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**PATHFINDER ATOMIC POWER PLANT
TECHNICAL PROGRESS REPORT**

OCTOBER 1959 - DECEMBER 1959

Submitted to
U. S. ATOMIC ENERGY COMMISSION
NORTHERN STATES POWER COMPANY
and
CENTRAL UTILITIES ATOMIC POWER ASSOCIATES
by

**ALLIS-CHALMERS MANUFACTURING COMPANY
ATOMIC ENERGY DIVISION
Milwaukee 1, Wisconsin**

Ref: AEC Contract No. AT(11-1)-589

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ALLIS-CHALMERS MANUFACTURING COMPANY

Under

Agreement dated 2nd Day of May 1957, as Amended
between
Allis-Chalmers Mfg. Co. & Northern States Power Co.
under
AEC Contract No. AT (11-1)-589

March 15, 1960

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March 15, 1960

TECHNICAL PROGRESS REPORT
OCTOBER 1959 -- DECEMBER 1959

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FOREWORD

This is the ninth of a series of reports covering technical progress on the research and development program being performed in connection with the design of the Pathfinder Atomic Power Plant. This plant will be located at a site near Sioux Falls, South Dakota and is scheduled for operation in June 1962. Owners and operators of the plant will be the Northern States Power Company of Minneapolis, Minnesota.

The U.S. Atomic Energy Commission, through Contract No. AT(11-1)-589, with Northern States Power Company, and Central Utilities Atomic Power Associates*(CUAPA), are sponsors of the research and development program.

Allis-Chalmers Manufacturing Company of Milwaukee, Wisconsin, under contract with Northern States Power Company, is performing the research, development, and design; and will construct the plant including the reactor, which is designated the Controlled Recirculation Boiling Reactor (CRBR) with nuclear superheater. Pioneer Service and Engineering Company of Chicago, Illinois is providing the architect-engineer services to Allis-Chalmers. Portions of the R & D program, particularly in connection with fuel development, have been subcontracted by Allis-Chalmers.

Conceptual engineering and component research and development is well under way at Allis-Chalmers Greendale Laboratories. Erection of a number of major experimental facilities, which will provide important design data, has been completed. A Critical Experiment Facility was constructed and is now in operation.

*CUAPA Member Companies:

Central Electric and Gas Company	Northern States Power Company
Interstate Power Company	Northwestern Public Service Company
Iowa Power and Light Company	Otter-Tail Power Company
Iowa Southern Utilities Company	St. Joseph Light and Power Company
Madison Gas and Electric Company	Wisconsin Public Service Corporation
Mississippi Valley Public Service Company	

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DESIGN DATA

CRBR WITH NUCLEAR SUPERHEATER

Plant

Power, boiler region.....	164,000 kw
Power, superheater region.....	39,700 kw
Steam flow at rated power.....	615,000 lbs/hr
Total core power.....	203,700 kw
Gross electrical capability.....	66,000 kw
Net electrical output.....	62,000 kw
Net efficiency.....	30.5 per cent
Steam outlet pressure.....	540 psig
Reactor operating pressure.....	600 psig
Temperature, boiler region.....	489 F
Outlet temperature, superheater region.....	825 F
Gross heat rate.....	10,370 btu/kw hr
Reactor building size.....	50 ft dia x 120 ft

Reactor

Vessel size.....	11-1/2 ft x 31-1/2 ft
Total core dimensions.....	6 ft x 6 ft
Dimensions of superheater region.....	6 ft x 30 in
Fuel, boiler (Al clad).....	approx. 1.8 per cent enriched U ²³⁵
Fuel, superheater (S.S. clad).....	approx. 93 per cent enriched UO ₂
Fuel, loading, boiler (U-235).....	117 kg
Fuel, loading, superheater (U-235).....	42 kg
Power density (boiler core coolant).....	87 kw/liter
Average heat flux, boiler region.....	130,000 btu/hr/ft ²
Average heat flux, superheater region.....	80,000 btu/hr/ft ²
Maximum heat flux, boiler region.....	462,000 btu/hr/ft ²
Maximum heat flux, superheater region.....	250,000 btu/hr/ft ²
Recirculation rate.....	60,000 gpm
Recirculation pump power.....	794 kw

