UNITED STATES ATOMIC ENERGY COMMISSION

PATHFINDER ATOMIC POWER PLANT
TECHNICAL PROGRESS REPORT FOR
APRIL 1, 1958—JUNE 30, 1958

June 30, 1958

Allis-Chalmers Manufacturing Company
Milwaukee, Wisconsin

Technical Information Service Extension, Oak Ridge, Tenn.

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For Northern States Power Co. and Central Utilities Atomic Power Associates.

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TECHNICAL PROGRESS REPORT
APRIL 1, 1958 - JUNE 30, 1958

Submitted to

U. S. ATOMIC ENERGY COMMISSION,
NORTHERN STATES POWER COMPANY
and
CENTRAL UTILITIES ATOMIC POWER ASSOCIATES

Under

Agreement dated 2nd Day of May 1957 and Supplemental
Agreement dated 22nd Day of November 1957 between
NSP and A-C under AEC Contract No. AT (11-1)-589

by

ALLIS-CHALMERS MANUFACTURING COMPANY
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INTRODUCTION

This is the third of a series of quarterly reports titled "Technical Progress Reports." The reports chronicle in some detail the work progress relative to the design and construction of the Pathfinder Atomic Power Plant during a three-month period. Distribution of the reports is made to the U. S. Atomic Energy Commission, Northern States Power Company, and Central Utilities Atomic Power Associates.

It will be noted that accelerated effort has been initiated on determining suitable materials for fuel cladding. With the completion of one corrosion loop and two more to be assembled in the near future, more and more tests will be conducted to determine corrosion properties of materials under dynamic conditions.

The main features of the Critical Facility are described. Details are provided in the "Preliminary Safeguards Report - Allis-Chalmers Critical Facility," Report No. ACNP-5809.

1. FUEL ELEMENT RESEARCH AND DEVELOPMENT

1.1 FUEL MATERIAL CLADDING AND BONDING

Fuel element jacket material properties of aluminum and zirconium base alloys are being studied. Aluminum alloys X2219, X8001, and M400 appear to be promising materials, judged from static corrosion tests. Under dynamic tests, however, these alloys indicate increased corrosion rates. Considering the static condition, alloys X2219 and X8001 show good comparable corrosion resistance to water at elevated temperatures. Gaged by strength standards, X2219 is the better of the two alloys. But this strength can be achieved only through solution and aging heat treatment. On the other hand, the X8001 alloy appears to possess better corrosion resistance when tested under dynamic conditions and elevated temperatures.

Following is a chart showing the average uniform corrosion of X8001 with variation of flow rate and water conditions at two different temperatures, as reported by government laboratories.

Temperature	Water Flow Rate feet/sec.	Average Penetration in mils/yr.		
		pH 7	pH5.5 H2SO4	pH 5.5 H3PO4
500 F	18	6.3	4.3	--
	7	2	2.5	--
600 F	18	12(17)*	9.5	4.3
	7	6.5	5.8	2.0

* 17/mil/yr for X2219 (6% Cu) is a fairly typical comparison showing somewhat higher corrosion rate for this alloy.

A difference in behavior could exist between corrosion specimens under corrosion loop conditions and fuel element cladding subjected to reactor conditions. However, data from various laboratories seems promising, especially that recorded from tests conducted at low pH with phosphoric acid. It is noteworthy that both Argonne National Laboratory and Hanford Atomic Products Operations Laboratory have initiated programs to determine the effect of phosphoric acid on structural materials such as Type 304 stainless steel.

Corrosion Test Loop

All equipment for the corrosion test loops with the exception of the conductivity recorder has been received and will be installed during the first part of July. The completed loops prior to insulating are illustrated in Figs. 1, 2, and 3. Also, an autoclave for static and semi-static corrosion testing has been preconditioned to prepare it for screening tests of materials. In addition, a demineralizer of laboratory size has been obtained to provide high purity water for corrosion tests. Some 1200 pounds per year of total corrosion products are expected in reactor water utilizing aluminum fuel elements, according to calculations. A break-down of the many elements present in these corrosion products was supplied to the physics group for shielding calculations on filters and ion exchangers.

The corrosion resistance of various stainless steel alloys is under study with special attention to stress corrosion factors. A comparison of corrosion properties was prepared from information available on 304, 405, and 410 stainless steel alloys.

Boiler Fuel Element Cladding

The double-alloy tube concept – X2219 or 5083 clad externally with X8001 – for cladding the boiler fuel element has received considerable study. Such a tube would have the strength necessary to resist internal pressure from gases generated in the fission process, as well as corrosion resistance to the external flowing water and steam mixtures. Proposed designs have been discussed with aluminum producers and metal fabrication experts. It appears that the manufacture of tubing to meet core fuel element requirements cannot be accomplished with commercial methods now in practice. As a result, a laboratory scale development program would be necessary to establish feasibility and practicability of any proposed techniques for tube manufacture. However, it appears that the cost of manufacture of the double wall tube may decrease the economic advantage of aluminum over the single wall zirconium cladding.

The single alloy tube of corrosion resistant material can be used satisfactorily if design features such as void spaces on the ends of the fuel rods are incorporated to avoid pressure buildup of fission products.

Fuel Element Design

In order to vary fuel loading over the core length it has been proposed that the fuel tube be of a "step-down" configuration. A conceptual alternate to the four bundle design given in Fig. 1.8 of ACNP-5803 is shown in Fig. 4 and incorporates the step-down rod. Such a design would decrease the number of end-closure welds markedly and appears to present no economic disadvantage because of the several pellet sizes needed. The step-down operation does not offer any tube manufacturing problems but may increase cladding costs.

The end closure proposed for the fuel rods in Fig. 4 would be accomplished by a metal spinning operation. End closures have been attempted and satisfactorily produced with 3003 in a variety of configurations using this technique in conjunction with a small spot fusion weld to insure sealing of any unbonded material. Other end-closure welding and brazing techniques have been attempted and show promise. Spinning appears to be the most desirable because a minimum amount of material and operations are involved to produce a satisfactory closure. Spinning will be recommended for use on the critical facility boiler fuel elements.

Diffusion bonding experiments of zirconium to austenitic stainless steel with tin have been conducted under pressure applications at temperatures of 1000 F and 1200 F. Visual examinations showed no bond formation. Micro-structural examinations are presently underway on the test specimens to determine whether any of the tin coating diffused into either metal. Additional experiments are presently being conducted and results will be reported in subsequent quarterly reports.

An agreement on scope of work for uniformly coating uranium dioxide particles with zirconium was reached with The Martin Company and a development contract was signed. Studies on the vapor phase deposition of zirconium on uranium dioxide will start in the next quarter.

1.2 HEAT TRANSFER AND FLUID FLOW INVESTIGATIONS

Heat Transfer Loop

All the equipment for the loop, with the exception of the test section, has been specified and inquiries have been sent out to various manufacturers for quotation. Piping contractors have been contacted to design, supply, and install the loop supports and components. The order for the circulating pump was placed with the Allis-Chalmers Nuclear Pump Section. The Unipolar Generator-Motor Set is being built by the Allis-Chalmers Motor-Generator Department. No other orders have been placed at this writing, but it is expected that all the equipment will be ordered by August 15. The boiling test section has been designed such that a full fuel element may be tested. This test section is flexible enough to enable testing of from one to 81 simulated fuel rods.

Fuel Element Holddown

The hydraulic holddown tests gave promise that the fuel elements could be held down without external restraint. However, the degree of uncertainty involved with the two-phase pressure drop correlations now available has encouraged the investigation of mechanically holding the element in place.

Methods for holding down the fuel elements from the bottom have received particular emphasis. One method that shows promise involves the use of a quick release pin to secure the element to the grid plate. Two of these pins have been ordered and a test is planned by modifying the dummy fuel element manufactured for the hydraulic holddown tests.

1.3 FABRICATION OF FUEL FOR THE CRITICAL FACILITY

The boiler and superheater elements for the critical facility have been completely designed and detailed. A small model of the boiler fuel element assembly was built to ascertain the easiest method of loading the element (See Fig. 5). Three clipping arrangements were tried, with the one shown in the illustration proving the best. The superheater element does not present any problem insofar as assembling is concerned. One person will be able to assemble a superheater element in five minutes.

Specifications for manufacture of the fuel elements have been written and are being duplicated. Request for bids will go out in the next quarter.

1.4 FUEL ELEMENT MANUFACTURING, RESEARCH AND DEVELOPMENT

A series of calculations still are in progress to determine optimum fuel pin diameter and number in the boiler fuel element. The box size is fixed at 4.993, as shown in Fig. 6. The effect of adding more fuel elements is also being investigated. The effect of these parameters on fission gas pressure, fabrication, pressure drop and maximum fuel temperature is considered.

Calculations of fission gas pressure within the fuel element have been refined since the last progress report. The results for one fuel configuration are shown in Figure 7. Based on these results, it is estimated that 3 inches void length in each fuel pin will be required to maintain reasonably low stresses in the aluminum alloy cladding material. This does not appear to be an excessive amount to add to the length of each fuel rod.

