooling is provided by allowing a portion of the coolant to flow up through the core support plate and shroud and then out the upper grid plate.

The control rod drives are of the conventional rack and pinion design. The rack and pinion are located in a pressure tube mounted on the pressure vessel head. A small stream of organic coolant maintains lubrication of the parts in the pressure tube and bleeds through an orifice into the reactor vessel. The pressure tube is steam traced to prevent solidification of the coolant.

The control rod latch mechanism is operated by a latch magnet which, when de-energized, releases the control rod from the drive, and the control rod falls freely into the core. The velocity of the falling rod is reduced during the final 6 in. of travel by the action of a hydraulic cushion in the core support plate.

Position indication is achieved through a synchro transmitter and receiver operating from the drive motor through a precision reduction gear.

The method of approach to reactor control is satisfactory as it is typical of the design used in the OMRE, EBWR, and other reactors. The design of the control rods and drives is similar to that of OMRE and EBWR.

Fuel Handling and Storage

The fuel plates are inserted in the stainless steel sheath and the top end fittings welded in place. The top end fitting is equipped for fuel handling and element hold-down. The fitting has a small sleeve with 4 pins at 90 degrees to each other and a rod across the diameter. The fuel handling tool grasps the rod and upon turning the rod, retracts the 4 pins into the end fitting, thereby releasing the element from the core structure. The upper guide grid in the top of the reactor aids in locating the fuel with respect to the core position as the organic fluid is opaque. The fuel is withdrawn into a fuel transfer coffin which has an index mechanism corresponding to the guide grid for accurately positioning the coffin over the element. The top head of the reactor is replaced by an indexing plug and head for locating fuel during reloading operation. This head also seals the reactor from personnel as the organic fluid is maintained at a high temperature during shutdown periods and the organic vapors have a degree of toxicity.

The fuel handling coffin carriage is operated on rails set in the concrete floor above the reactor. After raising an element into the coffin, the coffin is rolled over to the fuel transfer slot. In the fuel transfer slot, the fuel is cleaned of organic coolant before transfer to the water-filled fuel storage pool.

The reactor is maintained on a six-month fuel cycle, i.e., every six months the reactor is shutdown for the rearrangement, removal, or addition of fuel elements. The shutdown period of two weeks enables the reactor to maintain a plant factor of 92%. Any plant maintenance and repair is also performed during this shutdown period.

As the core life is estimated to be 4 years, approximately one-eighth of the core is replaced during each shutdown period with new fuel and fuel which has been rearranged axially in the fuel element sheath. The remainder of the core is rearranged radially to obtain the best fuel economy. The fuel which has been removed from the reactor for first time is sent to the reactor fuel storage pit for disassembly and axial reorientation of the fuel in the fuel element sheath. The second time the fuel is removed from the reactor, it is sent for reprocessing.

Axial reorientation of the fuel in the sheath is accomplished in the fuel storage pit in the following manner: The top fuel handling fitting is removed from the element sheath by cutting the weld joint. After removal of the fitting, the fuel is pushed up from the bottom of the fuel element while the sheath is held fixed. The two vertical fuel

sections are interchanged in position after removal, reinserted into the sheath, and the top fitting welded in position. The axial movement of fuel enables the attainment of high burnup. Axial reorientation is felt necessary as the effective core life is reduced by 30% without this feature.

Primary Cooling System

The decision to use diphenyl is based on the following:

(a) the melting point of diphenyl is one-half that of Santowax R and, therefore, has less steam tracing and preheating requirements, (b) the heat transfer characteristics of diphenyl are 20 to 30% better than Santowax R and (c) the pumping power requirements are 12% lower. The pyrolytic effects are insignificant at temperatures below 600°F and, therefore, is not a factor at these temperatures.⁽⁵⁾

The boiler is a conventional horizontal shell and tube reboiler type with organic coolant on the tube side, and water and steam on the shell side. The boiler is a two-pass shell and U-tube type having a carbon steel shell and Monel tubes. The tubes are 0.625 in. O.D. #14 BWG. The boiler is equipped with a steam separator, pressure relief valves, steam dump valves, and feedwater distribution system to eliminate excessive cooling of the organic fluid. The tubes do not have to be welded in because the primary side pressure is lower than secondary side and any in-leakage to the organic fluid is disposed of by the degassification system.

The centrifugal pumps are of steel construction and are of a type commonly used in Dowtherm or hot oil service. Spare pumps are provided. Valves are of steel construction and the piping is SA-106-A carbon steel.

The 75 psi operating pressure of the primary system is maintained by pressure control valves located in the line to the degassing tank. Variations in pressure are handled by increasing or decreasing the flow through the valve. The coolant flowing into the degassifier tank is replaced into the system by one or both of two 200 GPM centrifugal pressurizing pumps. The pumps are standard vertical, sump-type pumps equipped with mechanical seals.

This method of controlling pressure is not unique to the OMR type system. A similar system has been recommended for a pressurized water reactor.⁽⁶⁾

Any gases or vapors produced in the organic coolant due to radiolytic decomposition of the organic fluid, gases diffused into the system by the surge dome in the reactor vessel, or small water or steam leaks are continuously removed by the degassifier system. The degassing tank also serves as the expansion tank for the system. System surges are accommodated by the surge dome in the reactor vessel.

Purification System

The purification system is sized for a purifying 500 lb/hr of organic coolant by a continuous distillation method. The system provides a means for maintaining the high boiler residue content of the organic coolant at a predetermined level and purifies the new coolant before its injection into the system.^{(1),(2)} The purification system also decontaminates the coolant as most of the radioactivity remains with the residue. The average formation rate of residue for the process steam reactor is estimated to be the same as the Piqua reactor, 50 lbs/hr.

Coolant from the primary system or new coolant is sent to the flash feed heater where it is heated to 750°F before being sent to the distillation column. The feed flashes in the distillation column and leaves any residue in the bottom of column to be withdrawn to the waste tanks. The overhead vapors are condensed and flow to the product receiving tanks before returning to the system.

The tar or residue from the purification system is transferred to the waste disposal system where it is held in storage tanks to permit decay of the radioactive elements in the residue. A holdup time of six months is considered adequate. After sufficient decay, the residue, which is a heavy tar-like hydrocarbon, is burned. The combustion gases are filtered to remove any long-lived radioactivity.

The purification system employs standard oil refinery distillation processes. The tar disposal method should provide an adequate method of disposing of waste and a possibility of utilizing the combustion heat for space heating or providing steam for primary system heating during startup and shutdown periods. This method also allows some cost savings on eliminating the cost of packaging and burial of waste material such as is done with ion exchange resins used in water reactor systems.

Shielding (Fig. 7.3)

The thermal shield on the reactor is divided into two parts, one portion inside and the other outside the reactor vessel. The inner shield is 1.5 in. thick. It is cooled by the influent organic flowing downward between the thermal shield and the reactor vessel. The outer thermal shield is 6 in. thick carbon steel cylinder adjacent to and supported by the reactor vessel and extends in a region 1 ft. above the upper grid plate to 1 ft. below the lower grid plate.

Surrounding the reactor vessel from the vessel support ring down is a 0.375 in. carbon steel tank. Between the tank and the reactor vessel is a layer of 4 in. stainless steel wool insulation and a 3 in. air gap, except in the region of the outer thermal shield and bottom head. Cooling coils are used on the outer surface of the tank and are embedded in the concrete biological shield.

The biological shielding requirements for the organic reactors are comparable to that of the water reactors. External to the tank is a column of ordinary concrete 8 ft. thick which serves as the biological shield and the reactor vessel support structure. Shielding above the vessel consists of the organic fluid, the top head closure, and 1 ft. thick concrete shielding blocks. A height of 13 ft. (approximately) is required to reduce the radiation level at the surface of the organic fluid to 7.5 mr/hr 30 minutes after shutdown following 300 days of full operation. This height may be reduced to conserve on reactor vessel structure and the steel and concrete shielding increased above the reactor top to compensate for the reduction in organic shielding. The latter is a design and economic investigation and is too involved for this study. Shielding below the vessel consists of the organic fluid, the vessel walls and concrete. Bags of lead shot are placed between the control rod drive thimbles for shielding this area.

Since the coolant has low activation properties, shielding for inspection and minor maintenance during operation is easily provided. Inspection can be allowed once each day. Major repairs would be performed only on shutdown.

Instrumentation

The instrument systems were not studied. These are assumed to be of comparable magnitude to those of the PWR and BWR.

7.2.2 Secondary System (Process Steam) (Figs. 7.1, 7.2)

The secondary system is identical to that described in 5.2.2.

7.2.3 Containment (Figs. 7.6, 7.7, 7.8)

A rupture of the primary system does not flash any liquid into vapor phase nor is there any exothermic reaction between the coolant and fuel. Consequently, the containment proposed is not subject to any significant pressure requirements. It is, however, required to be relatively leak tight.

The superstructure is better than standard type construction as it must have sufficient integrity to conform with the requirement that the maximum permissible leakage rate from within the shell to the outside atmosphere be limited to less than 1% per psi overpressure per day.

The exterior surface of the basement walls is completely protected by continuous membrane waterproofing. A 2 ft. thick concrete slab covers the basement area. Access to the basement area is through an airlock passageway. The above provisions reduce the leakage rate to the requirement of 1% per psi per day.

The main purpose for having any degree of containment is to prevent any damage which may occur from the potential hazards accompanying the operation of an organic moderated reactor. Some of the hazards are coolant system rupture or leakage, coolant fire, and release of fission products.

A rupture of the coolant system would cause the coolant to be sprayed into the process room. The process rooms are large enough so that the pressure buildup does not exceed 1 psig. During this time there is a potential fire hazard from the hot coolant. A sprinkler system is provided to quench any fire and all electrical equipment is vapor proof to prevent the start of any fire. Should a coolant system break occur simultaneously with a fuel element rupture, the process rooms can contain the radioactivity. The wall thickness is sufficient to reduce the radiation level to a maximum permissible exposure level of 25 REM.⁽⁷⁾

7.3 Heat Analysis

7.3.1 Steady State

The reactor is sized for a thermal output of 40.6 Mwt. The heat analyses use the hot channel factors computed for the OMRE⁽⁴⁾ as a basis for the calculations. They indicate that the system has a required flow rate of 7.9×10^6 lb/hr and a temperature rise across the core of 29.2°F. The core heat transfer surface is finned to increase the heat transfer area. The fin efficiency is calculated to be approximately 88%. Heat transfer coefficients were based upon the formula presented by C. A. Trilling.⁽⁵⁾ Orificing the core to obtain even temperature rise across the core was not considered, but it should not be difficult to install orifices on the lower core support plate if found necessary.

The plant is designed as a load following system and the reactor power is varied by means of the control rods in accordance with the demands for steam.

7.3.2 Unsteady State

In the case of a complete loss of pumping power, the core decay heat is removed by means of natural circulation through the emergency cooler. In addition, a diesel-generator unit can be provided to maintain forced circulation. In addition some of the decay heat can be stored by allowing the temperature of the coolant in the reactor to rise to 600°F. This also creates a larger system temperature differential which promotes a higher natural circulation rate and increases capacity of the cooler.

7.3.3 Startup, Shutdown, and Control

The primary coolant system is preheated before start-up by the steam tracing system and the startup heater. Startup commences with the coolant temperature of 450°F with the coolant system pressurized, and with the coolant circulating at about 20% of rated flow. By manual control of the control rod drives, the reactor operator withdraws the control rods until the reactor is on a 25 second period. This period is maintained by manual control until 10% of full power is being generated. At this time, the plant is permitted to attain equilibrium temperatures. Above 10% and below 35% of rated load, the desired power is obtained by manual control of the coolant outlet temperature. At or above 35% of rated load and with steady state operation existing, automatic control is introduced, after which coolant flow and reactor power are automatically controlled to meet changes in demand load.

The shutdown is initiated by full insertion of the control rods with the rod drives. Following the controlled shutdown, the reactor decay heat and plant stored energy are dissipated through the decay heat removal system and/or the main steam header as determined by the reactor operator.