1. FUEL ELEMENT RESEARCH AND DEVELOPMENT

1.1 FUEL MATERIAL CLADDING, BONDING, AND IRRADIATION TESTING

1.1.1. Cladding

1.1.1.1. 8001 Aluminum. Dynamic Corrosion Tests. Different methods of reducing the corrosion rates of 8001 aluminum alloy were investigated. These methods involved: 1) variations in the aluminum-surface-area--to--water-volume ratio in the corrosion loops, and 2) static prefilming of the corrosion samples.

The test involving a high area-to-volume ratio in Loop 2 was described in the previous quarterly report.¹ The test was discontinued after 57 days of sample exposure to the following test conditions: 500 F, 1200 psi, pH 6.1, and a flow velocity of 27.6 fps. The test results indicated no improvement in the corrosion rate, probably because of the particular loop flow arrangement. The average corrosion rate after the initial period was 23 mils/year. A plot of the average corrosion rate is shown in Figure 1.1.

A new test was started in Loop 2. The system piping and flow were changed to the original arrangement. The ion exchange resins, the filter,

1. Pathfinder Atomic Power Plant, Technical Progress Report, July 1959 -- September 1959, ACNP-5924, p.1

and the conductivity cell were replaced in the purification stream, and a 13.125-inch length of 0.42-inch o.d. X8001 aluminum tubing was inserted in the J section of the main piping upstream from the sample holder. The purpose of this tubing was to increase the aluminum-surface-area--to--water-volume ratio in the loop to $134.6 \text{ cm}^2/\text{liter}$, which corresponds to the area-to-volume ratio expected in the Pathfinder. The conditions during this test were as follows: 500 F, 1200 psi, pH 5.8, and a flow velocity of 27.9 fps. The samples were inspected after 10.9, 22.8, 35.7, 47.0, and 56.7 days of exposure. The test is not yet complete, but preliminary evaluation indicates that the initial low corrosion period is being prolonged considerably. After 56.7 days the samples were still in the initial corrosion stage. It is expected that besides prolonging the initial corrosion stage, this test will also yield a lower long-term rate than previously obtained.

The test on statically prefilled samples in Loop 3, described in the previous quarterly report,² was resumed, and completed. The samples were exposed for a total of 72.6 days at the following test conditions: 500 F, 1200 psi, pH 6.0, and 27.7 fps flow velocity. The long-term corrosion rate obtained was considerably higher than in any previous test; although the initial period was somewhat prolonged, and the initial rate was lower than in all but one of the previous tests. During the exposure period of from 61.7 to 72.6 days, the average corrosion rate was about 48 mils/year. These results seem to agree with results obtained at other laboratories. A plot of this average corrosion rate is also given in Figure 1.1.

2. Ibid., p.2

According to a recent report from Chalk River, the static prefilming, when done at the same temperature as the dynamic test, may not reduce the long-term corrosion rate, even though the initial corrosion rate may be somewhat improved; whereas, static prefilming at a temperature higher than the dynamic test temperature may reduce the corrosion rate.

A new test in Loop 3 was begun with samples statically prefilmed at the following conditions: 600 F, 1543 psia, pH 5.6, and exposure time of 3.1 days. The dynamic test is being run at 500 F, 1200 psi, pH 6 - 7, and a flow velocity of 28 fps. Results of this test are not yet available. Dynamic tests in Loop 1 have not yet been resumed. The sample holder will be modified to accept round samples.

Extruded Tube-Plate Cladding Studies. The localized corrosion of X8001 aluminum tube plates was studied no further than previously described.³ No additional development work is planned.

1.1.1.2. Aluminum Double-Alloy, M400 on X2219. Static Corrosion Tests. A static corrosion test of M400 aluminum cladding on X2219 alloy was completed. The test conditions were: 500 F, 681 psia, pH 5.4 - 6.4, and total exposure time, 33.6 days. The samples were inspected after 8.1, 17.9, 25.7, and 33.6 days of exposure. Before the corrosion test, the samples were heated at 932 F for 100 hours in a muffle furnace, in order to observe diffusion phenomena. The corrosion rates obtained were in very good agreement with test results obtained previously on X8001 alloy and on X2219 clad with M400. Results of this test and a comparison graph of the results of static tests of both alloys are shown in Figures 1.2 and 1.3, respectively.