The following characteristics represent the latest reference boiler fuel element:

Box Size	4.993" x 4.993"
Fuel Rods per Box	9 x 9 = 81
Max. Rod Diameter	0.338 inches
Number of boxes	96
Cladding Thickness	0.040 inches
Pressure Drop through Core	15 psi
Maximum Fuel Temperature	4100 F
Average Heat Flux	<u>110,000 Btu</u> Hr - Ft ²
Maximum Heat Flux	<u>272,000 Btu</u> Hr - Ft ²
Fuel Density	0.95 theoretical

One of the current designs considers reducing the fuel loading in the upper portion of the core by reducing fuel rod diameter in two successive steps. This requires cladding tubing of a stepped longitudinal cross-section, discussed previously.

The use of Zircaloy-2 cladding, with heavier fuel loading for longer core life, is also being studied. This fuel element may yet prove to be more economically or technically feasible than the present reference design employing aluminum cladding. The high corrosion resistance of Zircaloy-2 will permit operation of a fuel assembly for two to three years. This increased irradiation time would require a higher fuel loading to provide the required excess reactivity for burnup.

The most reasonable way of increasing the fuel loading appears to be achieved by using the maximum complement of 148 fuel elements (Fig. 6). This arrangement decreases pressure drop, central fuel temperature and fission gas pressure.

Continued reduction in Zircaloy tube prices and the development of lower cost fabrication techniques will make this material much more attractive for use in CRBR.

1.5 NUCLEAR HANDLING TOOLS, COFFINS, SHIPPING CONTAINERS

Fuel storage in the turbine room still appears promising. Several different designs for the handling tool and the fuel element latching mechanism are being studied, along with methods for decelerating and stopping the fuel element at the end of the transfer chute.

2. REACTOR MECHANICAL STUDIES

2.1 VESSELS AND STRUCTURES

The reactor vessel arrangement, based on the use of external recirculating water pumps (Fig. 2.3 of ACNP-5803) was used as the pattern for the preparation of vessel drawings suitable for preliminary bids. Drawings of the reactor vessel and the vessel cover (Figs. 8 and 9, respectively) were prepared specifically for the purpose of obtaining preliminary quotations from interested companies. These drawings along with the "Manufacturing Specification for Reactor Vessel," specification 43-058-002-01-401, were sent to suppliers for preliminary quotations. Preliminary price estimates are expected to be available some time after the middle of July. In addition, design studies on the reactor vessel are continued in the following areas:

Pipe Nozzles and Lower Head

Calculations indicate that the reinforcements around the recirculating water nozzles in the lower head region should be improved. Several schemes are now being considered to make this section easier to analyze, as employing a hemispherical head, reducing the size of the water inlet section, and using 22-inch suction pipes instead of the present 24-inch pipes. Modification of the lower vessel head should eliminate the need for extensive proof testing and theoretical analysis.

Closure Designs

A segmented ring type closure has been used as the reference design throughout the program because it is simple, reliable and theoretically easy to operate quickly. The basic design has many years of successful operation experience to its credit in high pressure heaters, flash boilers and other similar equipment. A particular design problem associated with the segmented ring type closure arises in the seal requirements. Inasmuch as the cover is not bolted in place, no initial force is available to compress a gasket. Any gasket or seal must derive its sealing ability from the pressure developed within the reactor vessel. A gasket of this type has been termed a "self-energizing seal."

Two self-energizing seals of this type are illustrated in Fig. 10. The seals are being tested for the CRBR reactor in a special autoclave (See Fig. 11). A flexible elastomer seal mounted on the test cover illustrated in Fig. 10 shows promise as the initial seal in the reactor vessel. Tests are under way to determine whether a seal of this type will perform under CRBR operating conditions for the required period of time. A second elastomer seal design which will facilitate cover assembly has been drawn and will be ordered shortly for testing. All test gaskets are nominally 12" O.D. compared to about 7½ feet required for the reactor vessel.

The tapered metallic wedge type seal (the standing ring in Fig. 10) will also be tested for sealing quality. This seal is placed between the cover and segmented rings. Pressure within the vessel forces the cover upward and the angle of the wedge forces the seal ring to expand tightly against the outer wall of the vessel. Seals of this type are normally employed on turbine throttle valves and have been used on several special designs for pressure vessels.

Another seal method has also received attention. This seal uses a torus shaped ring which is welded to flanges. One flange is bolted to the cover plate and the second flange is bolted to the pressure vessel. Metallic gaskets would be employed beneath both gaskets as a seal. Since the cover would be held down securely by the segmented ring, the bolting requirements would be based only on the gasket needs.

Any closure design now under consideration will require a secondary seal between the cool water of the shield pool and the reactor vessel to prevent cool water impingement on the hot surface of the pressure vessel. Ideally this seal could be formed by venting the areas beneath the shield pool. The water pressure above the seal could thus be employed to maintain a tight seal, much as the stopper works in a present day laundry tub. Such a seal with a backup drain system could prove to be the best and most economical method of isolating the reactor vessel from the joint between the vessel and shield pool.

External Piping Analysis

An analysis of the external recirculating water piping has been in progress. The Kellogg method was used as the basis and preliminary results have been obtained. The numbers indicate that the maximum stress expected in the piping due to thermal expansion, plus the friction force expected in the sliding pump supports, will not exceed 12,000 psi. The maximum occurs at the mitered suction elbow of the pump. The average stresses in the piping will be about 5000 psi or less.

The effect of moments and forces on the vessel due to the piping is now being considered and results should be available shortly. To date only a single, complete pipe loop has been analyzed, selected on the "worst case" basis.

The piping wall would consist of 5/8-inch carbon steel with a 1/8-inch inside cladding of Type 304 stainless steel. Clad piping is considered superior to the stainless steel for the following reasons:

1. Greater economy.
2. Reduces the thermal expansion of the piping system.
3. Virtually eliminates any differential thermal expansions which might result from using solid stainless steel piping with a clad reactor vessel.
4. Reduces the problem of stress corrosion anticipated with the use of solid stainless steel.

Support Structure

As an alternate to a continuous vessel support skirt, a system of five support lugs was designed. The recirculation piping locations beneath the reactor vessel permit positioning five main support columns for the vessel. Preliminary calculations have been made on these supports columns without accounting for the forces exerted by the external piping. Additional study is required on the lugs which have some advantage over the continuous skirt in that thermal stresses due to an axial temperature gradient are eliminated.

Superheater Mechanical Design Study

The superheater occupies the central region of the core as shown in Fig. 6. The superheated steam manifold, located below the core, and the fuel containing superheater tubes are insulated from the vessel water by a stagnant steam film. All of the superheated steam passages within the vessel are of double wall construction to provide the insulating layer of stagnant steam.

A complete design analysis of the superheater assembly was recently initiated. At the completion of the study, it is hoped that the physical arrangement and detail design will be defined with a fair degree of completeness. The study will cover the following general areas:

1. General system layout
2. Complete stress analysis, including:
 - a) Mechanical stresses under normal and abnormal operating conditions
 - b) Thermal stresses during steady state and transient conditions
3. Study of relative thermal expansions and their possible effect on alignment and dynamic response of the system as a whole
4. A lifetime estimate of individual components considering:
 - a) Fatigue due to mechanical and thermal cycling
 - b) Radiation damage
 - c) Corrosion
 - d) Erosion
5. Materials selection
6. Fabrication studies
7. Flow studies
8. Entrainment separator requirements
9. Heat transfer and insulation studies
10. Evaluation of required test programs

An analysis of the superheater grid plate was recently begun. A computer program has been set up to determine the temperature distribution in the grid plate and to evaluate thermal distortions. When this study is complete, the results will be used to design a compatible mechanical arrangement so the grid plate deflections will be at a minimum under all operating conditions.

A literature survey of promising entrainment separator configurations has been started. Some manufacturers have been contacted in regard to the entrainment problem.

A fabrication test to investigate methods of welding thin-walled tubes into tube sheets is being set up. Some preliminary results should be forthcoming in the near future.

Corrosion Resistant Vessel Coating

The first metal sprayed coating corrosion test, initiated during the double vessel study, has been terminated. The object of this test was to find a vessel coating which would prevent corrosion for 20 years on a double vessel submerged in 150 F water saturated with oxygen.

Metal-sprayed coatings are porous and it is necessary for the coating to be anodic with respect to the steel vessel in order to prevent corrosion of the steel. Since zinc tends to become cathodic to the steel at about 140 F, aluminum was selected as the most desirable coating. Another fact in its favor was the claim that a .010" to .012" thick aluminum coating would be non-porous. Any oxide formation would plug the pores and reduce overall porosity.

A total of 10 samples were obtained from two sources. This included six various aluminum samples plus one each of nickel, zinc, and carbon steel flashing coatings, and a bare steel plate. Some aluminum-coated samples were given a thin initial spray coat of carbon steel (flash coat) for better adherence of the aluminum.

A humidity cabinet maintaining a temperature of 100F, and 100% relative humidity was selected for the test because it closely approached vessel conditions. Only a relatively short time is available to get results. Therefore, it was decided that if negative results were obtained in one year the coating would be worthless for this application. The test was planned as a visual inspection for corrosion inasmuch as the primary purpose was to determine if the coatings were non-porous. If any rusting or loosening of the coating occurs, it would be possible to see it.