7.4 Physics

The physics analysis of the OMR for process steam production is based on the same ground rules used in the other studies. The critical enrichment is calculated as in the water moderated reactor analyses, as discussed in Appendix A. For a core employing 52 fuel elements an enrichment of 1.5% in U^{235} yielded a core with excess reactivity of 12.8% in the hot, clean case. This core is shown as lattice 9 in Table A-3.

To approximate the uniform burnup situation upon which this preliminary analysis is based, it is necessary to rearrange the fuel elements periodically according to a predetermined routine. If axial rearrangement of the fuel plates is not possible, the effective core life available is reduced by about 30% due to the higher burnups near the core center.

Atomics International has based some fuel cycles estimates for uranium-3.5% molybdenum plates on an average lifetime of 6600 Mwd/-M.T. of uranium. A 2.5 yr. core life amounts to about 6000 Mwd/M.T. with uniform burnup. However, such design involves peak burnups of 11,000 Mwd/M.T. With both the radial and axial rearrangement of fuel suggested, a 4 yr. fuel cycle can be specified. This amounts to an average burnup of about 9900 Mwd/M.T.

The various reactivity increments which must be controlled are itemized in Table 7.1. Two percent is allowed as a margin for shut-down. Normally, the auxiliary heating system keeps the diphenyl liquid at 250°F or higher. However, after irradiation it is possible for diphenyl to remain liquid at room temperature. For safety reasons the margin is increased to 3.2% to cover the additional 2.4% made available by cooling from 250°F to 80°F. Since the worth of the rods is less when the moderator is cool, it appears that the 9-rod array of cross-control rods is required.

It is conceivable, to ease the control requirements, that the initial core loading be less than 52 elements, and fresh fuel be added when fuel is rearranged. Perhaps a 44 to 48 element core with only about 8% excess reactivity available would be adequate for operation initially, building up to a full size core as fuel is burned up. There is no need to keep excess reactivity in the reactor and compensate for it with control rods throughout the cycle. Each loading should provide only the reactivity margin for burnup until the next scheduled shutdown. With this saving and use of some burnable poisons, a five control rod design might be feasible.

7.5 Capital Costs

The capital costs for the OMR system were not studied in the same detail as for the two water-cooled systems. Table 7.2 presents a rough cost estimate using the same form as Tables 5.3 and 6.3. All items in Table 7.2 are taken directly from Table 5.3 excepting for C-Equipment, Piping, etc. which are estimates derived from the OMR at Piqua.

Table 7.2

OMR CAPITAL COSTS

A-Land	Not Incl.
B-Structures	\$ 561,000
C-Equipment, piping, etc.	1,090,000
D-Electrical	112,000
E-Miscellaneous equipment	22,000
F-Personnel training program	Not Incl.
G-Startup supervision	Not Incl.
H-Contractors overhead and profit	Incl. Above
Subtotal (A through H)	\$ 1,785,000
I-Contingency - 10%	179,000
Subtotal	\$ 1,964,000
J-Top charges - 15%	296,000
K-Allowance for escalation	Not Incl.
Grand total	\$ 2,260,000

The most profitable area for cost improvement of the OMR system is that of the fuels. The unit fabrication cost of the fuel is relatively low based on the current range of fuels. However, the present metallic fuels have either burnup limitations due to radiation damage or require higher enrichment due to larger poison and dilution effects of alloying elements. Thus, the area of fuel development aimed at unit fabrication costs of the same range as the aluminum clad uranium fuels, at maintaining low enrichment and at low radiation damage effects, is of greatest potential benefit to the OMR. The realization of these benefits is quite probable within the current fuels development programs in the uranium oxide, carbide, or improved alloys field. A second region of potential economic benefit is in the development of better organic coolants. The term "better" is used in the sense of any feature or features of coolants which reduce their effect on cost of the reactor's product, i.e., steam or electricity. At the present time research is being performed on the coolant in order to develop an optimum coolant which, possibly through the addition of certain additives,⁽⁸⁾ will have better heat transfer characteristics, lower decomposition rates, and a lower melting point than the present coolants.

The OMR does not have materials problems other than the fuels and its containment situation is not so severe as with the water-cooled reactors. As a consequence, the economic improvements in the design area are expected to show a slight steady improvement which will result from operating experience. Major improvements which make a step change in the costs are not likely.

References - Section 7

1. "Piqua Organic Moderated Reactor System Description," NAA-SR-300.
2. "11,400 KW Nuclear Power Plant Employing an Organic Moderated Reactor," NAA-SR-2304.
3. U. S. Atomic Energy Commission, "Reactor Core Materials," Vol. 1, No. 2, May, 1958.
4. "Organic Moderated Reactor Experiment Safeguards Summary," NAA-SR-2323.
5. Trilling, C. A., "The OMRE - A Test of the Organic Moderator - Coolant Concept," Geneva Conference, A/Conf. 15/P/421, June, 1958.
6. Campbell, W. R., "Analysis of a New Method of Regulating Pressure in a Power Reactor," WAPD-TM-66.
7. "Preliminary Safeguards Report for the Piqua Organic Moderated Reactor," NAA-SR-3100.
8. "Organic Moderated Reactor Quarterly Progress Report," NAA-SR-2057, April-June, 1957.

FIG. 7.1
OMR HEAT BALANCE

FIG. 7.3
OMR REACTOR ELEVATION

FIG. 7.4
OMR FUEL ELEMENT PROTOTYPE

FIG. 7.5
OMR CORE ARRANGEMENT

FIG. 7.6
OMR BASEMENT PLAN

FIG. 7.7
OMR FIRST FLOOR PLAN

FIG. 7.8
CMR BUILDING ELEVATION

8. MATERIALS

8.1 General

The selection of materials for a given reactor complex is determined by an evaluation of many factors as well as the primary ground rules laid down for the design study. Major items of direct concern in materials selection are:

1. "State of the art" of the material with regard to knowledge of basic data, availability and fabrication know how.
2. System compatibility as corrosion, etc.
3. Service life.
4. Cost of the material.

The Process Steam Reactor study has many materials associated with the three reactor systems (PWR, BWR, and OMR) which were considered. Table 8.1 shows the major materials in the three proposals as presented for final selection. Advantages and disadvantages (Table 8.2) can be pointed out in each given area of material selection. However, a thorough economic analysis is needed to adequately determine the proper selection.

Table 8.1

MATERIALS FOR PROCESS STEAM REACTOR SYSTEMS

<u>Item</u>	<u>PWR</u>	<u>BWR</u>	<u>OMR</u>
Fuel	UO ₂	UO ₂	U-3.5 w/o Mo- 0.5 w/o Si
Clad	Zr-2	Zr-2	2S Aluminum
Coolant & Moderator	H ₂ O	H ₂ O	Diphenyl
Pressure Vessel	SA-212-B	SA-212-B 304 SS Clad	SA-212-B
Piping	SA-106-A	304 SS	SA-106-A
Heat Exchanger - Tubes	Monel	Monel	Carbon Steel
Vessel	Carbon Steel	Stainless Steel	Carbon Steel
Control Rods	2% Boron SS	2% Boron SS	2% Boron SS
Thermal Shield	1% Boron SS	-	SA-212-B

Table 8.2

MATERIALS - COMPARISONS

Fuel	UO ₂	U-3.5 w/o Mo-0.5 w/o Si
Advantages	Burnup > 10,000 Mwd/M.T.	High U Density
Disadvantages	Low U Density	Burnup 8,000 Mwd/M.T.
Clad	Zr-2	2S Aluminum
Advantages		Inexpensive
Disadvantages	Costly	
Coolant-Moderator	H ₂ O	Diphenyl
Advantages	Inexpensive	Non-corrosive
	Good Heat Transfer Prop.	Low Vapor Pressure
Disadvantages	Corrosive	Expensive
	High Vapor Pressure	Dissociates
		Poor Heat Transfer Prop.
Pressure Vessel & Piping	Clad Carbon Steel	Carbon Steel
Advantages	Low Corrosion Rate in H ₂ O	Non-corrosive to Diphenyl
Disadvantages	Expensive	Inexpensive
		Corrosive in H ₂ O
Heat Exchanger Tubes	Monel	Monel
Advantages	Standard Item	Standard Item, Low Cost
Disadvantages	Expensive	Expensive

The material selection is geared to a particular system and set of operating parameters, and thus, a cross check of advantages does not give a true picture. A material which may operate very satisfactorily in one system may be unsuited for operation in a different system.

The prime areas where major differences can be pointed out are in:

1. Fuel materials where burnup reflects itself in fuel costs.
2. Moderator materials and costs.
3. The costs associated with the materials and construction of the pressure vessel, piping and secondary boiler.

Pertinent data on fuel materials, clad materials, and structural materials are listed in Table 8.3 and discussed in following sections.

Table 8.3

MATERIALS CHARACTERISTICS

Item	Clad Material			Fuel Materials		Vessel Material
	Al. Alloys	Zr-2	SS(300 Series)	U Alloy	UO ₂ ⁽¹⁾	Carbon Steel
Neutron Section Microscopic	~.23 barns	~.18 barns	~3 barns	-	-	-
Thermal Conductivity Btu/(hr)(ft)(°F)	Good ~ 100	Fair ~ 10	Fair ~ 15	Fair ~ 18	Poor (2,3) ~1 (2) Decreases 10 fold at 10 ¹⁸ nvt at 650°F	Fair ~ 15
Corrosion						
Water	(4) 1.3 mdd 550°F 5 ppm PO ₄ add.	(5) 0.18 mdd at 680°F (9,10) Excellent resist. 2 yr.	~0.1 mdd at 600°F	(6) U-5 Zr-1.5 Nb 3180 mdd at 550°F (11) U-10.5 Mo 56-176 da. at 650°F	(7) Inert in 680°F	(8) 5 mdd at 600°F 100 cc H ₂ /1. LiOH ion exchanger
Organic	Negligible at 575°F	(13) 0.66 mdd at 750°F	(12) 0.33 mdd at 575°F	U-3.5 Mo 3.3 mdd at 575°F		(12) 1.65 mdd at 575°F
Explosive or Violent Chem. Reac. with Mod.						
Water	No	Possible	No	Possible	No	-
Organic	No	No	No	No	No	-
Irrad. Behavior	Good	Good	Good	(14) U-10.5 Mo to 31,000 Mwd/M.T. at 212°F	(15,16,17,18) to 33,000 Mwd/M.T.	(19) ASTM A-201 ductile brittle transition temp. increases
Mechanical Strength	Good to 500°F	Excellent to 600°F	Excellent to 600°F	Good to moderate temp. 900°F	Poor Thermal Fractures	Good to 600°F

NOTE: Numbers in parenthesis refer to references at end of text.

8.2 Fuel

8.2.1 Fissile Material

The fuel for the PWR and BWR is selected on the basis of the following criteria:

- a. That the fuel withstand high burnup $\bar{>}$ 15,000 Mwd/M.T.
- b. That the fuel be at an advanced "state of the art" presently.
- c. That it be relatively inexpensive or has a reasonable chance of future cost reductions based on other fabrication methods.
- d. That it be stable upon exposure to the moderator-coolant, i.e., a moderate amount of ruptured or defected fuel rods be tolerated with no resulting system contamination.

To fulfill the above and other requirements, UO_2 is selected as the fuel material for the water-cooled reactors. The substantiation of the above selection is based on data from many sources as indicated in Table 8.3. It appears presently that in terms of fuel burnup UO_2 has an advantage over that of low alloy uranium fuels. Data from WAPD UO_2 irradiations indicate good stability at irradiation levels of $\bar{>}$ 12,000 Mwd/t. UO_2 also has the added advantage of being essentially inert to water reactions in the event of clad ruptures thus permitting operating of the reactor with a few fuel clad defects. Tests in EBWR and Borax IV on clad $\text{ThO}_2\text{-UO}_2$ fuels and elsewhere on clad UO_2 fuels indicate that no prolonged residual contamination of the system exists when fuel is exposed to the moderator due to clad defects, and only gaseous activity evidences itself in the reactor. (20,21)

It would appear at first that the logical choice for a low power density, long life core is a metal alloy fuel since it has high uranium density which, coupled with the close packing, i.e., high metal to water ratio, affords a high conversion ratio. The fabrication of uranium alloy fuel plates is also at an advanced state of the art. However, the selection of metallic fuel induces penalties in the operation of a water-cooled reactor system. Uranium and its alloys are severely attacked by water at low-moderate temperatures (6,22,23) and in the event of clad ruptures considerable contamination of the entire system is possible. Alloying additives and proper heat treatment reduce the corrosion rate somewhat but the heat treatment for corrosion stability is not compatible with the irradiation requirements of the system. (24) This imposes a severe and stringent cladding requirement on the alloy fuel. The inability of the uranium metal fuel to reach 12,000 Mwd/M.T. without the addition of a considerable amount of high neutron absorbing material, which must be compensated for by increasing the initial fuel enrichment, is a second penalty of the metallic fuel system.