3. Ibid., p.4.

Annealing Tests. Sheet X2219 clad with M400 alloy was annealed for 100 hours at 1000 F to produce diffusion of copper equivalent to 1 year at 500 F. The M400 was exposed to increasing amounts of copper by machining off varying amounts of clad. Static corrosion tests of the material were made as previously described. Metallographic evaluation of the samples is being made.

1.1.1.3. Aluminum Double-Alloy, 1100 on 5083. No work was done during the quarter on 1100 cladding on 5083. Recent reports from other laboratories indicate the 1100 alloy with a silicon content less than 0.002 per cent has increased corrosion resistance. With such an alloy, diffusion of magnesium should not result in precipitation of Mg_2Si at the grain boundaries, and magnesium diffusion may not be detrimental. However, other techniques would have to be developed to detect and evaluate the diffusion of magnesium.

1.1.2. Irradiation Testing

1.1.2.1. Boiler Fuel Elements. Capsule Tests. Capsules 1, 2, and 4 were inserted in the GETR for the October cycle. Soon after startup of the reactor, it was found that several heaters and thermocouples failed. The failures were attributed to insecure heaters and thermocouple junctions within the capsule.

The capsules were exposed to a short cycle during October and a full cycle during November. The temperatures recorded during the November cycle showed a peak at the end of the cycle of 433 F in the bottom of the 7-per cent enriched sample. In the 6-per cent enriched samples, the temperature leveled off at 400 F, short of the 500 F that was expected.

During the November cycle, a control rod was shadowing the capsules and reducing the flux by an estimated 10 to 20 per cent. The control rod was

raised before starting the December cycle, so that the shadow was eliminated.

At the end of the November cycle, a new flux profile was determined. This profile was different and should be more accurate than the profile determined initially.

Capsule 3 was inserted in the reactor at the same level as the other capsules for the December cycle. The heaters were not turned on; so that the thermocouple readings could be used as a reference for the other three capsules. At the end of the cycle, a study will be made of the temperature and heat flux plots to determine if the capsules should be repositioned to obtain the required values.

Subassembly Tests. The pressure drop and by-pass flow tests of the 9-pin subassembly were completed at Greendale Laboratory. Results are reported in section 1.2. Due to a recent modification of the seat of the facility tube, it will be necessary to re-run a portion of the by-pass test in the facility tube.

Analysis of the subassembly indicated a hot spot adjacent to the aluminum spacers in the fuel pins. Flux peaking as high as 37 per cent was calculated for fuel pellets adjacent to the bottom spacer. This would cause the maximum allowable heat flux for the fuel pin to be exceeded.

The hot spot was eliminated by placing a 2 per cent natural boron stainless steel shield 0.25 inch thick around the subassembly. The shield covers the aluminum spacer at the bottom two fuel-pin joints of the subassembly. Nuclear analysis of this problem is discussed in section 3.2.

A safeguards analysis of the modified test loop, which will be used for the subassembly tests, was prepared for the purpose of obtaining an AEC license amendment to operate the loop. The analysis should be submitted to AEC the early part of January. It is expected that six to eight weeks will be required to obtain the amendment.

The modifications on the loop and facility tube are nearly complete. The installation of the facility tube and shakedown of the loop should be completed before the AEC license amendment is issued. Currently, it appears that the earliest possible insertion date will be early March.

The particular end-cap and spacer geometry used on the boiler irradiation subassembly required three types of welds. A type A weld is used where a solid end cap is welded into an empty tube. After welding, the tube is loaded with pellets and drawn. A type MF weld is used when an end cap with a hole through the middle is inserted and welded. While making this weld, a slight vacuum is effected through the hole. These caps are then machined and threaded into male and female parts for further joining. After these caps are screwed together, they are seal-welded with a type J weld.