The various samples are described and shown in Figs. 12, 13, and 14, before placing in the humidity cabinet.

When the double vessel study was completed, it was decided not to use this type of vessel, and, consequently, the corrosion test was terminated. After 3200 hours exposure in the humidity cabinet, the samples appeared as shown in Figs. 15, 16, and 17.

Test results showed rust spots after only 24 hours of exposure on the nickel-coated (1), flash-coated (5), and bare steel (4) samples. The zinc-coated sample (2) had developed its characteristic white oxide, and the aluminum-coated sample (3) became dark grey. The remaining samples show no visible change. The changes which took place after the first 24 hours were very gradual.

When the test was terminated, the nickel-coated sample (1) had rust spots on all surfaces. The zinc-coated sample (2) was covered with white oxide. A small rust spot was also present on one edge. The aluminum-coated sample (3) was dark grey with many white spots and several rust spots. The steel (4) and steel-coated (5) samples were extremely rusty. The remaining aluminum-coated samples showed only slight discoloration and one small isolated rust spot on the edge of sample (7).

This test showed no negative results for the recommended coating samples (9) and (10). However, no positive conclusion could be drawn from such a short test.

Because present reference vessel design will be operated at 490 F in air, it was decided to run a second coating corrosion test. The same samples have been placed in an electric furnace at 500 F and are removed to the air from time to time and allowed to cool. Visual examination is again being used and the test will be run as long as possible, or until negative results are obtained.

2.2 RECIRCULATION PUMPS

The purpose of the recirculation pumps for the CRBR reactor is to force the core coolant and moderator through the core at a higher rate than could be accomplished with natural circulation, thereby removing more heat from the core with a lower exit void fraction.

As stated in the last quarterly report, the reference design utilizes three external low speed pumps with 24" suction and 22" discharge piping. Because this pump system requires much less NPSII than the previous internal pumps, it has been possible to increase the reactor feed water temperature from 300 F to 360 F, thereby substantially increasing overall plant efficiency.

Preliminary layouts of the pump have been made, showing a vertical pump with the impeller overhung from the motor shaft. There will be an A-C fluid piston bearing located at the suction wearing ring which will limit the shaft deflection and increase the critical speed well above the operating speed.

Cost estimates are being prepared for the pump design described above. A rough estimate indicated that the pumps would cost more than had been expected. Therefore, an alternate arrangement involving only two pumps is being studied. By eliminating one pump, the price of the pump and loop would be saved. But the power loss would increase as a result of higher velocities in the pipes. If larger pipes are used the cost would increase in addition to introducing some difficulties in vessel design. The required NPSH would be increased to a point where operation of the two-pump system with the 24" - 22" pipe loop without the subcooling effect of the reactor feedwater would be impossible. With the larger pipes it would not be advisable because of cavitation.

2.3 STEAM SEPARATOR

The major work done in the last three months on steam separators has been performed with a test apparatus using 3" diameter models of the proposed steam separator design. The models are built hydraulically to scale and provide a quick and effective method of testing improvements to design. The apparatus also provides visual indication of simulated reactor conditions at the outlet of the core and downcomer region.

Figure 18 shows the test tank without the separators. The water-air mixture enters through the bottom and flows up through the center tube. The tube simulates the reactor core. The screen prevents excessive splashing and re-entrainment.

Figure 19 shows the test apparatus in operation without separators. The water-air mixture is being discharged upward out of the "core" and the downward velocity around the core is approximately one ft/sec. The test gives an indication of the bubble carryover without separators. The downward velocity in the CRBR without separators would be 2.54 ft/sec which would give considerably more carryover.

Figure 20 shows the separators in place and water flowing at an equivalent reactor rate but without air flow. The vortex can be discerned upon close examination.

Figure 21 shows the separators in operation separating air bubbles from water. The spots on the lower section of the tank are water droplets on the outside of the tank.

Work is continuing on the separators to improve operation under maximum reactor power conditions. Additional descriptive information on the separators is given in previous quarterly reports.

2.4 CONTROL RODS, GUIDE TUBES AND CONTROL ROD DRIVES

Two alternate drive motor units for the control rod drive mechanisms are shown in Fig. 22 and Fig. 23.

Figure 22 shows an induction motor with spur gearbox and brake. The brake will be used to stop the motor very rapidly, thereby preventing coasting of the control rod, and also holding the rod at any position of the stroke. Also shown are the fail-safe magnetic scram clutch and limit switches.

Figure 23 shows a servomotor mounted on a double-worm gearbox. It is anticipated that the low inertia rotor of the servomotor will stop very rapidly because of the load and will not require a brake to prevent coasting of the rod. The worm gearbox is self-locking; therefore it will hold the rod at any position of the stroke. Again, the clutch and limit switches are shown.

Position indicators are not shown in either Fig. 22 or Fig. 23 since the type and quantity have not yet been decided.

The control rod latch design and housing mounting dashpot design are still under study, as is a method for rotating the rods and rod latch externally without having to open the rack housing.

The specification for the critical facility control rod drives has been completed and requests to quote have been sent to potential bidders.

Studies on materials for control rods, guide tubes and control rod drives are found in Section 10, Materials and Metallurgy.

3. NUCLEAR ANALYSIS

3.1 REACTOR PHYSICS (STATICS)

Core Power Distribution

The power distribution has been determined for the reference core (see Fig. 6) utilizing a two dimensional reactor code for the IBM 704. Included in the calculation were the expected distribution of operating steam voids in the boiler core, the non-uniform axial distribution in the boiler core, and the internal nuclear superheater.

The axial distribution of fuel in the boiling core was taken to be 81 rods per assembly in the bottom half of the core, 70 rods per assembly in the upper middle quarter, and 46 rods per assembly in the upper core quarter. This axial fuel distribution was found to give good flattening of the power per tube over the core height. A uniform axial fuel loading depresses the flux and power in the upper, more voided regions, and results in poor power distribution.

The average axial power distribution in the superheater and boiler core regions is shown in Figs. 24 and 25, respectively. In the nuclear superheater, the power distribution is essentially a cosine shape since the fuel loading is even. The asymmetry is caused by the influence of the flux distribution in the boiling water core, which, it is seen, is peaked in the lower core region. In Fig. 25, the top curve represents the thermal neutron flux or power per unit fuel volume, and the lower curve represents the power per unit total volume. The curves are normalized to the power in the lower core region.

The average radial power distribution is shown in Fig. 26. Across the superheater, the average power is quite uniform. Across the boiler core, the power distribution assumes a J_0 shape with peaking near the reflector. The following table summarizes the maximum to average power ratios.

Maximum to Average Power per Tube Ratios

	<u>Pm/Pa (Radial)</u>	<u>Pm/Pa (Axial)</u>	<u>Pm/Pa (Total)</u>
Superheater	1.00	1.40	1.40
Boiler	1.31	1.21	1.58

Disturbances of this distribution due to control rod insertion and peaking around water channels are not included. Also, the changes caused by burnout have not yet been considered.

Control Requirements - Reactivity Change over Cycle

A study of the reactivity change over the exposure cycle is being made to determine the control requirements of the system. The control system requirements can be stated as follows:

- (1) Control to override the negative reactivity effects of increase in temperature and formation of steam voids during rise to full power.
- (2) Control to override the buildup of saturable fission product poisons xenon-135 and samarium-149 during first few days of power operation.
- (3) Control to compensate for the burnout of fuel and the buildup of non-saturating fission products and the conversion products of U-238 (plutonium-239, 240 and 241).

The reactivity effect of operating temperature and steam voids has been determined to be $-3.9\% \Delta k/k$. This is the reactivity change found by assuming the initial conditions to be cold water in the boiler and the superheater (with voided steam passages in the super -

heater), and the final conditions to be hot water and an operating steam void distribution in the boiler and hot water in the superheater (with steam in the steam passages).

The buildup of xenon-135 and samarium-149 takes place over the first three days and twenty days of power operation, respectively. These isotopes are then approximately at equilibrium and have a combined reactivity effect of about $-3.1\% \Delta k/k$. The reactivity effect of these isotopes during reactor power transients is not considered here since it represents only a short term demand on the control system.

The burnout of fuel and the buildup of the non-saturating fission product poisons and converted isotopes represents a long term reactivity change over the operating cycle. This change, over an eight-month period at 0.8 load factor, is about $-10\% \Delta k/k$. The eight-month exposure period was chosen for this study since it is intended that the inner third of the boiler core will be replaced every four months. At the same time, the less exposed two thirds of the core is moved inward to regions of higher flux and exposure. This represents an average fuel exposure of about eight months for each year's operation.

The combination of these effects represents a total control requirement of about $-17\% \Delta k/k$ from start to finish of a cycle. Since the total reactivity worth of the control system as determined from preliminary calculations is of the same order as the control requirements, it is desirable to incorporate a burnable poison into the system to provide a shutdown margin of control. A plot of excess reactivity versus operating time is shown in Fig. 27 for the core operating with and without a burnable poison. The sharp initial decreases in reactivity are due to the first two effects described. The latter portion of the upper curve falls off until no excess reactivity exists at the end of the cycle. The lower curve shows that in the first few months of operation the excess reactivity increases because the burnable poison is diminishing more rapidly than the fuel. At this time, the effect of fuel burnout becomes predominant and the excess reactivity decreases to zero. At no time in the cycle can the excess reactivity exceed the initial unpoisoned value.