In summary, the use of a metal alloy fuel in a water moderated and cooled reactor system is limited by four factors.

- a. Low relative burnup levels attainable in low alloy addition materials.
- b. High alloy additions of high cross section materials required to stabilize radiation growth and attain long irradiation times commensurate with high burnup levels.
- c. Stringent cladding requirements to protect the fuel from corrosion.
- d. Poor corrosion resistance of the fuel which does not permit operation of the system with a fuel clad failure.

In the case of the OMR system, the use of a metal alloy is possible and, based on Atomics International data, desirable except for the relatively high neutron absorption in the alloying ingredients required for radiation stability. A maximum guaranteed burnup level for U-3.5 Mo is 11,000 Mwd/M.T. with the average at 6,600 Mwd/M.T.⁽²⁵⁾ It is hoped that the alloy U-3.5 w/o Mo - 0.5 w/o Si may attain average burnup levels of 11-13,000 Mwd/M.T.⁽²⁶⁾ Should this goal be achieved, the economic competition of the alloy fuel with UO_2 could become very great. The state of the art of the aluminum clad uranium alloy plate fabrication is sufficiently advanced to cause no development or economic barrier. Since the U-Mo alloy does not corrode in the organic moderator, the Al cladding serves only for fission product retention and, it is postulated that slight clad failures could be tolerated in the OMR system with only moderate activity increase in the system due to fission gas release.

8.2.2 Clad Materials

Selection of the clad material for the PWR and BWR is based on the following items:

1. Nuclear cross section
2. Corrosion resistance
3. Material required for free standing cladding and its ratio to maximum fuel content.

The calculations for free standing cladding of oxide pellets were made on the basis of a vessel hydrostatic of about 1.5 times design pressure at room temperature and the pressure encountered at operating conditions. Both elastic stability and collapse conditions are calculated and in all cases, the collapse thickness requirements are controlling.

Calculations indicate the amounts of clad thickness for a clad temperature of 500°F at operating pressure conditions of 400 psi for 0.5 in. diameter rods as noted in Table 8.4.

Table 8.4

CLADDING METAL THICKNESSES

<u>Material</u>	<u>Clad Thickness</u>	<u>Clad Thickness with Corrosion Allowance</u>
Al-X-8001	0.030 in.	0.050 in.
304 SS	0.013 in.	0.015 in.
Zr-2	0.015 in.	0.018 in.

The above clad thickness values do not appear out of line. However, the higher corrosion allowance for aluminum cladding can displace significant quantities of fuel. As fuel elements become smaller and time in the reactor becomes larger, the factor grows in importance. For example, assuming 0.5 in. fuel rods are used, the cladding metal thickness listed in Table 8.4 yields about 25% less fuel in a rod when aluminum clad is used than when zirconium clad is used. The corrosion aspects of the aluminum alloys are discussed later in this section. The selection of Zr-2 as the cladding material is based on its low cross section and good corrosion properties. Furthermore, major cost reductions should be expected in Zr-2 clad uranium oxide fuels.

The use of thinner clad with lead bonding between the clad and pellets was considered but not utilized for various reasons. Thinner clad thickness with bonding may lead to thermal ratcheting and to possible increase in the number of cladding failures. Lead bonding, while acceptable and possible, is not an inexpensive process.

In the OMR system, aluminum is the obvious choice since it has desirable properties of low cost and low cross section and negligible corrosion in the organic medium.

8.2.3 Bonding Material

In both the PWR and BWR fuels, the fuel-clad gap is initially filled with helium. This serves a useful function in that it allows testing of the final seal welds on the fuel rod by helium leak tests.

The OMR fuel system tends to the ideal in bonding method, i.e., the metallurgical bond. The improved heat release properties resulting from better conduction in metal over oxide fuel and metallurgical bond over a gas gap are more important to the OMR since the organic coolant has relatively poor heat transfer properties.

8.3 Structural Materials

8.3.1 Steels

The design and construction of the Process Steam Reactor pressure vessel and system would be in accordance with the ASME Boiler Code requirements. Section VIII of the Code covers the construction requirements of unfired pressure vessels and covers corrosion allowances and clad vessel construction.

The prime moving factor in the evaluation of vessel material for the water reactors is the cost factors between the use of a plain carbon steel vessel and piping versus that associated with a stainless steel clad vessel and stainless steel piping system. In the case of a stainless clad vessel, no credit is taken, in terms of added strengths for the clad material.

Table 8.5 lists typical values in terms of strength for carbon steels contemplated as being usable. The list includes pipe, plate and forging quality steels. No high strength steels are included in the list at this time for several reasons, the prime one being more critical field welding conditions which reflect in construction costs. Data on ductile-brittle transition temperatures indicate an increase in transition temperature as alloy additions increase. Also less data are available on irradiation and corrosion effects.

Table 8.5

CARBON STEEL SPECIFICATIONS⁽²⁷⁾

<u>Material and Specifica- tion No.</u>	<u>Composition</u>	<u>Grade</u>	<u>Specified Min. Tensile psi</u>	<u>Stress Level in psi per Metal Temp. -20 to 650°F</u>
SA-212	C-Mn-Si	A	65,000	16,250
SA-212	C-Mn-Si	B	70,000	17,500
SA-106	C-Mn-Si	B	60,000	15,000
SA-285	C-Mn	C	55,000	13,750
SA-201	C-Mn-Si	B	60,000	15,000
SA-105	C-Mn	I	60,000	15,000
SA-181	C-Mn-Si	I	60,000	15,000

Survey of the table indicates that from the strength standpoint, SA-212-B is the probable choice; though on low pressure systems, the SA-285-C may be desirable.

The major factor in the allowable usage of carbon steels in reactor vessel and internals is the corrosion rate of these materials. Considerable work has been done along these lines by WAPD, B&W, and KAPL.

The scope of the data and conditions in reference reports^(8,28,29,30,32) make it difficult to correlate results between the groups and tests. However, it appears that, with proper water treatment and processing conditions, it is possible to use carbon steel in the system without a severe penalty. According to Blaser and Owens, tests indicate that "the corrosion rate decreases rapidly with time of exposure. After 6 months, indicated corrosion rates at 600°F are below 20 mg/dm²/mo (.67 mdd) or about 0.0001 in. per year. Specimens in semistatic water retain more scale and have a lower rate of total corrosion and loss of iron to the water than the specimens at high velocity." Vaughan and Ferguson⁽⁸⁾ state "carbon steels showed more corrosion than SS in water at temperatures of 600°F. This corrosion was not sufficiently high to prevent the use of carbon steel for reactor applications." A good summary of carbon steel corrosion data appears in WAPD-LSR(c)-134, Review of Carbon Steel Corrosion Data in High Temperature High Purity Water in Dynamic Systems, by Tackett, Brown and Esper, 1955. This is a compilation of B&W, WAPD, and KAPL testing results.

The water treatment having hydrogen overpressure to achieve a concentration of 100 to 500 cc/l coupled with LiOH additive and ion exchanger to maintain a pH of ~11 can give a system capable of operating effectively with a corrosion rate of ~1.5 mdd or ~0.27 mil/yr uniform penetration. Since most of the corrosion data from the reference documents are for temperatures of 600°F and flow velocities of ~30 ft/sec, it appears that in the process steam system, the corrosion rate should be less.

The use of carbon steel system, with its ease of fabrication and resulting cost savings, is intriguing to designers. The process steam system has the ingredients necessary to further the cause of carbon steel, i.e., (1) low temperature <450°F, (2) a completely enclosed primary loop and (3) an immediate incentive to a low (capital) cost installation in contrast to a development prototype merely to prove operability.

Though much information may be gathered and written and many loop tests run, until an actual carbon steel system is built and operated, all that we will have is an opportunity to extrapolate loop test data to a full scale system.

The process steam system also offers the opportunity to use carbon steel in both the PWR and BWR system because of the closed primary loop. In terms of base material costs, the following base price (plus extras) figures are given,

Table 8.6

BASE PRICE OF STEELS

<u>Item</u>	<u>Base Price ¢/# FOB Mill⁽³³⁾</u>
Carbon Steel Plate	5.30
Hi-Strength Low Alloy Plate	7.95
304 SS Clad (10%) on Carbon Plate	28.80
304 SS Plate	45.70

Not only are there initial material cost savings, but also a savings in field erection costs. Coupled with this is the economic factor of a fast write off on commercial chemical installations which may cover the uncertainty factor of the long time (>10 yr.) serviceability of the plant system when carbon steel is used.

8.3.2 Monel

The use of Monel tubes for the boiler appears quite acceptable for use in the type of water environment best for low corrosion steels. Most data indicate^(8,28,29,30,32) lower corrosion rates of Monel than of stainless steel.

8.3.3 Aluminum

A comprehensive survey of aluminum for use in reactors is presented in Appendix B.

The use of aluminum as fuel cladding metal was considered in the early design efforts of PWR and BWR. It was concluded that the operating temperatures of the reactors are at a level where aluminum corrosion rates can be excessively high for the desired long term operation.

It appears that there are probably operating conditions and temperatures as high as are required for the two water-cooled reactor systems which will result in low corrosion of aluminum. However, the corrosion rate appears to be sensitive to more factors than can be accounted for by the normal factors of pH, water purity, flow velocity, and temperature.

For example, there are variable corrosion rates which may be attributable to variable ratio of aluminum surface to water volume. Apparently, an aluminum sample put into a dynamic test loop which has previously operated with identical samples will corrode at a different rate. The effects of exposure time and of irradiation are not completely established. The magnitude of these little understood factors is too great to recommend aluminum for use in water at 450°F.

Aluminum is incompatible with carbon or low alloy steels in an aqueous medium where operating conditions require pH adjustment for either the steel or the aluminum. The optimum condition for aluminum requires water slightly on the acidic side of neutral. That for carbon or low alloy steels is appreciably on the basic side. The structural steels which can tolerate the presence of aluminum in an aqueous medium are the 300 series stainless steels. However, even with the stainless steels, there is evidence that the slightly acidic water needed for the aluminum may cause pitting of the stainless.

Aluminum is a completely acceptable material from a corrosion standpoint in the organic moderated reactor. Its choice in that reactor system is limited to the strength, temperature and economic considerations.

8.4 Materials - Future

8.4.1 Fuel

The clad UO_2 rods are an acceptable fuel type for water-cooled reactors. The radiation damage and low corrosion rate of the UO_2 are good. The most serious drawback, at present, is the cost of fabrication. Two methods of fabrication which appear to offer potential cost savings are that of swaging⁽³⁴⁾ UO_2 in stainless steel or Zr-2 clad, and extrusion of UO_2 fuel rods.⁽³⁵⁾ Both of these methods should prove to be more economical than fabrication of pellet type rods. Irradiation⁽³⁶⁾ testing of fabricated prototypes is in progress.

With regard to future fuel potential in the OMR system, the use of UO_2 fuel and sintered aluminum product (SAP) should not be excluded from consideration in the immediate future. Though there are several disadvantages to the use of oxides, especially their poor thermal properties coupled with the poor thermal properties of diphenyl, proper design may circumvent these obstacles and allow a higher operating fuel element and coolant temperature. This reactor system should gain most from successful developments of the uranium carbide fuels. This type fuel represents a material having higher uranium density than the oxide, possesses better thermal conductivity, is compatible with the organic

fluid in case of clad failure and has potentialities of high burnup. The major drawback is the carbide reaction with water resulting from in-leakage from the process steam cycle.