A total of 464 MF welds were initially made under various conditions. Of these welds 89 per cent were found satisfactory after a not too stringent metallographic examination. The welds were characterized by extreme non-reproducibility.

A total of 445 J type welds were found satisfactory after a not too stringent metallographic examination. Again, results were spotty and not satisfactory.

All destructive tests of type A welds showed them to be within specifications, and no further work on this type of weld was done.

Data recording and cleaning procedures for the type MF and J welds were apparently not complete enough to maintain control over the welding process. Therefore, various weld parameters, chills, cleaning, preheating, pressure control, etc., were investigated more completely. Welding with direct and alternating currents were investigated separately. Metallographic sections taken through two planes were made; and welds with any voids within 0.040 inch of the surface, strings, cracks, or lack of penetration were rejected. Results of this examination were correlated with X-ray examinations.

Direct-Current Welding. Most of the rejecting defects in the MF welds were eliminated or minimized by improvements in techniques. Some 60 samples were welded to check reproducibility. Of these samples, 16 per cent were rejected for voids of 0.002 to 0.005 inch diameter or the "thin-wall defect" shown in Figure 1.4. Of the 16 per cent, 12 per cent were previously detected by X-ray examinations, and 4 per cent were found in subsequent metallographic examinations. Therefore, 95.5 per cent of samples approved in X-ray examinations were within specifications.

Work on the J type weld using conventional welding techniques was discontinued, when no improvement was realized. Work was initiated using an evacuable box. Even preliminary work with this device showed striking improvement. Of later sample welds, 95 per cent or better were found to be within specifications. Destructive tests indicated that 99 per cent of samples approved in X-ray tests were within specifications.

The evacuable box used consisted of a T-shaped glass chamber as shown in Figure 1.5. Prior to welding, the chamber was evacuated to 29 inches of mercury and the nichrome heaters were turned on. The parts being welded were rotated through the rotary seals. A helium back-fill was started with the arc establishing itself at about 3 inches of mercury. The weld cycle was completed under a pressure of 12 psig.

In general, results indicated that if penetration was greater than 0.100 inch, all porosity was greater than 0.040 inch below the surface.

Alternating Current Welding. Basic welding parameters and gas mixtures were determined. It was found that equipment alignment and cap-tube clearances were very important. Knurling of the mating surface of the cap produced the best results. An interference fit of the cap in the tube was not satisfactory. Penetration was the limiting factor with A-C welding. The weld width-to-depth ratio was 4 with alternating current compared to a ratio of 1.5 to 2 with direct current. It was decided that this could be improved by superimposing a direct current on the alternating current. However, work on A-C welding was discontinued due to equipment limitations, and because good results were obtained with direct current.

Fabrication of the subassembly was completed except for autoclave and final helium-leak check and inspection. During fabrication all type A or MF end-cap welds passed inspection by X-ray. One J-type weld was rejected because of poor surface. This gave a high degree of confidence much higher than expected.

1.2. HEAT TRANSFER AND FLUID FLOW INVESTIGATIONS

1.2.1. Heat Transfer Loop

All material for the superheater addition of the heat transfer loop were received. Construction will be started and should be completed in January.

About 50 per cent of the material for the superheater test section was received.

A dummy superheater test section was completed. This section will be used as a jig for aligning the superheater loop addition and as a hot pressurizing test section.

Construction of the 10-8-6 pin boiler test section is about 80 per cent complete. All the one-pin test sections were received. Differential thermal expansion was more severe than calculated, and the test section was modified to minimize this problem.

The unipolar generators were modified to eliminate the rotor insulation problem that resulted in several shutdowns. A new insulation was applied to the rotor, and the generator has run satisfactorily since November 20, 1959.

While running hot pressurizing tests on the one-pin test section, both of the small canned-rotor pumps failed due to cavitation in the fluid piston bearings. As a result, one-pin burnout tests were made using natural circulation. Thus far, a heat flux of $400,000 \text{ Btu/hr-ft}^2$ has been reached without burnout. Test data is shown in Figure 1.6. All temperature readings were corrected for thermocouple error due to position in the rod and voltage drop across the thermocouple.