The reactivity effects described above represent an average exposure distribution in the core. The effects of non-uniform power distribution and the rotation of fuel are presently being studied.

Reactivity Required – Fuel Loading and Enrichment

From the reactivity requirements discussed in the preceding section, it has been determined that approximately $17\% \Delta k/k$ excess reactivity will be needed at the start of the exposure cycle in an unpoisoned reactor. To meet this requirement, the fuel loading tabulated in the following table has been determined.

Fuel Loadings – Superheater and Boiler Core Regions

	<u>Number Elements</u>	<u>Number Per Element</u>	<u>Total U-235</u>	<u>Total U</u>	<u>Enrichment</u>
Superheater	428	2 tubes	28 Kg	105 Kg	26.6%
Boiler	96	69 pins (avg.)	117 Kg	6500 Kg	1.8%

Control Rod Effectiveness

A survey of methods of control rod calculation has been made to determine the best approach to multiple rod evaluation. The method of Hurwitz and Roe (KAPL-1336) has been selected as the most applicable. This method permits the rod or group of rods in an associated region to be represented by an equivalent absorption area. The neutron source in that region is modified in proportion to this absorption area. The reactivity of the system can be calculated by available one- and two-dimensional diffusion theory computer codes. This method affords considerable flexibility in that treatment of partial rod insertion is possible and weighting effects due to a variable neutron flux and non-uniform fuel loadings are automatically included.

Calculations have been made on EBWR as a sample problem, using the absorption area method. The results obtained show good agreement with values quoted for EBWR. The same case was calculated as a central rod and as a central rod with a concentric ring of rods by the boundary value method of Murray and Niestlie (Nucleonics, February 1955). Excellent agreement was obtained. The case of the central rod also agreed with the perturbation theory approach, as outlined by W. H. Arnold (WCAP-380). These latter methods are inadequate for calculating the CRBR but are useful in evaluating the absorption area approach. These preliminary calculations were made on the basis of thermal absorption by a totally black rod. Closer attention is being given to greyneess of rods and epithermal absorption.

Considerations such as the proposed rod geometry, non-uniformity in fuel loading and the present status of control rod theory suggest that the absorption area method in conjunction with the diffusion theory computer codes affords the most practical scheme for control rod evaluation in this reactor, and, therefore, represents the current effort.

Temperature Coefficients

There is a prompt negative temperature coefficient in the fuel caused by the Doppler broadening of the Uranium-238 resonance lines. A search was made of the available recent literature for consistent values of the Doppler coefficient of reactivity for systems comparable to CRBR. Calculations are also being made on the basis of a theoretical model using the known resonance absorption lines of Uranium-238 together with scattering by UO_2 .

The homogeneous resonance absorption integral has been calculated for fuel temperatures of 68 F and 750 F. These results are being compared with experimental work by M. V. Davis at Hanford with uranium metal and oxide under different conditions.

Flooding of Highly Enriched Superheater Elements

Calculations have been made to determine the reactivity change caused by the flooding of the steam passages in the enriched superheater element. Since the enrichment of the superheater fuel is high (20% or greater), the reactivity effects upon flooding because of resonance escape probability and the fast fission effect will be negligible. Local cell calculations (P_3) have shown that the change in thermal utilization will be small ($-0.6\% \Delta k/k$ for flooding). The major reactivity effect will be the decreased axial leakage. This contribution to the superheater reactivity is about $+2.9\% \Delta k/k$, giving a net effect of $+2.3\% \Delta k/k$ for complete superheater flooding. The effect on the total core will be substantially less since the superheater produces only about 1/5 of the total core power. The previous progress report listed the reactivity change for the slightly enriched superheater elements as $+1.3\% \Delta k/k$.

Nuclear Constants

Several computational methods are presently being compared for adequacy and accurateness in reactor calculations. Cross-section files for the materials used in CRBR have been set up for 3, 7, and 19 neutron energy groups. These cross-sections were obtained from the standard measurements (BNL 325) and grouped in convenient lethargy intervals.

By the use of the LOST program developed at KAPL for the IBM 650, along with a 170 point cross-section file, a set of nuclear cross-sections may be obtained for an arbitrary number of energy groups. The results of LOST are being compared with the 3, 7, and 19 group cross-sections that have been assembled.

In addition, two group methods which utilize experimentally measured values of ρ , ϵ , τ , and D are being used. It is intended that these methods will be compared by calculating known reactor configurations.

Reactor Shielding

Since the choosing of a reference core, two dimensional plots of flux are being obtained for various points in the primary shield. These calculations are being performed for the fast neutron flux, the thermal flux and the gamma flux in five energy groups. Various types of shielding materials and various combinations of materials will be evaluated to arrive at the optimum shield design.

From the above flux plots, heating in the vessel wall and primary shield can be calculated. Because of the large amount of water surrounding the core, the maximum heat generation in the core vessel was found to be 5000 Btu/ft.³Hr. This value is quite nominal, and, therefore, eliminates the need for a thermal shield. The total heat deposited in the primary shield was found to be approximately 14 kilowatts.

A shielding computer program for the IBM 704 has been obtained and is presently being adapted for our shielding calculations. The program uses cylindrical source geometry. It has the following capabilities: It can handle up to 100 different regions; up to 20 different materials in the regions. The program will compute neutron flux and dose rates at any vertical or radial position. The computer program will be used for a check of work already completed, and for the calculation of flux and dose rates vertically and radially from the core for various shielding configurations and materials. It will also be used for shielding and dose rates around the plant.

In preparation for determining dose rates around the plant, the N^{16} concentration in steam and its production rate are being calculated. The calculational method assumes that N^{16} retains the same molecular form it had prior to activation; that one pound of steam carries off an amount equal to the amount in one pound of reactor water; and that the N^{16} diffuses completely and reaches equilibrium concentration before being discharged to the steam line. The calculational method was applied to EBWR. The calculated result multiplied by a constant agreed quite closely to the measured concentration (see ANL Test Report No. 34). This constant, probably due to the sweeping action of the steam, was then applied to the results.

With the above N^{16} concentration and the shielding computer program, a study of the dose rates and shielding requirement is being determined for the turbine building of the Pathfinder plant. Different shielding materials and various combinations of these materials will be considered in order to arrive at the optimum type of shielding for the plant.

3.2 REACTOR AND SYSTEM DYNAMIC ANALYSIS

The analysis to determine stability of the reactor system at full power is underway. The reactor system includes the boiler and superheater cores, recirculating pumps, drives, and controls, steam hold-up chambers and piping, and reactivity poison effects. The present studies are devoted toward the safety aspects of reactor operation. The results are to be included in the Preliminary Safeguards Report for the Pathfinder Plant.

After the necessary data for the Safeguards Report has been obtained, the entire system will be simulated in much greater detail with the aid of an analog computer. The effects of equipment failure, operator errors, and abnormal operation will be determined from this model. A mock-up of the operating console in conjunction with the analog simulator is being considered. This should prove to be a valuable aid in training the reactor operators.

One of the major purposes of the dynamic studies is to obtain the philosophy and requirements for the control scheme. The regulating signals could be any combination of steam flow, turbine inlet pressure and temperature, and electrical load. The regulators could be any combination of control rods, recirculation pumps, and inherent nuclear feedback properties of the reactor.

3.3 CRITICAL EXPERIMENTS

Safeguards Analysis

The Preliminary Safeguards Report for the CRBR critical facility, ACNP-5809, has been completed and submitted to the AEC along with an application for a construction permit. The report contains a somewhat detailed physical description of the facility; an operating procedure and experimental program; and a safety analysis of the critical assembly.

Specifications

Specifications have been written and sent to possible suppliers for bidding on the control rod drives, nuclear instrumentation, neutron source drive and boiler core fuel rods for the critical facility. During the next quarter, orders should be placed for the above items.

Mechanical Design

Most of the mechanical design of the critical assembly and its accessories has been completed. The boiler fuel element is shown in Fig. 28. It is made up of four bundles of fuel tubes (see Fig. 29) located and held by tube spacers, Fig. 30. The tubes are grouped in sections consisting of nine tubes and two spacers. Nine sections comprise one fuel bundle. Four bundles are assembled in the fuel element box (see Fig. 31) to make up the final assembly. The voids caused by boiling in the Pathfinder Reactor will be mocked up in the critical facility using expanded plastic pieces of low density. A typical void simulator is shown in Fig. 32. Holes in the simulator will be used to vary the void fraction throughout the critical assembly. A cross-section of the boiler fuel element with void simulators in place is illustrated in Fig. 33. The fuel element will be loaded by alternating void simulators and the nine tube fuel sections.

Variations in the average enrichment can be made by using tubes of different enrichments. Variations in fuel to water ratios, fuel tube diameter and cladding materials can be made by using new tubes and spacers in the standard box.