8.4.2 Structural Materials

Structural materials problems are of little consequence to the OMR system. In the water-cooled systems, this may also be true, but is unproven. The major development problem in the water-cooled systems is in the area of water quality control. At present, there is insufficient effort applied to understand the details of operation and collection of data of the water quality control systems on existing reactors. In the case of carbon steels, there appears to be a reluctance to try it on a reactor system just because stainless steel works so well.

Aluminum as fuel cladding and system structural material requires more basic understanding of the factors contributing to corrosion rate. It has excellent potential as fuel cladding metal for a few months operation in water in the 400 - 500°F range. As a system structural material (reactor vessel, pipes, valves, etc.) it may have good potentialities for lower pressure process steam systems. This is a situation which may be answered by an economic study of an all aluminum versus an all carbon or stainless steel reactor system.

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9. ECONOMICS

9.1 General

The section on economics is presented because it illustrates the costs achievable and the influence of the ground rules applied to a study. The latter point is particularly important for it readily seen that a change of ground rule can have major influence on the choice of core loading, the reactor system design and probably, in some cases, the type of reactor.

Table 9.1 lists the range of steam costs from each of the designs presented in this report. The details of the factors making up the costs are presented in subsequent portions of the section. The cost range results largely from the range of ground rules applied.

Table 9.1

STEAM COSTS - ¢/1000 LBS STEAM

	<u>Capital</u>	<u>Operating</u>		<u>Fuel</u>	<u>Total</u>
		<u>Non-Fuel</u>	<u>Fuel</u>	<u>Inventory</u>	
PWR - Stainless Steel	22-90	15-20	20-26	9-14	66-150
- Carbon Steel	20-81	15-20	20-26	9-14	64-141
BWR	18-73	15-20	28-39	22-40	83-172
OMR	18-74	21-26	24-32	12-22	75-154

9.2 Capital Charges

The capital charge as applied to process steam plants varies over a wider range than does a similar charge for private utilities. In discussions with manufacturing and engineering groups, it appears that industries interested in nuclear process steam applications would apply annual capital charges varying from about 8 to 25% of the initial capital investment. The minimum charge represents about 5% for depreciation and 3% for insurance and taxes. The maximum represents items such as 10% depreciation, 6% for insurance and taxes, and 9% for the cost of money and general overhead.

A second factor in the influence of the capital cost on the price of steam is the plant factor. This is defined as the total amount of steam a reactor could produce running at capacity 100% of the time divided into the amount of steam actually produced. Reasonable plant factors for the type operations assumed in this report are 70 to 90%.

Table 9.2

CAPITAL INFLUENCE ON STEAM COSTS
¢/1000 LBS OF STEAM

Annual Capital Chgs. Plant Factor	8%		15%		25%		Total Plant Capital Cost
	90%	70%	90%	70%	90%	80%	
PWR - Stainless Steel	22	29	42	54	70	90	\$ 2,763,000
- Carbon Steel	20	26	38	49	63	81	2,486,000
BWR	18	23	34	44	56	73	2,226,000
OMR	18	24	34	44	57	74	2,260,000

9.3 Operating Costs

9.3.1. Costs Other Than Fuel

The operating costs are assumed to be equal for all three of the reactor types studied. The single exception is the added operating cost which the OMR has for the organic coolant makeup. The annual operating costs used are as follows:

Labor - direct	\$ 70,000
- indirect	15,000
Supplies and maintenance	40,000
Manufacturing Expenses	<u>25,000</u>
	\$ 150,000

This annual operating cost amounts to 15¢/1000 lb of steam at a 90% plant factor and 20¢/1000 lb of steam at a 70% plant factor. These are costs applicable to the PWR and BWR.

The OMR, with an organic coolant breakdown of 1.1 lbs/Mw hr and a coolant cost of 17¢/lb has an added operating cost of 6¢/1000 lbs of steam. The total OMR operating cost is 21¢/1000 lbs of steam at 90% plant factor and 26¢/1000 lbs of steam at 70% factor.

9.3.2. Fuel

Table 9.3 lists the influence of the equilibrium fuel feed rate on the cost of steam. These costs are arrived at using the unit costs listed in Section 9.5. The factors included are price of uranium as UF₆, conversion of UF₆ to oxide or metal, processing charge including conversion of the uranium and plutonium nitrates to UF₆ and plutonium metal and finally credit is taken for the residual values of uranium and plutonium in the spent fuel.

The fuel operating charge depends only on the unit burn-up of fuel (Mwd/M.T.). It has no relationship to plant factor. The fuels used for the PWR and BWR reactors are zirconium clad. If aluminum clad flues were used, based on the unit costs listed in 9.5, the steam costs would be about 5¢ less/1000 lb for the PWR and about 7¢ for the BWR.

Table 9.3

FUEL OPERATING INFLUENCE ON STEAM COSTS

	<u>Axial and Radial Fuel Reshuffle</u>		<u>Radial Only Fuel Reshuffle</u>	
	<u>Mwd/M.T.</u>	<u>¢/1000 lbs</u>	<u>Mwd/M.T.</u>	<u>¢/1000 lbs</u>
PWR	17,200	20	12,700	26
BWR	12,300	28	9,100	39
OMR	8,900	24	6,600	32

9.4 Inventory

The quantities of fuel in inventory are based on assumptions made at an equilibrium fuel situation with refueling operations taking place on six month intervals. It is assumed that in addition to the fuel in the reactor, there is the equivalent of one month's supply of fuel in fabrication, six months' supply of new fabricated fuel, six months' supply of partially spent fuel waiting to be reloaded, three months' supply of fully spent fuel in decay storage, and one month's supply of spent fuel in shipping and process. The fuel in the core loading is the predominant quantity. Table 9.4 gives the quantities of uranium in inventory under various conditions.

Table 9.4

URANIUM INVENTORIES - Kg

	<u>Plant Factor, %</u>	<u>Core Loading Only - Kg</u>	<u>Total Inventory - Kg</u>
PWR	90	3,380	4,400
	70	3,380	4,200
BWR	90	10,300	11,800
	70	10,300	11,400
OMR	90	7,650	9,700
	70	7,650	9,200

For purpose of determining an inventory charge, it is assumed that the total quantity of uranium has a 4% rental charge based on its original value in UF₆ form. A 6% cost of money charge is placed on the

fabrication cost of all fuel ahead of the reactor. Finally, the fabrication value of the quantity of fuel in the reactor is treated as a capital item and charged at either of the two lower charges, 8% and 15%, given in Section 9.2. There is a slight difference between the fuel situations of axial plus radial and the radial only fuel reshuffling. However, the differences are small in this particular instance due to the quantity of fuel in the core being such a dominant inventory factor.

As with the fuel operating cost, if aluminum fuels were used in the PWR and BWR reactors, the inventory costs would be less by about 2 and 4¢ respectively/1000 lbs of steam. Table 9.5 gives the inventory charges.

Table 9.5

FUEL INVENTORY INFLUENCE ON STEAM COSTS

		Steam Cost - ¢/1000 lb	
	Plant Factor, %	8% Core Charge	15% Core Charge
PWR	90	9	12
	70	11	14
BWR	90	22	32
	70	28	40
OMR	90	12	17
	70	15	22

9.5 Unit Costs and Charges

The following unit costs are used:

Uranium as UF ₆	AEC Price Schedule, \$
UF ₆ → UO ₂ at 1.5% enrichment	14/Kg U
at 1.8% enrichment	16/Kg U
UF ₆ → U at 1.5% enrichment	12/Kg U
Fuel Fabrication Costs (all items from UO ₂ or metal to finished subassembly, including scrap recovery and shipping)	
Zr clad UO ₂	140/Kg U
Al clad UO ₂	85/Kg U
Al clad U-Mo alloy	50/Kg U
Process Charge - all fuel types	30/Kg U
UNH → UF ₆	5.60/Kg U
PuNH → Pu	1.50/gm Pu
Pu Metal	12/gm

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Westinghouse Electric Company

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Appendix A

SUMMARY OF PHYSICS CALCULATIONS1. Theory

The neutrons diffuse according to 2-1/2 group theory. That is, fast migration occurs without capture but with removal to the resonance or epithermal group. In the latter no migration occurs; only capture or removal to the thermal group. In the thermal group migration and capture occur in the usual way. The energy cut between the thermal and resonance band has been taken to be 0.4 ev in this work. All resonance capture occurs in the resonance band along with epithermal $1/v$ capture and fission. Fission neutrons are born only in the fast group and fast fission is viewed as an internal process which yields an incremental bonus, ϵ , to ν , the number of neutrons released per fission. The neutron diffusion equations are then:

$$\begin{aligned} \text{i)} \quad & -D_f \nabla^2 \phi_f + \Sigma_{Rf} \phi_f = \epsilon \nu_{th} \Sigma_{Fth} \phi_{th} + \epsilon \nu_r \Sigma_{Fr} \phi_r \\ \text{ii)} \quad & (\Sigma_{Ar} + \Sigma_{Rr}) \phi_r = \Sigma_{Rf} \phi_f \\ \text{iii)} \quad & -D_{th} \nabla^2 \phi_{th} + \Sigma_{Ath} \phi_{th} = \Sigma_{Rr} \phi_r \end{aligned}$$

The subscripts A, F, R refer to absorption, fission, and removal processes, while f, r, th refer to fast, resonance and thermal groups. Equation ii can be solved for ϕ_r and the result substituted into equation iii yielding:

$$\text{iii')} \quad -D_{th} \nabla^2 \phi_{th} + \Sigma_{Ath} \phi_{th} = \Sigma_{Rf} \frac{\Sigma_{Rr}}{\Sigma_{Ar} + \Sigma_{Rr}} \phi_f$$

If equations i and iii' are to correspond to two-group theory and age theory in the appropriate limits we must have

$$\text{iv)} \quad \frac{\Sigma_{Rr}}{\Sigma_{Ar} + \Sigma_{Rr}} = P = \left(e^{-\int_4^{E_0} \Sigma_A(dE/E)/\xi \Sigma_s} \right) \left(e^{-\frac{N^{28} R I^{28}}{\xi \Sigma_s}} \right)$$

The last factor in iv is the familiar probability of escape from resonance capture in U^{238} and is denoted by P_{28} . The first factor is the probability of escape from capture in the resonance band by $1/v$ type processes. This factor is denoted by $P_{1/v}$.

$$v) \quad P_1/v = e^{-1/2 \Sigma A(.025)/\xi \Sigma s}$$

For a system with buckling B^2 the critical equation is:

$$vi) \quad (1 + L^2 B^2) (1 + \tau B^2) = P \epsilon K_{th} + \epsilon K_r (1 - P) (1 + L^2 B^2)$$

$$\tau = D_f / \Sigma R_f; \quad K_{th} = \nu_{th} \Sigma F_{th} / \Sigma A_{th}, \quad K_r = \nu_r \frac{\Sigma F_r}{\Sigma A_r}$$

$$P = P_1/v \quad P_{28}.$$

Introducing $\alpha + n = \Sigma_{Ath}^{25} / \Sigma_{Fth}^{25}$, $f_{th} = \Sigma_{Ath}^{25} / \Sigma_{Ath}$, etc., and assuming $\nu_r = \nu_{th}$ we can solve for the system eigenvalue, ν :

$$vii) \quad \nu = \frac{(1 + L^2 B^2)(1 + \tau B^2)}{\epsilon \left[\frac{P f_{th}}{1 + \alpha_{th}} + \frac{f_r (1 - P)(1 + L^2 B^2)}{1 + \alpha_r} \right]}$$

The reactivity, ρ , is given by

$$viii) \quad \rho = \frac{2.48 - \nu}{2.48}$$

The number of resonance absorptions per thermal absorption is given by R which can be found from:

$$ix) \quad R = \frac{\eta_{th}}{\frac{(1 + L^2 B^2) \Sigma A_r}{(1 - P) \epsilon \Sigma_{Ar}^{25}} - \eta_r}$$