1.2.2. Gamma-Ray Attenuation Equipment

The Gamma-Ray Attenuation equipment that will be used for void determination in two-phase flow tests consists of the following components:

1) scintillation detector, 2) analytical count-rate meter, 3) radiation analyzer, 4) electronic recorder, 5) rack and positioner for vertical and horizontal positioning at a rate of 6 inches/minute, 6) line noise filter, and 7) two gamma sources: a) 10-mc Cesium 137 source (662 kev energy peak), and b) 10 curie Thulium 170 source (85 kev energy peak).

The equipment is presently being used to make void determinations of air-water mixtures in the low pressure air-water flow test at atmospheric pressure. It will be used to make void determinations of steam-water mixtures at 615 psia and 489 F in the heat transfer loop.

The reference points of 0 per cent voids and 100 per cent voids were obtained at various points in the test section by measuring attenuation through water and air at atmospheric conditions. The reference points for the heat transfer loop will be obtained at various points in the test sections by measuring attenuation through saturated water at 615 psia and 489 F, and through air at atmospheric conditions.

The accuracy of the procedures and equipment in making intermediate void determinations is presently being evaluated by making measurements through lucite blocks. As soon as this evaluation is completed, attempts will be made to correlate slip ratios and voids, and pressure drops and voids.

1.2.3. Irradiation Subassembly Pressure-Drop and By-Pass Fluid-Flow Tests.

Flow tests were performed on an unfueled aluminum clad 3 x 3 irradiation subassembly, which will be irradiated in the GETR. The tests were

necessary to insure enough by-pass flow to prevent burnout of the facility tube by gamma heating, and enough flow through the subassembly to prevent boiling burnout. The tests indicated that the design is adequate.

1.2.4. Low-Pressure, Air-Water Tests

The test program was delayed because the loop was used for the by-pass flow tests described above. The same boiler element mock-up will be used but more instrumentation will be provided. Most of the effort thus far was made to iron out problems associated with the test element and equipment.

Preliminary results indicate that the single phase friction factor is higher than that used for drawn tubing of equivalent hydraulic diameter.

The data also seems to indicate that the single phase entrance and spacer losses calculated are conservatively high.

1.2.5. Superheater Air Flow Tests

An experimental program to determine the frictional pressure drops through a superheater fuel element was initiated. Air at room temperature is blown through a model of the fuel element, and pressure measurements are taken at intervals along the length of the element. A drawing of a typical test section is shown in Figure 1.7, and the test section housing is shown in Figure 1.8.

An effective friction factor in an annulus may be calculated from pressure drop measurements by the relation,

$$f = \frac{(D_i - D_o) gc}{2 \rho v^2} \frac{dP}{dx} \quad 1.1$$

where,

f = friction factor

D = diameter

P = pressure

ρ = density

x = axial distance

During the initial phase of testing, the friction factor in a single annulus will be determined. In addition, the effect of varying the wire spacer pitch, wire spacer fit, annulus diameter ratio, and Reynolds number will be determined.

Results of these tests will be used to prepare specifications on the superheater wire spacers.

The second phase of testing will be with an exact model of the superheater fuel element of the reference design.

A seven-pin scalloped flow tube (Figure 1.9) is presently being considered as a design for a low enrichment superheater fuel element. A high pressure drop, as indicated by data from tests in which liquid sodium was used as the fluid, is the major drawback of this design; and tests will be made in the air-flow test loop to determine if the pressure drop with air would be acceptable. The test housing for this tube is shown in Figure 1.10 and 1.11.

1.3. FABRICATION OF FUEL FOR CRITICAL FACILITY

1.3.1. Boiler and Superheater Fuel

Delivery of all boiler fuel rods from the vendors was completed.