The Critical Facility superheater fuel element is shown in Fig. 34. The fuel elements positioned in the critical assembly are shown in Fig. 35. The elements slide into and rest on the core support plate and are guided at the top by the upper support structure and the superheater tube sheet. Cruciform control rods are used and run in guide tubes bolted to

the core support plate and located at the top by the upper support structure. Electric heaters are used to control temperature and an agitator keeps temperatures uniform throughout the tank. An overflow pipe will keep the water level in the desired operating range.

The Critical Facility piping diagram is shown in Fig. 36. Normally water is pumped by the fill pump from the dump and storage tanks to the reactor tank directly above. On shutdown the water is dumped through the two dump valves to the dump tanks. The deionizer maintains high water purity. Any leakage in the pit runs to a sump from which it is pumped to the retention tank. If it is necessary to change the water, it can be pumped from the dump tanks to the retention tank. After the water has been monitored and the radioactivity level is within limits it can be pumped to a sewer.

4. CHEMISTRY

4.1 CORROSION

The corrosion characteristics of many materials considered for use in the Pathfinder plant have been carefully surveyed. These investigations constitute a search for information relative to corrosion resistance of materials under conditions to be experienced in the CRBR operation. As the available information is assembled and analyzed, it becomes apparent that certain materials must be tested for corrosion resistance under design conditions for the CRBR. Corrosion studies on various alloys of aluminum and stainless steel are discussed in Section 1.1, Fuel Material Cladding and Bonding.

4.2 RECOMBINER, REACTOR WATER PURIFICATION SYSTEM, AND ION EXCHANGER

Recombiner

To explore current developments in Recombiners, contact was made with Baker and Company of Newark, New Jersey in regard to their experiments in recombining low concentrations of hydrogen and oxygen in 825 F steam. Their data indicate very satisfactory results with the catalyst employed, but they are continuing experiments with catalyst supports and saturated steam.

Reactor Water Purification System

Three flow schemes were originally proposed for reactor water clean-up. Refer to Fig. 37 of this report and Figs. 4.1 and 4.2 of ACNP-5803.

These methods have been studied in greater detail to ascertain their relative advantage and disadvantages. On the basis of information accumulated in this study, it has been decided to proceed with a detailed design of the low pressure system with flash tank. In order to arrive at a decision at this time, it was necessary to assume that uncertainties regarding the flash tank can be worked out to full satisfaction. This will be discussed in more detail under "Flash Tank."

Prior to the decision, each system was evaluated on the basis of initial cost, expected life of equipment, space requirements, effect on overall plant design, and personnel safety. Since operating costs are essentially the same for all of the systems under study, little emphasis was placed on this factor.

The low pressure system, using the condenser-type heat exchanger, was eliminated in the early stage of the study (refer to Fig. 4.2 of ACNP-5803). The heat exchanger required was considered impractical from both the design and operation standpoint and too costly to warrant further consideration.

Preliminary design specifications were forwarded to manufacturers of the various components required in the remaining two systems. (High Pressure and Low Pressure with Flash Tank).

These specifications covered:

1. Ion Exchangers
2. Filters
3. Heat Exchangers
4. Flash Tank
5. Flash Tank Controls

No attempt was made to include the cost of valves or piping.

An analysis of the estimates received indicates the expected cost of low pressure equipment to be 60% of that for high pressure. If valves and piping were included in the analysis, they would tend to significantly decrease this percentage in favor of the low pressure system. The obvious economic advantage became the most important factor in selecting the system.

Additional advantages of a low pressure system are:

- (1) Non-dependence on additional pump to maintain flow during normal plant operation. (The high pressure system would require an extra "booster pump" to maintain sufficient differential).
- (2) Less probability of equipment leakage. (Should leakage occur, it would be less troublesome).
- (3) No radioactive water contamination of either cooling tower water or condensate resulting from an undetected heat exchanger leak. (The pressure differential causes flow in the opposite direction, or into the reactor water side).

Should leakage occur into the reactor water side, it will be cleaned up in passing through the ion exchangers and after-filters before entering the main condenser and back into the reactor.

There is one obvious disadvantage to the Low Pressure System, i.e., the difficulty of controlling the flash tank and subsequent system flow control during reactor startup and shutdown. However, preliminary cost estimates indicate that effective control can be achieved with standard equipment at relatively low cost.

Low Pressure System Design

(A) System Design (Refer to Fig. 37)

Work on the Low Pressure System is progressing on the basis of cycling reactor water continuously at a flow of 100 gpm. Since this design flow is based primarily on corrosion of aluminum-clad fuel elements, it is subject to appreciable change should some other cladding material be used.

Operating pressure of the system will normally be maintained at 30 to 40 psig. It is anticipated that automatic control of the flash tank will be accomplished by controlling steam outlet flow as a function of water level.

(B) Component Design

Flash Tank: Since the low pressure system was originally proposed, the flash tank has received considerable attention. Earlier there was some question on whether a practical design of reasonable size and cost could be achieved. Preliminary design calculations have given some assurance as well as the fact that there are at least two similar tanks now being used in other installations. Both the calculations and operating experience with similar equipment have established the flash tank's feasibility.

Preliminary design indicates that the flash tank dimensions will be approximately 3 feet in diameter and 10 to 12 feet in height. Reactor water will be throttled through first a valve and then a nozzle during normal operation. Designing the nozzle for a maximum flow of 120 gpm will place only a small part of the throttling duty on the valve, thereby prolonging its life. High velocity flashing water will be introduced tangentially on the flash tank circumference. The high initial kinetic energy of the jet issuing from the nozzle will be dissipated through friction and turbulence in a continuous forced vortex.

Moisture separation will be accomplished at the steam outlet with suitable baffles or scrubbers. A pressure relief valve will also be provided on the tank and will be bled to a relief line connected to the main condenser.

All surfaces wetted by reactor water in the flash tank, as well as all other system components and piping, will be fabricated of Type 304 stainless steel.

Location of the flash tank has not been determined. Shielding requirements based on the presence of N^{16} and corrosion products will soon be evaluated. This will form the basis for deciding whether the tank is located in or outside the containment building.

Filters

Where possible, conventional filter units will be employed. Necessary remote handling, because of concentrations of radioactive corrosion products, will probably require at least modifications on conventional units.

Cotton filtering elements are receiving the greatest amount of attention at present. However, materials such as treated paper, glass, and metallic elements are also being evaluated.

Calculations are in progress to determine radioactivity buildup on the pre-filters due to accumulation of corrosion products. These calculations will determine shielding requirements and handling methods.

Three filter manufacturers have shown interest in the problem and are currently working out design proposals.

Ion Exchanger

New high strength, high flow resins are being investigated with the objective of using these data to reduce the size of ion exchangers, and extending the life of the resin to minimize resin inventory. Also investigated are various methods for handling the resin. One method employs resin in a cartridge or series of cartridges; the other makes use of hydraulic filling and removing of resins from the vessel. Final design will be based on an optimum scheme for resin handling.

5. CONTROL

Three basic schemes for the control of reactor water recirculation flow rate are under consideration. They are (1) Step-Type Control, (2) Variable Speed Drive Control, and (3) Throttle Control.

Step-type control may be accomplished by two techniques. Each of the three pumps can be operated either full "ON" or "OFF" or a dual-speed drive motor, permitting flow rate at either full capacity or half capacity, can be used. The advantage of this scheme of control is its low cost. However, problems may develop with transients caused by sudden changes of coolant flow rate inherent with types of control that do not provide smooth, continuous flow variation.

Variable speed pump drive could be accomplished by supplying the pump drive motor with variable frequency and variable voltage. The supply generator should be driven by either a steam turbine equipped with a variable speed governor, or a motor coupled to it through an eddy-current clutch. The major advantage of this scheme is that the coolant flow can be controlled to keep the core void fraction constant regardless of power level. The disadvantage is the high cost.

The third method of control utilizes the technique of throttling the recirculating flow with butterfly valves in the pump discharge lines. Cavitation in the valves would result when the flow drops to less than three-quarters of its rated value. At reduced flow, the head loss through these valves becomes considerable. The advantage of this scheme is its low cost and simplicity of operation, although the range of operation is very limited.

A study was made of a scheme consisting of one variable speed pump, and two units with step-type control. At lower flow rates some of the pumps have insufficient NPSH to prevent cavitation. Only a few operating points would be possible and operation of the system would be rather complicated.

Further engineering studies and cost estimates will be made on these control schemes so that the optimum can be selected.

6. PLANT SAFETY ANALYSIS

Information for the Preliminary Safeguards Report is being compiled and work on safeguard analysis has been started. It is planned to submit the report to the Safeguards Committee of the AEC some time this fall.

Stability analysis of the reactor has been initiated and the results will be incorporated into the safeguards report. Pioneer Service and Engineering Company is doing structural analysis of the containment shell under conditions of various postulated accidents. Two materials for the containment shell are under consideration: (1) steel shell, and (2) prestressed reinforced concrete shell. The containment shell selected will be based on economic considerations and the result of the safety analysis.

Investigation is directed toward determining the worst accident that would result if all the safety instrumentation failed, and, in addition, a gross operating error due to gross negligence was made. This by no means implies that such an accident would happen because the possibility for the above considerations to be satisfied is virtually nil. But it will define the limiting case and will indicate the degree of inherent self-regulation of the system. The search is narrowing the possibilities. However, detailed calculations, which will be initiated shortly, will be required to determine the worst conditions.