R' , the ratio of epithermal fissions to thermal fissions, is given by

$$x) \quad R' = R \frac{\eta_r}{\eta_{th}}$$

2. Isotope Production

The Argonne analogue computer facility was used to compute Pu isotope buildup. The equations used were:

$$xi) \quad \frac{d N^{25}}{d t} = -\phi(t) \sigma_{Ath}^{25} N^{25} (1 + R)$$

$$\text{xii) } \frac{d N^{28}}{d t} = -\phi(t) \left\{ \sigma_{\text{Ath}}^{28} + \mu_1 \left[\sigma_{\text{Ath}}^{25} N^{25} \eta_{\text{th}}^{25} (1 + R') + \sigma_{\text{Ath}}^{49} N^{49} \eta_{\text{th}}^{49} + \sigma_{\text{Ath}}^{41} N^{41} \eta_{\text{th}}^{41} \right] + \mu_2 \left[(1 + R') \sigma_A^{25} N^{25} \right] \right\}$$

$$\text{xiii) } \frac{d N^{49}}{d t} = - \frac{d N^{28}}{d t} - \phi(t) \sigma_A^{49} N^{49}$$

$$\text{xiv) } \frac{d N^{40}}{d t} = \phi(t) \left[(\sigma_A^{49} - \sigma_F^{49}) N^{49} - \sigma_A^{40} N^{40} \right]$$

$$\text{xv) } \frac{d N^{41}}{d t} = \phi(t) (\sigma_A^{40} N^{40} - \sigma_A^{41} N^{41})$$

$$\text{xvi) } P (\text{power}) = \left[N^{25} \sigma_F^{25} + N^{49} \sigma_F^{49} + N^{41} \sigma_F^{41} \right] \phi(t)$$

The superscripts on the isotopic densities, N^i , refer to U^{235} (25), U^{238} (28), Pu^{239} (49), Pu^{240} (40), and Pu^{241} (41). The flux is determined by the power equation (xvi). For constant power operation the reactivity (ρ) of the system is computed (vii and viii) as a function of time.

3. Nuclear Parameters

a. Cross Sections

Thermal cross sections are averaged over a Maxwellian Spectrum which is characterized by the temperature of the moderator. Basic cross section data were taken from BNL-325⁽¹⁾ and ANL-5800.⁽²⁾ Where necessary, disadvantage factors were calculated on an IBM-704 using the I-2 Code.⁽³⁾ The cross section of Pu^{240} is adjusted by the method of Weil to give correct results over the cycle.

Table A-1

EFFECTIVE THERMAL CROSS SECTIONS

	σ_a (barns)	σ_c (barns)	σ_s (barns)	σ_f (barns)	η	ν
U^{235}	423	65	10	358	2.10	2.48
U^{238}	1.75	1.75	8.75	0	0	0
Pu^{239}	1103	370	10	733	1.912	2.88
Pu^{240}	1300	1300	10	0	0	0
Pu^{241}	1360	400	10	960	2.06	2.91
F.P.	50	50	10	0	0	0
H_2O	.427	.427	70	0	0	0
Al	.149	.149	1.4	0	0	0
S.S.	1.64	1.64	11	0	0	0
Zr	.16	.16	8	0	0	0

b. Resonance Integrals

The formula of Hellstrand as given in ANL-5800,⁽²⁾ page 344

$$RI = \left[4.15 + 26.6 \left(\frac{S}{M} \right)^{1/2} \right] \text{ in barns } \text{UO}_2$$

$$RI = \left[2.81 + 24.7 \left(\frac{S}{M} \right)^{1/2} \right] \text{ in barns } \text{U}$$

together with Dancoff-Ginsburg correction factors were used to find the resonance integrals (RI) of U^{238} .

c. Fast Effect and Fast Conversion Ratio

These quantities were calculated by the methods outlined in ANL-5800,⁽²⁾ pages 127 and 355. The cross sections used were taken from ANL-5800 and are given in Table A-2.

Table A-2

FAST EFFECT CROSS SECTIONS

	σ_f	σ_c	σ_{in}	ν
U^{238}	.53	.04	2.1	2.65
U^{235}	1.3	.1	1.4	2.65
H_2O	0	.024	2.81	0
Mg	0	0	.05	0
O	0	.024	.171	0
Al	0	.0003	.38	0
Zr	0	0	.56	0

$$x = .575$$

d. Fast Migration and Diffusion

The age and fast diffusion coefficients were calculated by the method of equivalence factors outlined in ANL-5800,⁽²⁾ page 165. Uranium was taken to act exactly like H_2O on a volume per volume basis while for the oxygen in UO_2 the equivalence factors are:

group	α	β
fast	.04009	.51248
epithermal	.01199	.22191

e. Generalized Initial Conversion Ratio

We wish to compute the conversion ratio where there is an appreciable amount of epithermal fission and capture. The conversion ratio is the ratio of the number of fertile (e.g., U^{238}) atoms destroyed by neutron capture to the number of fissionable atoms (e.g., U^{235}) destroyed by capture.

The number of U^{235} atoms destroyed by neutron capture is: $\Sigma_{Ath}^{235} \phi_{th} + \Sigma_{Ar}^{235} \phi_r$ where the subscript r refers to a resonance region or epithermal flux. We will for the moment neglect processes of fast capture and fission, since these can be accounted for separately, provided that we use a proper definition of the fast fission factor, ϵ .

The thermal conversion ratio is then:

$$T.C.R. = \frac{\Sigma_A^{238} \phi_{th}}{\Sigma_{Ath} \phi_{th} + \Sigma_{Ar} \phi_r} = \frac{\Sigma_A^{238}}{\Sigma_{Ath}^{238} (1 + R)}$$

$$\text{where } R = \Sigma_{Ar}^{235} \phi_r / \Sigma_{Ath}^{235} \phi_{th}.$$

The resonance conversion ratio is the result of capture by fertile atoms in the resonance region of a fraction of those resonance energy neutrons which haven't leaked out of the system. The fraction involved is clearly $1 - P_{28}$. The number of fast neutrons born (per second) is $\eta_{th} \Sigma_A^{235} \phi_{th} + \eta_r \Sigma_{Ar}^{235} \phi_r = \eta_{th} \Sigma_{Ath}^{235} \phi_{th} (1 + R \eta_r / \eta_{th})$.

As a result of the process of fast leakage accompanied by fast fission and fast capture,

$$\frac{\epsilon \eta_{th} \Sigma_A^{235} \phi_{th} (1 + R \eta_r / \eta_{th})}{1 + \tau B^2}$$

neutrons are injected into the moderator in the resonance range. (This assumes all fast migration takes place at energies above resonance capture, all thermal migration at lower energies.) The resonance conversion ratio is, then,

$$R.C.R. = \frac{\epsilon \eta_{th} (1 - P_{28})}{1 + \tau B^2} \frac{[1 + R (\eta_r / \eta_{th})]}{(1 + R)}$$

Since $\eta_r < \eta_{th}$, R. C. R. is less than if there were only thermal fission.

The fast conversion ratio, F. C. R., is arrived at as follows:
 $\Sigma_{Ath}^{235} \phi_{th} (1 + R)$ absorptions yield

$$\frac{\Sigma_{Ath}^{235} \phi_{th}}{1 + \alpha_{th}} \left(1 + R \frac{(1 + \alpha_{th})}{(1 + \alpha_r)} \right)$$

fissions which in turn yield

$$F \frac{\Sigma_{Ath} \phi_{th}}{1 + \alpha_{th}} \left(1 + R \frac{1 + \alpha_{th}}{1 + \alpha_r} \right)$$

fast fissions, or

$$F(\nu_f - 1) \left[\frac{\Sigma_{Ath} \phi_{th}}{1 + \alpha_{th}} \left(1 + R \frac{1 + \alpha_{th}}{1 + \alpha_r} \right) \right]$$

extra neutrons.

$$\nu_{th} \frac{\Sigma_{Ath}^{235} \phi_{th}}{1 + \alpha_{th}} \left(1 + R \frac{\eta_r}{\eta_{th}} \right) + F(\nu_f - 1) \frac{\Sigma_A^{235} \phi_{th}}{1 + \alpha_{th}} \left(1 + R \frac{1 + \alpha_{th}}{1 + \alpha_r} \right)$$

is the total neutron production, and

$$\epsilon \nu \frac{\Sigma_{Ath}^{235} \phi_{th}}{1 + \alpha_{th}} \left(1 + R \frac{\eta_r}{\eta_{th}} \right)$$

re-enter the chain. At this point it is obvious that if there is to be any justice, $\nu_r = \nu_{th}$, for in that case

$$F. C. R. = \frac{\nu_{th} (1 - \epsilon) \left(1 + R \frac{\eta_r}{\eta_{th}} \right) - F(\nu_f - 1) \left(1 + R \frac{\eta_r}{\eta_{th}} \right)}{(1 + \alpha_{th}) (1 + R)}$$

$$= \left[\frac{\nu_{th} (1 - \epsilon) - F(\nu_f - 1)}{1 + \alpha_{th}} \right] \frac{1 + R \frac{\eta_r}{\eta_{th}}}{1 + R}$$

The term in brackets is, to the initiated, just the F. C. R. calculated for only thermal fissions.

The generalized conversion ratio is then:

$$\begin{aligned} \text{G. I. C. R.} = & \frac{\Sigma_{\text{Ath}}^{238} / \Sigma_{\text{Ath}}^{235}}{(1 + R)} + \frac{\epsilon \eta_{\text{th}} (1 - P_{28})}{1 + \tau B^2} \frac{1 + R \left(\frac{\eta_r}{\eta_{\text{th}}} \right)}{1 + R} \\ & + \left[\frac{\nu_{\text{th}} (1 - \epsilon) - F(\nu_f - 1)}{1 + \alpha_{\text{th}}} \right] \frac{1 + R \frac{\eta_r}{\eta_{\text{th}}}}{1 + R} \end{aligned}$$

In the two and one-half group formulation one has

$$(\Sigma_{\text{Ar}} + \Sigma_{\text{Rr}}) \phi_r = \frac{\epsilon \eta_{\text{th}} \Sigma_{\text{Ath}}^{235} \phi_{\text{th}} + \epsilon \eta_r \Sigma_{\text{Ar}}^{235} \phi_r}{1 + \tau B^2}$$

or:

$$(1 + \tau B^2) = \frac{\epsilon \eta_{\text{th}} \Sigma_{\text{Ar}}^{235}}{(\Sigma_{\text{Ar}} + \Sigma_{\text{Rr}}) R} + \frac{\epsilon \eta_r \Sigma_{\text{Ar}}^{235}}{\Sigma_{\text{Ar}} + \Sigma_{\text{Rr}}}$$

But

$$\frac{\Sigma_{\text{Ar}}^{235}}{\Sigma_{\text{Ar}} + \Sigma_{\text{Rr}}} = \frac{\Sigma_{\text{Ar}}^{235}}{\Sigma_{\text{Ar}}} (1 - P)$$

where P is the total escape probability from the resonance group. Thus,

$$\frac{\eta_{\text{th}}}{R} = \frac{1 + \tau B^2}{(1 - P) \epsilon} \frac{\Sigma_{\text{Ar}}^{235}}{\Sigma_{\text{Ar}}} - \eta_r$$

4. Lattices Studied and Results

Uniform burnup studies of the various lattices are summarized in the table below. The enrichment specified is that which will provide about 16% excess reactivity at operating conditions at the beginning of the cycle (before xenon buildup). This reactivity includes allowances for the structure necessary in an actual core configuration, for equilibrium xenon and samarium, and for burnup and accumulation of heavy isotopes and fission products. The set of equations described in Part 2 of this Appendix were solved on the analogue computer and the resulting curves for reactivity, concentration of U^{235} , and concentration of Pu^{239} are shown in Figs. A-1 through A-9. The cycle length is determined by the time when the reactivity margin available for burnup falls to zero. No consideration is given to corrosion or possible material limitations here.