Upon receipt of materials for the superheater, M & C Nuclear Corporation undertook final development of the fuel tubes. Certain variables had not been considered in the ordering of stainless steel tubing, and it was very difficult to maintain the cermet thickness at the specified values. In

addition, considerable difficulty was experienced in loading the tubes prior to drawing operations, and concentricity of the tubes and homogeneity of the cermet was not maintained. Development continued on the outer fuel tube throughout the period from September to December. Trial production of five tubes containing enriched uranium dioxide to check out final changes in loading technique began in Mid-December. However, evaluation of the tubes indicates that the required fuel homogeneity and dimensional requirements have not been maintained.

Development of the inner tube will commence as soon as problems with the outer tube have been solved.

1.3.2. Flux Monitoring Pellet Fabrication

1.3.2.1. Design Status. A fabrication study was initiated during this period on two types of pellets for flux monitoring experiments in the Allis-Chalmers Critical Facility. One type of pellet consists of a flat disc, 0.080 inch thick, which will be cut from a boiler fuel element pellet, 0.338 inch in diameter. The other type is a slotted pellet, which also will be cut from boiler pellets. This latter type will consist of two halves attached together with five 0.020 by 0.020 inch holes traveling the length of the pellet. These fabrication operations will be performed on a diamond cut-off wheel.

1.3.2.2. Tests and Studies. A plexiglas enclosure was fabricated for the diamond cut-off saw, and is now being connected to the radioactive materials ventilation system for the Fuels Laboratory. Special jigs required for cutting these two types of pellets were machined, and initial checkout cutting studies are being performed on cordierite pellets of similar size.

Actual fabrication studies on uranium dioxide pellets of three enrichments will be initiated during the first part of the next period. Requirements call for 10 wafer type pellets of each enrichment; natural uranium dioxide, 1.8 per cent uranium-235 in uranium dioxide, and 2.6 per cent uranium-235 in uranium dioxide. Three slotted pellets of the 1.8 per cent enrichment, and two each of the natural and 2.6 per cent enrichment grades will be required.

1.4. FUEL ELEMENT MANUFACTURING RESEARCH AND DEVELOPMENT

1.4.1. Boiler Fuel Element

1.4.1.1. Zircaloy-2 Clad Fuel Element. Mechanical Design. Zircaloy 2 cladding is an alternate for aluminum cladding in the boiler core of the Pathfinder. Since the mechanical and physical properties of the two metals differ, a different mechanical design is required. A preliminary design, with which calculations and estimates could be made, was prepared. Zircaloy 2 is used for the fuel element box as well as for fuel rod cladding, and stainless steel is used for rod spacers and supports.

Fuel rods six feet long are arranged in a 9 x 9 square lattice. The diameter of the rods is stepped down from 0.388 inch in the lower half to 0.350 inch in the upper half. A pellet diameter of 0.338 inch is retained from the aluminum clad element design for the lower half of each rod, and a diameter of 0.300 inch is used for the upper half. The autoclave conditions used to put the corrosion resistant film on the cladding requires that the cladding have a minimum thickness of 25 mils.

The fuel rods are supported and spaced at the ends by tube grids. These grids are fabricated by brazing tube segments of proper dimension. The rods are spaced every 18 inches by stainless steel tube sheets. This design provides excellent lateral support but little axial restraint.

The fuel element box is designed so that it may be easily removed from the rod bundle and used as part of another fuel element assembly.

The nozzle and lifting bail are essentially the same as for the aluminum clad fuel element.

An alternate mechanical design is being considered for the Zircaloy 2 clad fuel elements. This design which is shown in Figure 1.12 uses a 9 x 9 square lattice in the lower half and an 8 x 8 square lattice in the upper half. Rods with a 0.388 inch o.d. are used throughout the element. Advantages of this design are that 1) uniform diameter fuel rods may be used, 2) the metal volume in the core is reduced, and 3) fewer rod sections are required.

1.4.1.2. Tests. Nozzle Galling Test. Stainless steel parts loaded and moving relative to each other have a well-known tendency to gall. This characteristic of stainless steel may be a problem with contact surfaces between the grid plate and fuel element nozzles. Horizontal loading of the nozzle will result from the thermal bowing of the element, and from misalignment, and may reach a magnitude of 400 psi in the worst case. The nozzle will move vertically relative to the grid plate during fuel element insertion and removal, startup and shutdown, and during operation. The differential thermal expansion between the fuel element and the boiler shroud, which supports the hold-down fixture, is expected to be the limiting condition.