Containment Shell

Pioneer Service and Engineering Company has undertaken a design criteria for the containment structure. An interim report has been prepared. Reactor equipment layouts were made to determine the minimum possible size of the containment shell, based on preliminary layouts of the reactor and its circulating system, provided by Allis-Chalmers. Layouts of a steel sphere, a steel cylinder, and a prestressed concrete cylinder were studied. Minimum sizes indicated for these structures were 84 feet diameter for the steel sphere, 50 feet diameter by 115 feet high for the steel cylinder, and 50 feet diameter by 95 feet high for the prestressed concrete shell.

Based on the economic comparison presented in the interim report, the steel cylindrical shell was selected for continued study. The design pressure for the shell is dependent upon selection of the maximum accident. Final selection of the maximum accident is contingent on further detailed analysis of the reactor and its auxiliaries. Work is continuing accordingly.

Some studies for the requirements for containing the results of violent or explosive failures were carried out by Pioneer Service and Engineering Company. It was found that any high speed fragments which might conceivably be released would not penetrate the radiation shielding except in some small critical areas. The worst problem is that of restraining large forces should a high pressure pipe be completely severed. Hot water from the reactor vessel would be discharged. The containment shell must be able to withstand the internal pressure developed by release of this stored energy. Calculations were made of the pressure expected from total instantaneous release of hot reactor water for a range of building volumes. Fortunately, it is believed at the present time that failures of this type will not occur.

If a primary system break of finite size should occur, some time elapses while steam is discharged. During this time some of the steam condenses on the walls and surfaces of the containment building. As a result, the maximum pressure reached is less than that expected from a total instantaneous release of hot water. Calculations of the effect of condensation were made. Maximum building pressure was related to a range of assumed sizes for a primary system break.

The possibility of condensing flashed steam by forcing it through a shield pool was also studied. In this approach, all high pressure systems are located beneath a sealed floor. In the event of a failure, pressure from escaping steam would break a barrier at the pool bottom. The large heat capacity of the pool water would be used in condensing steam; thus, the maximum building pressure would be lower than either of the previous cases. Calculations were made to relate pressure rise to a range of breaks for this arrangement.

It is necessary to provide leak-tight seals where pipes, ducts and electrical cables penetrate the wall of the containment shell. A study is under way to determine a practical method of providing these seals. An interim report on this work has been prepared.

7. FEASIBILITY STUDIES

7.1 ELIMINATION OF VAPOR CLOSURE

It is the consensus that the Pathfinder plant will have to be built with a containment shell because of insufficient time to explore in detail any alternate approach. Therefore, no appreciable work on elimination of vapor closure was done during the period covered by this report.

7.2 NUCLEAR SUPERHEATER

One more test will complete the planned series on the double wall insulating tube heat transfer tests. The data has been evaluated for the tests already finished and a report is being written. The test will be summarized in a subsequent quarterly and a separate report to be published.

An interim feasibility report on the nuclear superheater concept, ACNP-5808, has been issued. This report describes the reactor and plant and outlines the additional research and development necessary to add this feature to the Pathfinder reactor. Some discussion is also presented on the operational economics of a nuclear superheater as well as on the capital costs. Although the feasibility study of a nuclear superheater has not been completed, the following conclusions from the above report can be made with confidence:

- a) A workable nuclear superheater design can be accomplished at this time utilizing fuel elements of 20% or greater enrichment. These fuel elements have been developed to a degree of considerable reliability. The nuclear superheater is considered to be safe and reliable, and when incorporated into the center of a boiling water reactor, improves the operation of the boiling region over that which would be realized without a nuclear superheater.
- b) The fuel cost for a nuclear superheater utilizing fuel elements of 20% or greater enrichment would be slightly higher than those for a conventional fossil-fueled superheater. However, less than 20% of the total heat is produced in the nuclear superheater; therefore, the overall economics are not affected appreciably.
- c) The probability of lowering the nuclear superheater fuel cost to nearly one-half the cost of conventional fuels in the future by utilizing low enrichment fuels shows excellent promise.

Fuel Element Studies

Literature surveys on the pre- and post-irradiation properties of UO_2 and UO_2 -stainless steel cermets were continued. Preliminary data indicates that UO_2 dispersed in a stainless steel matrix exhibited good dimensional stability at some elevated temperatures. The data examined to date, however, does not contain sufficient detail to make a fair evaluation.

Startup and Shutdown Studies

During the past quarter particular attention has been given to estimating the performance of the nuclear superheater concept during periods of reactor startup and shutdown. A calculation has been made to determine the power dissipated by a fuel channel when fuel tem-

perature is limited to 1200 F and no steam flow occurs. Conduction and radiation through stagnant steam are the only mechanisms of heat transfer assumed to prevail. The resultant temperature distribution is shown in Fig. 38. Under such conditions, 1% of full load power can be dissipated.

As part of the shutdown cycle, consideration might be given to flooding the superheater region with saturated water at 489 F. Preliminary calculations have been made to determine the amount of shock imposed upon the fuel elements under such conditions. Assuming film boiling to be the only mode of heat transfer, the fuel temperature as a function of time after flooding is given in Fig. 39. From this figure, it appears that the maximum shock temperature change is about 500 F. From data available for Type 347 stainless steel, it appears that a fuel tube could stand approximately 8000 such shocks. This is far in excess of the anticipated number of shutdowns in the life of a fuel element.

A calculation of maximum thermal deformation of a fuel element under normal operating conditions results in a requirement that the fuel cladding have a minimum elongation of 0.3% at rupture. This condition is readily met by any of the austenitic stainless steels.

Preliminary Reactor Startup Procedure with Nuclear Superheater

The CRBR plant can be started using the following procedure. The procedure is based on using the nuclear superheater and assumes the following conditions:

- (1) Vessel cover off
- (2) Water in vessel and shield at 150 F
- (3) One-third of core is new

Startup Procedure:

- (1) Replace cover
- (2) Test vessel hydrostatically
- (3) Make control rod mechanical test
- (4) Check instrumentation
- (5) Stop return of cleanup water to reactor (bypass to condenser)
- (6) Bring reactor to criticality
- (7) Increase power to 9,150 tkw

When pressure reaches 100 psig:

- (8) Shut down reactor
- (9) Open valve to 4-inch discharge line to drain superheater.
- (10) Bring reactor power back to 9,150 tkw to raise pressure and temperature to operating conditions
- (11) Start turbine warmup
- (12) Place plant on line

Preliminary Shutdown Procedure with Nuclear Superheater

The following is a shutdown procedure for the CRBR plant, listing the major steps to be followed when it is desired to completely shut down the reactor for major operations such as refueling. It is assumed that the reactor has been operating at full power (200 mw) for sufficient time to have allowed fission product decay heat to build up to its maximum level. The turbine is assumed to be operating at 66 mw. A shutdown with the nuclear superheater requires a few precautions that are not present with the non-superheating boiling reactor. The major precaution that must be considered is the cooling of the superheater fuel elements. Since these are not submerged in water upon shutdown, they must be cooled by the surrounding steam. The amount of steam flow through the superheater required to maintain fuel element temperature below a desired point is dependent upon the power output, including decay heat of the superheater core. It has been estimated that if the total power of the reactor is below 1% of the rated point, the superheater can be kept cool with no steam flow, providing the reactor water temperature is below 327 F, i.e., the saturation temperature at 100 psig.

Shutdown Procedure

- (1) Reduce turbine generator load to zero in increments normal to unit of this size (total time, one hour). Reactor power is lowered by automatic control in step with the turbine load reduction.
- (2) Make reactor sub-critical by inserting control rods to full-in position.
- (3) Open steam bypass valve to condenser
- (4) Open 4-inch startup line valve to open position
- (5) Close main steam line valve.

The temperature of the reactor vessel is dropped at the rate of 200 F per hour by automatic control of the 4-inch steam line valve opening. The boiler feed pump bypass will be set to fill reactor steam dome at the rate of 300 gpm. When the reactor is full, water will flow through the 4-inch line and the reactor temperature will be reduced to 212 F at the rate of 200 F per hour.

The vessel cover can be removed and the condenser and pumps shut down. Shutdown cooling will be continued by circulating water to the condenser.

8. STEAM CYCLE

8.1 CYCLE ANALYSIS

Several additions and modifications have been incorporated in the basic steam cycle and its auxiliaries since report ACNP-5703 was issued. In general the changes represent improvements in the cycle as well as the results of more detailed analysis and calculation.

Cooling Tower Evaluations

To alleviate the potential problems associated with the water supply at the Pathfinder site, consideration has been given to the use of evaporative coolers to replace conventional cooling towers. The initial concern and subsequent investigation were prompted by the fact that the water available for cooling service at the site is expected to be both of poor quality and scarce during certain months of the year. An evaporative cooler would minimize this problem by requiring less make-up water than a conventional cooling tower.