Table A-3

LATTICE STUDIES AND RESULTS

Lattice No.	Rod Dia. in.	Non-water Water	Clad in.	Volume ℓ	Enrichment %	Cycle Length yr.	Mwd/M.T. Exposure	U Loading kg	End-of-cycle Pu ²³⁹ Inventory kg	Conversion Ratio
1 (BW)	.5	1	.050 Al	2026	1.78	4.35	15210	5180	55	.67
2 (BW)	1.0	1.27	.031 Zr	2026	1.44	6.5	14500	7978	49	.84
3 (BW)	1.0	1.27	.015 SS	2026	1.75	5.7	12060	8551	40	.85
4 (BW)	1.0	1.27	.025 SS	2026	1.93	8.2	18070	8204	72	.82
5 (BW)	.5	1	.017 Zr	2026	1.59	2.95	7580	7029	18	.78
6 (PW)	.5	1.27	.020 Zr	953	1.80	3.8	19070	3605	28	.80
7 (BW)	.5	1.27	.017 Zr	2026	1.50	6.53	15030	7865	59	.86
8 (PW)	.5	1.27	.017 Zr	1426	1.65	5.0	16560	5536	40	.83
9 (OM)	.150 ¹	1.64 ²	.030 Al	1475	1.50	4.4	9900	6238	50	.87

¹Plate thickness²Atom ratio: Hydrogen to Uranium

Lattice 9, the organic moderated core, is included only for completeness. Its initial reactivity is 12.8% based on a core calculated to include structure, spacing and rod channels. The organic moderated core uses plate-type fuel elements and metal fuel plates making a straight comparison of the systems difficult.

From the results tabulated above, the following conclusions may be drawn:

- 1) The allowable fuel exposure increases greatly with the volume ratio of non-water (fuel + clad) to water.
- 2) Enrichment must be increased as the fraction of neutrons captured parasitically is increased. Greater exposure is possible with a greater initial U²³⁵ inventory.
- 3) In a pressurized water core, greater exposure is possible from a given initial U²³⁵ inventory than in a boiling core; however, the fuel cycle is appreciably shortened.
- 4) The fuel cycle characteristics are insensitive to fuel rod diameter.

References - Appendix A

1. Hughes, D. J., and J. A. Harvey, "Neutron Cross Sections," McGraw-Hill Book Company, Inc., 1955.
2. "Reactor Physics Constants," ANL-5800, 1958.
3. "704 Program-I₂," AGT Flight Propulsion Laboratory, Evandale, Ohio.

FIG. A-1
LATTICE 1 - TABLE A-3

FIG. A-2
LATTICE 2 - TABLE A-3

FIG. A-3
LATTICE 3 - TABLE A-3

FIG. A-4
LATTICE 4 - TABLE A-3

FIG. A-5
LATTICE 5 - TABLE A-3

FIG. A-6
LATTICE 6 - TABLE A-3

FIG. A-7
LATTICE 7 - TABLE A-3

FIG. A-8
LATTICE 8 - TABLE A-3

FIG. A-9
LATTICE 9 - TABLE A-3

14

15

16

17

Appendix B

EVALUATION OF AL-ALLOYS FOR REACTOR APPLICATION

Aluminum alloys particularly 2S(1100) have been favored as a fuel cladding material in water cooled reactors operated at low temperature because of their low nuclear cross section, low cost, and adequate corrosion resistance. Attempts to use them at higher temperatures result in difficulties due to their increase in corrosion rate with temperature. The static corrosion rate curve⁽¹⁻²⁾ changes from about 0.05 to 1.3 mils per year from 250 to 390°F. Above 390°F catastrophic intergranular attack occurs instead of the uniform corrosion occurring at lower temperatures. It has been reported⁽¹⁻³⁾ that intergranular attack can be prevented by addition of acid, and the pH of minimum attack decreases with increasing temperature. The magnitude of this effect ranges from pH 6.5 at 120°F to pH 2 at 600°F.

As a result of further investigation into methods of improving the corrosion behavior of aluminum alloys at elevated temperature, Draley and Ruther⁽⁴⁻⁵⁾ found that intergranular attack could be eliminated or minimized by the use of a metal with high hydrogen overvoltage such as nickel in one of three methods: by introducing a nickel salt into the water; by plating the aluminum with metallic nickel; or by introducing the nickel into the aluminum as an alloy constituent producing a finely dispersed second phase. A theory covering the effects was propounded. Other metals found beneficial were Cu, Co, Fe and Pt; Ni however was considered the best. They found in distilled water static tests on a 1% nickel alloy in 2S aluminum uniform corrosion rates at different temperatures as follows:

482°F	1 mil per year
554°F	1½ mils per year
599°F	3 mils per year
662°F	9 mils per year

Of the three methods of protection proposed by Draley and Ruther, both the nickel plate and nickel alloy appeared more attractive for reactor application than the use of nickel salts, and were subjected to an extended series of dynamic engineering type tests⁽⁶⁻¹³⁾ over a period of several years by ANL-RED. The nickel plate was found to be subject to failure where the coating was faulty⁽⁶⁻⁸⁾ and tended to come off when exposed to neutron flux.⁽⁷⁾ For these reasons further testing of nickel plate was discontinued.

(8) The dynamic tests on Al-Ni alloys showed a marked velocity effect. Semi-static test sections in dynamic loops showed corrosion rates very close to those found in static autoclave tests, but the rates increased with increased velocities. At 7 ft/sec rates were higher than in semi-static

tests and at 16-21 ft/sec the rates were still higher. It was also observed that samples exposed to 16-21 ft/sec tests showed an accelerated erosion-corrosion attack on leading edges and other local areas of still higher velocity.⁽⁸⁾

Another observation,⁽⁹⁾ that defied explanation for a long time, was the difference in corrosion rates, in 500°F high velocity tests between originally loaded samples, and the replacement samples. Another baffling characteristic of the dynamic high velocity tests at 500°F was the lack of reproducibility of results between supposedly identical runs both at ANL and elsewhere.

The promising results with Al-Ni alloy and others obtained by Draley and Ruther in static tests on high temperature water were confirmed by other investigators⁽¹⁴⁻¹⁹⁾ in U. S., Canada, and England. The interest in this country was reflected in the activity of the Aluminum Alloy Task Group which from its inception in 1955 to December 1957 was sponsored by the Naval Reactors Branch of the AEC, and subsequently by the Engineering Development Branch.

The purpose of the Group was avoidance of unnecessary duplication and the quick interchange of information between organizations actively engaged in the development of aluminum alloys resistant to high temperature water. Although some of the reports are not readily available they constitute a valuable source of information to those who do have access to them.⁽²⁰⁾ Much of the information has been published subsequently.

Development activity has been aimed in several general directions as follows:

- a. Development of alloys of improved corrosion resistance.^(16-19,21-24)
- b. Development of corrosion resistant alloys of improved mechanical properties.⁽²⁴⁻²⁶⁾
- c. Getting a better understanding of the nature of the protective film.^(18,22,30)
- d. Learning more regarding the corrosion mechanism.^(22,28)
- e. Investigation of effects of water environment on corrosion behavior.^(19,22-24,29-32)
- f. Evaluating effects of test geometry.^(18,22,33)
- g. Studying effects of irradiation.^(7,10,11,18,33)

It should be understood that there is a great deal of overlapping among the references listed under the above categories.

As an aid in the evaluation of Al-Ni for reactor application, results of tests at ANL, WAPD, Hanford, and Chalk River in which corrosion rates were determined are summarized in Tables B-I, B-II, B-III and B-IV.

Information for the first three tables came largely from the Aluminum Alloy Task Group Reports.⁽²⁰⁾ The Chalk River data in Table B-IV came from references indicated in the table.

The RED results in the table, in combination with those in the references suggest the following conclusions:

1. The corrosion rate at 600°F is higher than that at 500°F.
2. The corrosion rate increases with the velocity.
3. The corrosion rate for replacement samples is higher than that for original samples. The difference between them is greater at 500°F than at 600°F.
4. The corrosion rate at pH about 5 is lower than that at about pH 7.
5. The corrosion rate is higher in the absence of hydrogen than when it is present.
6. The corrosion rate at 475°F appears higher than that at 500°F, but it is believed that other factors had more effect than the 25 degree difference in temperature.
7. Under boiling heat transfer conditions [422°F and 25,000 Btu/(ft²)(hr)] the rates obtained ranged from less than 1, to about 3 mils/yr. No significant pitting or other accelerated attack was observed.
8. The apparent effect of irradiation, was to reduce slightly the corrosion from that found in the out-of-pile section. The high rates found were attributed to poor average water quality in one case, and to erosion effects in the other, combined with low Al/H₂O ratio.
9. The corrosion rates in dilute H₃PO₄ solutions were lower than those in H₂SO₄ of the same pH.
10. Dynamic corrosion rates at 500°F showed a greater range of values and some higher values than those at 600°F.
11. The range of values at 500°F was greater than that exhibited by increasing the temperature from 500°F to 600°F. That is, the effect of temperature is not as important as some other not well understood factors such as Al/H₂O ratio. The Al/H₂O ratio effect appears to be influenced by other conditions such as pH, flow rate, temperature, etc. At pH > 7 a high Al/H₂O ratio does not seem to be as effective as at a lower pH; and at low velocities the ratio has no effect.

12. Among the alloys proposed for high temperature water application, the differences due to alloy composition are generally less significant than those due to differences in water conditions.

13. While the use of H_3PO_4 markedly reduces the uniform corrosion rate of high temperature Al-alloys, long term tests have been characterized by pitting attack at pH values ranging from 3.5 to 5.5 at 600°F. Alloys of different composition show differences in this respect. Also more attack of stainless steel occurs.

14. The high temperature Al-alloys appear to corrode by a spall-heal process that results in a considerable amount of aluminum oxide crud in the system. The amount of crud appears to be greater in systems maintained at low pH.

15. Corrosion rate increases with flow. A definite erosion-corrosion effect becomes evident around high turbulent areas at leading or trailing edges and around clamps.

16. Velocity effects appear to be accelerated at pH values above 7.0.

Some of the results obtained on X-9001 by WAPD⁽²⁰⁾ are summarized in Table B-II .

The WAPD results, all of which were obtained at 600°F or higher, suggest the following conclusions.

1. In pure water, rates at 680°F are slightly higher than those at 600°F, but there is not much difference between static tests in water or steam, and dynamic tests.

2. Rates at pH 3.5 are slightly lower than those at pH 7; and those at pH 7 are generally slightly lower than those at pH 8.5-9.0.

3. Rates in pH 3.5 solutions containing H_3PO_4 are lower than those containing nitric acid.

4. At 680°F a break occurs after about 70-84 days in deionized water after which the rate increases by about three times. After about 200 days a second break occurs with another factor of 3 increase in rate.

5. Under nominally similar conditions, rates obtained at WAPD are in general lower than those obtained at ANL. However, in one test at pH 8.4-9.4 a rate of 62.5 mils/yr was obtained at 600°F.

Some of the results obtained on X-8001 at Hanford⁽²⁰⁾ are summarized in Table B-III.

Some conclusions based on these results and on references reports are as follows:

1. The corrosion rate depends on flow. Between 0 and 2 fps the rate increases by twenty times. Between 2 and 12 fps there is no further change.

2. In long term tests in deionized water at 662°F and 685°F a break occurs, after which the rate is accelerated.

3. Corrosion increases with the amount of irradiation received.

4. Corrosion rates at 500°F range from 0.3 mils/yr in pH 4.5 H_3PO_4 water solution to 109 mils/yr in high purity water.

5. With a high ratio of metal to water the corrosion rate is decreased.

6. It is believed that the lower rates under semistatic conditions are due to the higher concentration of corrosion products in the water in contact with the sample. The effect of increased flow is to reduce this concentration and thus increase the corrosion rate. The high rates obtained by Hanford as compared with those at ANL and WAPD are believed to be due to a low metal to water ratio at Hanford as compared with the high metal-water ratio at ANL and WAPD.

7. H_3PO_4 solutions of pH 3.5 attacked sensitized SS304 at 302-356°F after 12-17 weeks.

8. 198X alloys prepared by ANL showed improved resistance over X-8001 and M-400, but Hanford melts were inferior to ANL melts.

9. High heat through-put appears to increase corrosion rate.

10. At pH = 4 (H_3PO_4) 480°F tests showed no attack of sensitized SS304 up to five months.

11. Al/ H_2O ratio and loop cleanup rate markedly influence corrosion rate.

12. The French Alloy Al, 2.1% Fe, 0.7% Ni appears to be as good as ANL 198X.

13. For temperature increase 180°C to 250°C in dynamic tests, the corrosion rate increases 3.5 times, equivalent to a factor of 0.5 for every 18°F rise.

14. A 203X showed very low uniform rates, but evidence of blistering at pH 4.5 (H_3PO_4) 570°F with 450 cm^2 Al/gal water.

15. Experimental evidence indicates that a combination of pH 4.5 (H_3PO_4) and high Al/ H_2O ratio is better than either alone.

16. Preliminary results indicate the possibility of a compatible carbon steel-aluminum alloy system at low pH values by using dichromate inhibitor and nitric acid to lower pH.

17. Dynamic loop tests at 390, 480, and 570°F indicate no replacement effect at 390°F but replacement effects at 480 and 570°F.

18. In these studies, an increase in temperature from 390 to 570°F increased corrosion rate by a factor 2-3.

19. Attempts to correlate corrosion rates at 390, 480 and 570°F based on the difference in solubility of the corrosion product indicated that observed rates were several times higher than predicted values. The presence of large amounts of aluminum oxide corrosion product in the form of flakes during a run suggests an erosion effect rather than anomalous high solubility effect.

20. The pronounced beneficial effect of pretreatment by autoclaving may result from an aging of the corrosion product which renders it less soluble. The aging of aluminum oxide hydrates with resulting marked decrease in solubility is well known.

21. Based on static tests in 0.2 M acetate buffer, the pH of minimum corrosion at various temperatures is as follows:

572°F	<3.1
536°F	<3.1
491°F	4.0
401°F	4.9

22. At 500°F the corrosion rate of X8001 was found to be 13.2 mils/year. Under a heat flux of $1.39\text{--}1.4 \times 10^5$ Btu/(hr) (ft^2) and average surface temperatures of 514-518°F tubes of X8001 corroded at a rate of 21.6 mils/year. Calculations showed that for every 9300 Btu/(hr) (ft^2), 2°F must be added to the calculated surface temperature to obtain the true corroding temperature. Film thickness determinations by eddy current and film weight calculations show it to be 2.5-2.7 mils with a density of 3.0 gm/cm^3 . Thermal conductivity estimates for the film are 1.4-1.7 Btu/(hr) (ft) (°F).

23. There are three general types⁽³²⁾ of aluminum corrosion. The most destructive type:

below 122°F	is pitting
122°F-482°F	is uniform attack
above 482°F	is intergranular attack

In investigating the effects of various ions on uniform corrosion it was found that pH was most important variable. Phosphate inhibited corrosion, citrate increased it. Corrosion rates were lowest at lowest flow rates; most important parameters of uniform Al corrosion are temperature, time, water composition and flow rate.

24. T. A. Ayres⁽³³⁾ attempted to predict the corrosion behavior of the high temperature aluminum alloys in reactors by means of equations, graphs, etc. It was based on the corrosion rate being a logarithmic function of time; that corrosion increases with irradiation (reactor power); that optimum pH = 4.0; and two assumptions i.e., (1) that the use of phosphate or other inhibitors in reactors will become practical and will decrease the corrosion rate by a factor of ten times over that found in the Hanford H-loop, and (2) by developing improved alloys and learning enough about the optimum loop conditions, the corrosion rate can be lowered by another factor of 5.

On these bases, and excluding the mechanical factors, he concluded that aluminum may well be potential material for cladding fuel elements for reactors operating at outlet temperatures up to 662°F. He has taken observed corrosion rates and compared them with calculated rates. He concluded that the predicted values do not fit data from corrosion experiments in deionized water in static or low flow systems either at HAPO or elsewhere; neither does it fit data from corrosion experiments in deionized water in flow tubes or pilot plants at sites other than HAPO.

He concedes that more accurate information is needed to make the predictions truly realistic.

25. The Chalk River results,⁽³⁴⁾ some of it included in Table B-IV, confirmed the earlier work on high temperature corrosion resistant alloys containing Cu, Fe and Ni but found improvement when Si, Bi, Zr and Ti were added.

The best alloys show acceptable rates 1.3-2.1 mils/yr in static tests at 572°F but under dynamic conditions at 18-20 fps rates increase about 65 times to about 130 mils/year. The low rates are reached in static tests only after corrosion films 10-20 μ in thickness are built up in at least two distinct layers and in a 15 week static run about 15 per cent is lost through spalling and dissolution.⁽²²⁾ The higher rates under dynamic conditions

are believed to be due to a combined erosion-corrosion effect that may be due to differences in test temperature, details of sample design and mounting and characteristics of individual loops. This may help to explain the differences in results found in different sites. Due to the accelerating effect of mechanical damage on corrosion it is difficult to establish corrosion rates at 18-20 ft/sec. Maximum corrosion rates for freely vibrating samples may be as high as 0.030 in./month, while rates for flat samples rigidly clamped have been as low as 0.020 in./yr.⁽¹⁸⁾

In regions of linear corrosion attack (dynamic 18-20 ft/sec) there was a continuous spalling and reformation of the corrosion films in an uneven manner. Pitting was observed and accelerated attack at leading edges. A 5X reduction in corrosion rate was observed by using 1000 ppm dissolved silica. The protection however was erratic, in loop tests it reduced the uniform corrosion and spalling of most specimens, but resulted in a severe localized pitting attack, particularly at leading edges.

Increasing the Al/H₂O ratio from 2 cm²/liter to 170, 360, and 565 cm²/liter decreased the corrosion rate from >130 mils/yr to about 25 mils/yr, apparently by saturating the water with dissolved corrosion product, and reducing film dissolution corrosion.

While methods of saturating the loop water have reduced corrosion by a factor of 5, there is still no explanation for the 10X increase over static tests.

Summarizing

1. Corrosion rates on Al-Ni alloys above 392°F are higher in dynamic tests than in static tests, and the higher the flow rate the higher the corrosion rate. Velocity effects are greater at pH values above 7.0 than below.
2. Under static and dynamic conditions the corrosion rate increases with temperature.
3. At 680°F the corrosion in stagnant static autoclaves over a long period of time makes two transitions, each rate being about three times the previous rate.
4. The range or spread in corrosion rates observed in dynamic tests at 500°F has been greater than that at 600°F.
5. The range observed at 500°F has been greater than the effect of temperature between 500°F and 600°F dynamic tests.
6. The difference between the corrosion rates between original and replacement samples in a dynamic loop has been greater at 500°F than at 600°F.

7. The corrosion rate in dynamic systems is higher above pH 7 i.e. 7-9, than below 7 i.e., 5-7.

8. The corrosion rate at a low pH produced by H_3PO_4 is lower than that of the same pH produced by HNO_3 or H_2SO_4 .

9. The use of low pH by means of H_3PO_4 may introduce corrosion problems with stainless steel, pitting of Al-Ni and increased activity problems in a reactor. Development work would be required before it could be used in a reactor.

10. The aluminum surface/water ratio is an important parameter in the corrosion rate of Al-Ni alloys. The high ratio is conducive to low rates and the low ratio to high rates. It is more important below pH 7 than above pH 7. But it is not important at very low velocities.

11. Pretreatment by autoclaving appears to favor low rates, at least for a time. Further work is needed in this area.

12. Data on the effect of irradiation on the corrosion of Al-Ni alloys is contradictory. ANL and Chalk River have found the effect to be minor. Hanford finds the effect significant. Further work is desirable.

13. Data on the effect of heat through-put is in a similar condition but both Chalk River and Hanford encountered excessive rates in tests with high heat transfer rates.

14. If the function of the increased Al surface to water ratio is to saturate the water with aluminum oxide corrosion product, then operation of a clean up loop is working in an opposite direction, to remove the dissolved (or suspended) corrosion product. More investigation is required in this area.

15. The Al-Ni alloys developed for use above 392°F show less effect of changes in alloy composition than by changes on water conditions, but under optimum water conditions, it is believed that the differences will become more apparent. More work is required in this area and in the fabrication of alloys such as A203, 198X etc.

16. There is indication that some of the apparently abnormally high dynamic corrosion rates may result from an erosion effect of aluminum oxide corrosion product suspended in the water, and high flow rates.

Recent ANL Experience with Al-Ni Alloys in an In-Pile Loop Test, and in Reactors

Static and dynamic tests of Al-Ni alloys out-of-pile and in-pile have for their purpose the investigation of factors which are surmised to effect the corrosion behavior so as to anticipate or predict the actual behavior as a cladding for a fuel element in a reactor.

Argonne has some actual experience with Al-Ni alloy clad fuel elements in reactors, and also with a clad fuel element of a type used in a reactor. Evaluation is incomplete but the results to date should be included in this survey.

Clad Element in In-Pile Loop Test

A BORAX IV type fuel element, $\text{UO}_2\text{-ThO}_2$ pellets in a X-8001 tube plate, was tested in ANL-2 at MTR for 5 cycles (1607 hrs at 465°F). Test conditions were:

temperature	465°F
pressure	600 psig
flow	12-13 ft/sec
maximum heat flux	800,000 Btu/(hr) (ft ²)
maximum burnup	1.5 w/o total core atoms
average burnup	0.5 w/o total core atoms
pH	6.0-7.4
resistivity	1.5-3.0 megohm cm

After the test, the in-pile and out-of-pile plates were subject to film stripping to determine metal loss due to corrosion. Since the sample represented only one point, the corrosion rate could not be determined but the metal loss for the in-pile plate was 15.9 mg/cm² while that for an out-of-pile plate was 8.57 mg/cm². The in-pile plate with boiling on part of its surface was at higher average temperature than the out-of-pile plate. Since 6.9 mg/cm² = 1 mil, the two weight losses could represent maximum corrosion rates of 12.4 mils per year for the in-pile plate with heat flux and irradiation and 6.7 mils/yr for the out-of-pile plate without heat flux and without irradiation.

The relatively good corrosion behavior shown in this experiment and in appearance of the samples in combination with the very low Al/H₂O ratio, 6 cm²/liter, and relatively high flow rates compared to semi-static conditions do not appear to be in accord with the results of the summary given previously. It does tend to support the Chalk River and ANL results with reference to effect of irradiation and heat through-put.

BORAX III fuel elements were Al-U alloy 93.21% enriched clad with 2S aluminum, water temperature was 419°F and pressure, 300 psig. Water

resistivity averaged 400,000 ohm cm and pH 6.0 - 6.5. Boric acid was present for about six months, 410 gm/1000 gal. Heat flux averaged 39,800-53,000 with a maximum of 92,800 Btu/(hr) (ft²). Punchings from elements removed from the reactor after 600 Mwd operation showed the surface to be non-uniform with many surface pits and oxide blisters. None of the pits appeared to penetrate the cladding but analysis of the water from the casks suggested the presence of traces of fuel.⁽³⁵⁾

The presence of boric acid throughout a considerable part of the total period may have helped to reduce the corrosion of the 2S.

BORAX IV fuel elements were thoria-urania pellets in X-8001 tube plates. Operating conditions were 420°F, 300 psig. Water quality was maintained at 300,000-600,000 ohm cm and pH 5.7-6.2 by cation resin in hydrogen form operating in parallel with a mixed bed. Total operation was 980 Mwd/t (about 2.3% total U²³⁵ burnup). Total exposure was about 100 operating days at 420°F and about 20 decay days 200°F-150°F. Maximum heat flux was about 288,000 Btu/(hr) (ft²). The run was terminated when evidence of fuel element rupture was obtained. Examination resulted in finding evidence of leakage in more than half of the elements.

Examination of two faulty elements in the cave at ANL resulted in finding seven leaks in 12 plates.

"The results of the bubble testing indicate two significant facts. No leaks were evident at the silicon bonded areas and the areas that did leak were not severely corroded. All the leaks, which were noted, were either in the top collapsed sections or near the bottom of the plates."⁽³⁶⁾

It is interesting to note that the fuel element failures all occurred over a short period of time after about a year of successful operation.