The economic study of the two systems was based on a ten-year period and the following factors:

1. Initial cost (installed)
2. Anticipated maintenance cost
3. Water treatment cost
4. Fan and pumping power cost
5. Make-up water requirements with respect to seasonal supply
6. Expected life of equipment

While it is recognized that evaporative cooler installations have proved practical elsewhere because of the relatively low temperature cooling requirements and relatively high design wet bulb temperature, no economic advantage would be realized with an evaporative cooler at this site. As a result of this study, no further work on the evaporative cooler is contemplated at this time.

Regenerative Feedwater Heating Cycle

As noted in the previous Quarterly Progress Report, the location of the recirculation pumps was changed from inside the reactor to outside the reactor vessel, permitting a change in pump design that substantially reduced the NPSH requirements. This was a marked improvement not only from the recirculation pump aspect but also from a standpoint of cycle efficiency. The earlier design limited feedwater temperature to 300 F in order to produce adequate NPSH for the pumps through subcooling the reactor water.

Because the present pump design removes the previous limitation on feedwater temperature, the final temperature has been consequently increased from 300 F to 360 F. The higher feedwater temperature suggests changing the number of regenerative feedwater heaters from three to four. While the added equipment increases capital cost, the operating cost saving realized from the improved cycle efficiency justifies the change.

Figure 40 reflects the feedwater heater cycle change, the reactor water purification cycle change described under Section 4.2, and the added steam line required for reactor start-up and shutdown as explained in Section 7.2.

8.2 VALVES AND PIPING

Design activity under this category centered on the recirculation pump piping system covered in Section 2.1 and 10 of this report.

Present plans call for butterfly valves in the recirculation pipe loops. Preliminary cost estimates for motor-operated butterfly valves have been received and study of operational limitations of the valves has been completed.

9. PLANT LAYOUT

The plant has been tentatively located on the site given in the topographical survey furnished by Northern States Power Company. Final location and orientation will be made after information is received on soil borings now being made.

A conceptual plant layout is being developed by Pioneer Service and Engineering Company. The presence of radioactivity in the steam is a major consideration in laying out the turbine plant. It is expected that equipment will tend to be grouped according to its respective radioactivity level. In this way the cost of necessary radiation shielding and adequate ventilation will be minimized. Means for providing access for inspection and maintenance of equipment during plant operation are being studied.

Work is progressing on a layout of turbine and reactor plant auxiliaries. These include water treatment facilities, offices, labs, shops, fuel shipping and handling areas, and others. Preliminary layouts are expected to aid in planning future detailed studies of these facilities.

10. MATERIALS AND METALLURGY

CLADDING MATERIALS

Fabrication studies were performed on various stainless steels for cladding the reactor vessel, main steam and recirculating water piping. Investigation on the weldability of Type 405 stainless steel demonstrates that a good, fairly tough weld can be produced on this steel. Ductility is lost in the process but can be restored by annealing the material at 1200 F – 1400 F. However, at these temperatures, a tendency exists toward excessive grain growth with grain sizes that cannot be refined by further heat treatment. Though Type 405 is inferior in corrosion resistance to Type 304, further investigation is planned in the areas of corrosion and fabrication.

A recommendation has been made for the use of Type 304L to clad the reactor vessel, backed by A-212 carbon steel. An alternative backing material is U.S.S. T-1 steel. This alloy appears excellent from a strength standpoint, permitting reduced vessel wall thicknesses. However, it must be heat-treated to obtain the desired mechanical properties. Special fabrication techniques must be followed in order to maintain these desirable mechanical properties. Studies conducted on the stress corrosion behavior of austenitic stainless steels indicate that under stresses of about 5000 psi, all 18-8 type alloys in chloride-bearing solutions appear to be susceptible to stress corrosion cracking. Extra low carbon stainless steel Type 304L apparently possessed the least tendency for cracking. In addition, the low carbon content minimizes the possibility of intergranular attack.

SUPERHEATER

Type 316 stainless steel has been investigated as a material for the superheater tubes. This material displays favorable metallurgical and fabrication qualities desirable for the severe service demanded by superheater application. Intensive research is indicated to arrive at a definite conclusion.

CONTROL ROD, GUIDE TUBES AND CONTROL ROD DRIVES

Control rod material studies of both natural boron and boron stainless steels, as well as the 80% Ag – 15% In – 5% Cd alloy and gadolinium oxide (Gd_2O_3) cermet were continued.

Recent test data on pre- and post-irradiation of 1 w/o and 2 w/o, as reported by various government laboratories, appear to be more encouraging than test data shown earlier. It was indicated that the boron stainless steels (2 w/o max) in the wrought condition exhibited fairly good mechanical and corrosion properties. Similar boron-stainless steel test specimens from ANL are now in the Materials Testing Reactor with a scheduled irradiation exposure of 10^{21} nvt. Tensile and impact test results will not be available for at least another year.

The ternary alloy, 80% Ag – 15% In – 5% Cd, appears to show considerable promise. Recent developmental information from WAPD indicates that this alloy can be welded, formed or easily extruded in cruciform shapes with cross-sectional thicknesses less than 1/4 inch. From various test data, it was indicated that this alloy has good creep properties at 550 F, in addition to excellent radiation stability and corrosion resistance in water of about pH 9.0–11.0. Suppliers for the material, and a supplier who will extrude the six-foot long cruciform section required for CRBR have been located and preliminary pricing information obtained. It is felt that more developmental effort including corrosion testing in the CRBR environment is required to verify the potential of this alloy.

After a survey of information pertinent to the fabrication characteristics and possible suppliers for the Gd_2O_3 - stainless steel cermet materials, recent communications indicate development so limited in this area that no applications of this cermet exist in reactor control. Due to lack of data, the cermet survey has been temporarily discontinued.

Boron stainless steel is being considered as the reference control rod material. Potential suppliers of boron stainless steel and fabricated rods of this material were contacted. One vendor is able to supply this material in rolled plates containing 2% natural boron. Based upon a firm price for this material, a comprehensive study of fabricating problems, methods, and techniques has been initiated.

A small amount of Type 304 stainless steel has been ordered to determine which of four possible welding procedures will produce the least distortion in a ten-inch cruciform section six feet long. The Allis-Chalmers Research Division is now experimenting with a new welding technique which, if successful, can be adapted to CRBR requirements to produce a rod with negligible distortion. At this time the process looks very promising for carbon steel, but as yet has not been tried on boron stainless steel. After fabricating tests determine which design produces the least distortion, a small quantity of boron stainless steel will be purchased for the final tests.

Preliminary drawings for both the boiling and superheating control rods for the critical facility have been completed.

Preliminary drawings for the control rod guide tubes for the critical facility have been completed. These tubes will be fabricated from 1100 aluminum with a minimum of welding to reduce distortion.

Inquiries to vendors for the fabrication of Zircaloy-2 guide tubes for the reactor have been made. As yet, no one has expressed a desire to fabricate these pieces.

PIPING

A layout was made of the recirculating water piping system to determine the amount of piping and fittings required to complete the recirculating water loop. Quotations were requested for these items.

Material for pipe fabrication has not been resolved. "Lectro Clad" (nickel plate) has possibilities, but further testing is required before it can be considered seriously. The piping sketch and specifications were submitted to three suppliers of centrifugally cast pipe. Upon receipt of these quotations, it is planned to compare stainless dual metal, rolled and welded clad pipe, and solid stainless, rolled and welded pipe from the standpoint of materials, fabrication and erection costs.

HEAT TRANSFER LOOP

Drawings were reviewed on some of the parts for the heat transfer loop. These drawings were analyzed from the standpoint of proper selection of materials and ease of manufacturing.

CORROSION LOOP

It is planned to machine some test specimens of T-1 steel for the corrosion loop. Other specimens tested will be Kanigen plated carbon steel and "Lectro-Clad," an electrically deposited nickel plate. The last two materials have possibilities for applications in piping, valve bodies, and stationary component parts where corrosion presents a problem. Extensive corrosion testing will be performed on these materials to obtain substantial data for proper evaluation.

11. SITE

Test pumping of the full scale well on the site was accomplished during the week of May 26. A yield of 600-650 gpm was obtained. Rather extensive data on the effect of this pumping were obtained and are being analyzed. Initial micrometeorological data have been obtained and analysis of them is nearly complete. A report should be available about August 1. A geological report will also be available August 1. Preparation for a background radiation survey is proceeding. It is planned to have a meeting to discuss the plan for this survey in August. The present plan is to invite AEC, USPHS, and State officials from South Dakota, Minnesota, and possibly Iowa to this meeting.

Fig. 3 - Corrosion Test Loop Showing Circulating Pump, Heat Exchangers and Ion Exchangers

196424

Fig. 2 - General View of Corrosion Test Loop

196423

Fig. 3 - Corrosion Test Loop Showing Circulating Pump, Heat Exchangers and Ion Exchangers

196424

Fig. 4—Boiling Fuel Element Step-Down Rod

43-500-599

Fig. 5 - Assembly Model - Boiler Fuel Elements for Critical Facility

196390

43-400-860

Fig. 6-Reactor Cross Section

INTERNAL FISSION GAS PRESSURE FOR CRBR 9x9 ELEMENT-96 ELEMENTS

FJC 6-23-58

Fig. 7 - Internal Fission Gas Pressure for CRBR 9 by 9 Element - 96 Elements

Fig. 8 - Reactor Vessel

FOR MAT'L SIZES & WELD INFORMATION SEE DWG. 43-30486 *

—78 SPHER. R—
(REF.)

43-301-169

- *1:- RUNOUT OF SURFACES MARKED 'X'- WITH REFERENCE TO THE 90.980 TURN MUST NOT EXCEED .001 I.R.
- *2:- ¹²⁵✓ FINISH ALL OVER UNLESS OTHERWISE SPECIFIED.
- *3:- ALL CLAD PLATES TO BE ULTRASONICALLY TESTED FOR 100% MECHANICAL BOND PRIOR TO FABRICATION.
- *4:- ALL WELDS TO BE 100% RADIOGRAPHED IN ACCORDANCE WITH UW-51 OF SECTION VIII OF THE UNFIRED PRESSURE VESSEL CODE. FIRST TWO PASSES AND THE FINAL PASS SHALL BE MAGNAFLUXED. MAGNAFLUX INSIDE AFTER BACK CHIPPING AND PRIOR TO CLAD DEPOSIT.
- *5:- COMPLETED COVER SHALL BE STRESS RELIEVED AT 1050°-1100°F FOR 6 HOURS FOR SSTL-304CLAD, AND 1150°-1200°F. FOR 6 HOURS FOR SSTL-405 CLAD IN ACCORDANCE WITH UCS-56 OF SECTION VIII OF THE UNFIRED PRESSURE VESSEL CODE.
- *6:- HYDROSTATIC TEST COLD TO 990 P.S.I.G.

MATERIAL:-

BASE METAL---CARBON STEEL SA-212 GR.B
CLAD-----STN. STL. TYPE 304 LOW CARBON
ALTERNATE CLAD-STN. STL. TYPE 405

Fig. 9—Reactor Cover

Fig. 10 – Silicone Rubber E-Seal and Wedge Type Metallic Seal with Covers and Segmented Ring

196337

Fig. 11 - Autoclave Seal Testing Unit with Silicone Rubber E-Seal in Position

196335

BEFORE EXPOSURE

Fig. 12

194032

Metal Sprayed Vessel Coating
Corrosion Samples

(L to R) No. 1 - .008" Nickel
No. 2 - .008" Zinc
No. 3 - .008" Aluminum

AFTER EXPOSURE FOR 3200 HOURS AT
100 F AND 100% RELATIVE HUMIDITY

Fig. 15

196001

Fig. 13

194033

Metal Sprayed Vessel Coating
Corrosion Samples

(L to R) No. 4 - 1020 Carbon Steel Plate
No. 5 - Carbon Steel Flashing
No. 6 - .006" Aluminum

Fig. 16

196002

Fig. 14

194034

Metal Sprayed Vessel Coating
Corrosion Samples

(L to R) No. 7 & 8 - .012 Aluminum
No. 9 & 10 - .012 Aluminum Over
Carbon Steel Flashing

Fig. 17

196003

Fig. 18 - Test Tank for 3" Diameter Steam Separators (Separators Removed and Zero Flow Condition)

Fig. 19 - Steam Separator Test Tank Showing Air Bubble Carryover, without Separators. Down-
comer Velocity Approximately 9 ft/sec

196505C

Fig. 20 - Test Tank Showing 3" Diameter Steam Separators Operating at an Equivalent Reactor Recirculating Rate of 70,000 gpm without Air Bubbles

196505A

Fig. 21 - Test Tank Showing 3" Diameter Steam Separators Operating at an Equivalent Reactor Recirculating Rate of 70,000 gpm with Air Bubbles

196505B

Fig. 22—Control Rod Drive Motor and Limit Switches

Fig. 23—Control Rod Drive Motor and Limit Switches

Average Axial Power Distribution
in the Superheater

Variables: Power
Core Height

FIG. 24

Fig. 24 - Average Axial Power Distribution in the Superheater

Average Axial Power and Flux
Distributions in Boiler

Variables: Top Curve - Flux
Bottom Curve - Power
Core Height

FIG. 25

Fig. 25 - Average Axial Power and Flux Distributions in Boiler

Average Radial Power Distribution

Variables: - Power
Core Radius

Fig. 26 - Average Radial Power Distribution

Control Requirement

Variables:

Excess Reactivity
Operating Time

Parameters:

Top Curve - No Burnable Poison
Bottom Curve - Burnable Poison

Fig. 27 - Excess Reactivity vs Operating Time

Months of Operation at 0.8 Load Factor

VIEW 'A-A'

ITEM NO.	ITEM	DESCRIPTION	MATERIAL	QTY	UNIT	REMARKS
1	1	C.F. FUEL ELEMENT ASSEMBLY (BOILING SECTION)		43-500-591	501	
1	2	FUEL ELEMENT BOX ASSEMBLY		43-500-561	501	
1	3	NOZZLE ASSEMBLY		43-501-181	501	
72	4	TUBE SPACER ASSEMBLY		43-501-178	501	
321	5	FUEL TUBE ASSEMBLY		43-501-161	501	
8	6	SIMULATOR ASSEMBLY		43-501-177	501	
12	7	#5-40NC-2A-1/4 LG. FLAT HD. MACH. SCREW	ALUMINUM 10001	THYS.	001

-55-

43-500-591

Fig. 28-C.F. Boiling Fuel Element Assembly

MR. 501 REQ.	ITEM	DESCRIPTION	MATERIAL	PART NUMBER				WT.
				DWG. No.	MARK			
	1	FUEL TUBE ASSEMBLY		43-301-161	501			
1	2	TUBE		43-301-162	001			
AS REQ.	3	FUEL -.336 ^{+.001} _{-.000} DIA. x MANUFACTURED LENGTH	99% DENSITY	THIS	003			
2	1	PLUG - 1/4" #14 GA. (.064) x 328 ^{+.005} _{-.000} DIA.	X8001 ALUMINUM	THIS	004

NOTE #1:-
 LEAK TEST AFTER BRAZING AT
 10 P.S.I.

43-301-161

Fig. 29 - Fuel Tube Assembly

HOLES OF THIS TYPE WILL BE USED TO VARY THE EFFECTIVE VOID FRACTION. SO THE NUMBER OF HOLES WILL CHANGE FROM PIECE TO PIECE

43-201-595

Fig. 32-Void Simulator

-09-

Fig. 33—C.F. Boiling Fuel Element with Void Simulation

NOTE 1:-
SURFACES MARKED 'X' MUST BE
CONCENTRIC WITHIN $\frac{1}{16}$ IN.
WITH EACH OTHER.

NOTE 1:-
AFTER ASSEMBLY ELEMENT
MUST BE WATERTIGHT AT JOINTS.

NOTE 2:-
SURFACES MARKED 'Y' MUST
BE STRAIGHT WITHIN $\frac{1}{16}$ IN.

ITEM	DESCRIPTION	MATERIAL	QTY	REMARKS
1	SUPERHEATER ELEMENT ASSEMBLY		1	
2	TUBE	43-30-160	508	
3	TUBE	43-30-160	508	
4	TUBE	43-30-160	508	
5	SPACER ASSEMBLY No 1	43-30-160	501	
6	SPACER ASSEMBLY No 2	43-30-160	501	
7	$\frac{1}{2}$ " SCH 40 (147 WALL) $\frac{1}{2}$ " LG PIPE	43-30-160	THIS	007
8	PLUG $\frac{1}{2}$ " I.D. DIA. $\frac{1}{2}$ " LG. RD. ROD	43-30-160	THIS	008
9	$\frac{1}{2}$ "-70 UNC-2B HWY. S.F. HEX NUT	43-30-160	0006015	
10	$\frac{1}{2}$ "-70 UNC-2B SHOULDER TYPE EYE BOLT	43-30-160	THIS	010
11	WASHER $\frac{1}{2}$ " O.D. $\frac{1}{2}$ " I.D. $\frac{1}{8}$ " THICK	43-30-160	THIS	011
12	WASHER $\frac{1}{2}$ " O.D. $\frac{1}{2}$ " I.D. $\frac{1}{8}$ " THICK	43-30-160	THIS	012
13	GASKET $\frac{1}{2}$ " O.D. $\frac{1}{2}$ " I.D. $\frac{1}{2}$ " LG	43-30-160	THIS	013
14	COATED WELD ROD	43-30-160	THIS	014
15	PLUG $\frac{1}{2}$ " I.D. DIA. $\frac{1}{2}$ " LG. RD. ROD	POLYETHYLENE	THIS	015

Fig. 34-C.F. Superheater Fuel Element Assembly

43-500-589

Fig. 35-Reactor Cross Section Critical Facility

Fig. 36—Critical Facility Schematic Piping Diagram

Fig. 37—Reactor Cleanup System-L.P. with Flash Tank

Fig. 38 - Superheater Cooling with Stagnant Steam at 1% Full Power

FIG 6-23-58

Fig. 39 - Flooding of Superheater with 489 F Sat. Liquid - Heat Generation = 0 (Negligible Effect)

Fig. 40—Schematic Flow Diagram for Controlled Recirculation Boiling Reactor

43-500-577