EBWR at 600 psig, 489°F contained 34 X-8001 dummy fuel elements which were in the reactor for about 2½ years about one-half of which was at temperature. The aluminum picture is confused because the condenser tubes are also aluminum. The line from the hotwell to the reactor, however, carries all of the water through full flow cotton filters, and it does not seem likely that significant quantities of aluminum oxide enter the reactor from this source. (An investigation of this point is planned.)

Although corrosion of the dummy elements does not appear excessive from examinations that have been made, estimated to be 3 mils/yr⁽³⁷⁾ a definite crud problem exists in EBWR. The aluminum oxide corrosion product along with nickel, iron, etc., deposits on the fuel elements, gradually spalls off and is carried to low velocity parts of the system, particularly control rod thimbles resulting in an accessibility problem. The channels

between fuel plates are gradually decreasing in size due to the scale build-up. Examination of a fuel element now underway gave preliminary indications that the scale was in places about 5 mils in thickness.

ALPR - This reactor, operating at 420°F and 300 psig has Al-17% U (93% enriched) fuel elements clad with, and silicon bonded to, X8001 aluminum alloy. Operating time has been so short that no significant data on fuel element behavior is yet available.

Conclusions

With regard to the corrosion behavior of Al-Ni base alloys, lack of sufficient knowledge regarding the mechanism of corrosion under reactor conditions prevents a realistic prediction under any given reactor environment. Presently unknown factors have more effect than does the simple increase in temperature. All that can be said is that the conditions in BORAX IV may be satisfactory for M-388. Failure of more than half of the elements over a short period after a year of successful operation may have been due to faulty mechanical design. Behavior of another core without faulty design might prove it. Conditions in EBWR may be satisfactory but the crud problem doesn't look attractive. The dummy elements containing no fuel are exposed to lower flows, lower temperatures, and less turbulence due to boiling than are the fuel elements. We cannot say with any degree of certainty that M-388 (X8001) would be satisfactory as fuel element cladding in EBWR, without experimental proof in a reactor.

Table B-1
ANL RESULTS ON X-8001 (M-388)

Loop	Run No	Temp °F	pH range	Additive	H ₂ cc/liter	Al/H ₂ O cm ² /liter	Type of sample	Corrosion rate mils/yr			Comments
								16-22 fps	6-7 fps	semi static ¹	
1	1	500	4.9-5.1	H ₂ SO ₄ ²	N, ³ 23-111	92	O ⁴	2.0	1.4	1.2	regular samples pretreated samples
1	4	500	5.8-6.1	H ₂ SO ₄ ²		25		10.3			
1	4	500	5.8-6.1	H ₂ SO ₄ ²		25		6.9			
1	5	500	5.0-6.8	H ₂ SO ₄ ²	N, 5-560	59	O	4.2	2.5		
1	5	500	5.0-6.8	H ₂ SO ₄ ²	N, 5-560	59	R	7.3	4.0		
2	1	550	8.9-9.3	NH ₄ OH ²	N, 100	0	O	7.1		2.6	
3	1	500	6.5-6.9	deion	N, 41-540	73	O	2.0	1.5	1.5	
3	1	500	6.5-6.9		N, 40-350		R	5.9	6.1		
3	5	475	7.1-7.5		N, 30-320		O	8.7	2.7	1.2	
3	5	475	7.1-7.5		N, 30-320	71	R	17.4	4.5	1.9	
3	4	500	6.8-7.0		N, DG, 0-310	49	O	8.5	5.1		
3	4	500	6.8-7.0		A, 220-250		O	10.7	4.5		
4	1	500	5.3-6.3	H ₃ BO ₃	A, 100		O	5.1			
4	2	500	6.3-6.9	deion	DG	12	O	11.0			
4	2	500	6.3-6.9	deion	DG		R	16.5			
3	6	500	7		N		O,R	6.3	1.9		
4	3	500	6.4-6.7		A, 150	8		11.0			
1	2	600	5.3-5.9	H ₂ SO ₄ ²	N, 31-250	84	O	6.0	4.7	2.8	
1	2	600	5.3-5.9	H ₂ SO ₄ ²	N, 31-250	84	R	8.6	3.2	3.5	
1	6	600	5-6	H ₂ SO ₄ ²	N		R	13.6	7.4		
2	2	600	4.5	H ₃ PO ₄ ²			O	0.77		0.36	
2	2	600	4.5	H ₃ PO ₄ ²			R	very high			
3	2	600	6.8-7.0	deion	A, 175-540	76	O	8.5	6.5	5.5	
3	2	600	6.8-7.0	deion	A, 175-540	76	R	10.9	8.8		
3	7	600	5.5	H ₃ PO ₄ ²			O	4.2	1.9	0.7	
3	8	600	5.5	H ₃ PO ₄ ²			O	2.6	0.7		
3	8	600	5.5	H ₃ PO ₄ ²			O	0.8	0.5		198 Xalloy
4	8	600	7.0-7.4		DG	16	O	11.7			
4	8	600	7.0-7.4		DG	16	R	16.6			
4	9	600	7.2-7.3		DG	30		9.8-12.5			A series of ALCOA alloy vel not known, natural circulation
Boiling rig #2	1,2,3	422	6.5-7.5	deion	DG	92	clad tube		3.3		
Boiling rig #3	1	422	6.5-7.5	deion	DG	57-131	clad heater	1.7			6.7 watts/cm ² { vel not known, natural cir
								1.2			
								9			
								4			
MTR-ANL-2	cart-ridge #7	485-500	6.2-7.8	deion ⁵	DG	3.5		11.6			15 fps, in-pile
								17.4			2-9.8 x 10 ²⁰ nvt th
											15 fps, out-of-pile
MTR-ANL-2	cart-ridge #9	485-500	5.6-9.5	deion	DG	4.4		16.7 ⁶			15 fps in-pile
								17.4			2.55-3.22 x 10 ²⁰ nvt th
											out-of-pile

¹Estimated flow 1/60 fps

²Acid added by means of ion exchange bed

³N = natural hydrogen buildup, A = hydrogen added, DG = degassed

⁴O = original sample, R = replacement samples

⁵ion exchange bed was turned off part of time inadvertently

⁶considerable erosion effect observed

Table B-II
WAPD RESULTS ON X-8001 (M-388)

Temp. °F	Type of Test		pH	Addition	Corr. Rate mils/yr	Comments
680	Static	Water			4.0	
680	Static	Steam			4.7	
600	Static	Water			3	
680	Semi-Dynamic	Water	6.5-7.5		4	
680	Semi-Dynamic	Water	3.5	HNO ₃	2	
680	Semi-Dynamic	Water	3.5	H ₃ PO ₄	0.5	
680	Semi-Dynamic	Water	8.5-9.0	NH ₄ OH	3.4	
600	Dyn. 30 fps	Water	8.5-9.0	NH ₄ OH	5.0	
650	Dyn. 20 fps	Water	8.5-9.0	NH ₄ OH	8.4	
680	Static	Water			5.2	
600	Static	Water			6.0	
600	Dyn. 20 fps	Water	8.5-9.0		6.7	
680	Static	Water			3.95	14-84 days
					13.8	84-126 days
680	Static	Water			3.47	14-84 days
					14.3	84-126 days
680	Static	Steam			4.6	14-70 days
					13.0	70-112 days
600	Static	Water			2.6	
680	Dynamic	Water	7.0		5.0	
650	Dyn. 20 fps	Water	8.5-9.0	NH ₄ OH	6	
650	Dyn. 20 fps	Water	3.5	H ₃ PO ₄	0	
650	Dyn. 4 fps	Water	3.5	H ₃ PO ₄	0	
680	Static		6.5-7.5	deion. water	3.27	14-98 days
					13.19	98-196 days
					39.06	196-238 days
					5.40	14-70 days
					8.55	70-182 days
					6.06	14-126 days
					25.31	126-210 days
680	Static	Water- steam	6.5-7.5		4.55	7-80 days
					12.60	80-140 days
					30.10	140-210 days
600	Static	Water	6.5-7.5		2.10-4.53	7-210 days
					6.08-6.23	77-224 days
680	Circulating 1.5 l/hr	Water	6.5-7.5	deion. water	5.6	14-95 days
600	Dynamic		8.5-9.4	NH ₄ OH	62.5	20 fps
					5.18	1 fps

Table B-III
HANFORD RESULTS ON M-388

Type of Test	In-Pile	pH	Additive	Corrosion Rate mils/yr ¹												
				Temperature °F												
				685	662	536	500	482	464	446	437	428	392	356	320	302
Dynamic	Yes	5.5	Cation I.E.			103	70	54	46	31		24	14	7		
Dynamic	Yes	High purity						109	84			50	28	15		
Dynamic	Yes	4.5	H ₂ SO ₄					52	42			24	14	7		
Dynamic	Yes	4.5	H ₃ PO ₄						10			7	4	3		
Dynamic 2-4 fps	No	-	Al ₂ (SO ₄) ₃					17.5								
Dynamic 2-4 fps	No	-	High ratio Al/H ₂ O					6-8								
Dynamic	Yes	6.0	-										72	60	38	
Dynamic	Yes	5.0	-					44			24		12			
Dynamic 10-15 fps	No	4.5	H ₃ PO ₄ I.E.													0.6
Dynamic 10-15 fps	No	3.5	H ₃ PO ₄ I.E.													0.6
Dynamic 20 fps	No	4.5	H ₃ PO ₄ I.E.					0.3								
Static (low flow)	No	7.0	Deionized		2-2.5 ² 5.5-6 ³											
Static (low flow)	No	7.0	Deionized	3 ⁴ 20 ⁵												
				662	572	554	500	482	441	435	356	347	302	Comments		
Dynamic 10-15 fps		3.5-4.5	H ₃ PO ₄										0.60			
Dynamic 20 fps		4.5	H ₃ PO ₄		1.3											
Dynamic		6-7	Deionized		24											
Dynamic		6-7	Deionized				5.4							High Al/H ₂ O ratio		
Static		6-7	Deionized	2.2												
Dynamic		8.5-9.5	NH ₄ OH			9.0								Low Al/H ₂ O ratio		
Dynamic		8.9	NH ₄ OH + Al(NO ₃) ₃					8.8								
Dynamic		4.0	H ₃ PO ₄							0.094						
Dynamic		4.0	H ₃ PO ₄				0.33									
Dynamic		4.5	H ₃ PO ₄		0.06									High Al/H ₂ O ratio		
Dynamic		4.5	H ₃ PO ₄						1.1					In-pile		
Dynamic		4.5	H ₃ PO ₄							120				In-pile very high		
Dynamic														heat flux A-2 alloy		
Dynamic		4.5	H ₃ PO ₄					1.44						In-pile A-2 alloy		

¹ In most cases estimated values taken from graphs

² Penetration after first 6 months

³ 49 weeks penetration

⁴ 1-9 weeks rate

⁵ 9-15 weeks rate

Table B-IV
CHALK RIVER RESULTS OF DYNAMIC TESTS ON Al 0.5% Ni, 0.5% Fe, 0.2% Si

Temp. °F	Flow, ft/sec	pH	Additive	Corrosion Rate mils/yr	Comments	Reference
500	2-3	9.7	Li OH	7-10	Irradiation ¹	(34)
500	20	6.5-7.0	Deionized	17-20		
550	20	6.5-7.0	Deionized	100-120	Irrad. ³ heat flux ²	(18)
500-550	20	6.5-7.0	Deionized	2	Prefilmed in 570°F static water	(16) (20)
				8	High Al/H ₂ O ratio	
				45	Low Al/H ₂ O ratio	
				45	Replacement samples	
572	Static		Deionized	1.2-2.1		(22)
500	Dynamic	6.5-7	Deionized	130		(22)
500	Dynamic	6.5-7	Deionized	25	170, 360, 567 cm ² Al/liter H ₂ O	(22)
500	Dynamic	6.5-7	Trace diss. SiO ₂	13	170	(22)

¹ The slight amount of ionizing radiation employed appeared to reduce the corrosion rate. The effect is not considered significant.

² Samples exposed to ~230,000 Btu/(ft²)/(hr) showed about 10% estimated corrosion more than those exposed to ~35,000 Btu/(ft²)/(hr).

³ Slightly accelerated effect with enriched UO₂ fuel - not considered significant.

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