Galling and/or seizure of the nozzle would present serious mechanical difficulties. Supercritical column loading of the fuel element and/or inability to remove the nozzle from the support grid without excessive damage may result. Consequently a test was planned to provide specific information on the galling problem under Pathfinder operating conditions.

The test fixture is shown in Figure 1.13. The vessel will be nearly filled with water and heated by six band heaters controlled by a thermocouple operated switch. Tests will be run at room temperature and at reactor operating temperature and pressure. Various horizontal loadings will be imposed with the two air cylinders. The force required to move the nozzle vertically will be noted with the different horizontal loads, and visual examinations will be made to determine the severity of galling. Should excessive galling be obtained with the present stainless-to-stainless design, tests will be made with coated or surfaced nozzles, and/or with nozzles of other materials.

Fuel Rod Vibration Test. Work began on phase one of the test program previously outlined.⁴ Parts for the test loop are on order or have been received. End-fixity brackets will be used with both the strap-type rod support and tube-grid, tube-sheet type support.

1.4.2. Superheater Fuel Element

1.4.2.1. Design Status. Reference Design. Work on the reference design using the ASTER EDFM Code is progressing satisfactorily. A superheater fuel element design was completed using previously published values

4. Ibid., p.17.

of hot spot factors.⁵ This design makes use of varying fuel element diameters to minimize the hot spot temperature in the hottest channels. The previously determined hot spot factors are presently being reviewed and those that have not been determined are now being evaluated. This analysis should be completed soon.

1.4.2.2. Tests and Studies. Thermal Fatigue Analysis. Reactor noise, fluctuations in reactivity inherent in the system, would be a possible cause of superheater tube failure if the frequency of the fluctuations and the magnitude of the strains were great enough. The problem was analyzed as follows. The temperature drop from the center of the tube wall to the outside surface was determined to be 6.7 F. With this temperature differential the thermal stress was computed to be 1820 psi. When compared with the allowable yield stress of 18,000 psi at the system temperature, it is apparent that the temperature differential would have to be considerably higher to raise the stresses into the plastic range, i.e. an extremely large increase in power generation would be required. Even when other strains, e.g. those resulting from mechanical restraints and column loading, are combined with the thermal stresses, the possibility of fatigue failure is remote.

Superheater Fuel Element Seal Test. The material, 304 stainless steel, is being considered for the superheater insulation-support tubes. Although all available information on the erosion-corrosion characteristics of the 300 series of stainless steel indicate it to be suitable for its long-term operation in the Pathfinder core, it would still be advisable to inspect the

5. Pathfinder Atomic Power Plant, Final Feasibility Report, Boiling Water Reactor with Internal Superheater, ACNP 5917, p.39, 1959

tubes from time to time. To inspect the inner surface of the outer support tube, it would be necessary to make the inner tube readily removable. This would be feasible if a sealing joint between the inner and outer tube can be provided, so that steam leakage during operation is minimized.

A program was initiated to develop a seal that can be easily replaced under water and still have a reasonable degree of integrity. The program is divided in two phases. The first phase will include a series of preliminary tests of different designs, so that poor seals are eliminated from consideration as fast as possible. The second phase of testing will further evaluate seals that appeared promising during the first phase.

The test fixture for the first-phase tests is shown in Figure 1.14. The test cell will be used to check leakage through individual seals. The cell can be used with saturated steam up to pressures of 100 psi. The differential pressure across the seal can be controlled from 0 to 100 psi.

During the second phase of the tests, seals will be tested in a cell for greater periods of time and at actual reactor conditions.

ASTER Code. The ASTER EDPM Code was modified to allow for the effect of pressure drops above and below the fueled region. The allowance includes expansion, contraction, and frictional effects before and after the flow is divided among the annular regions. Allowance is also made for the frictional effect of a non-fueled annular region.

The changes should give a more detailed description of fluid flow and heat transfer through the entire superheater section. The modifications are presently being checked out.

1.5. NUCLEAR HANDLING TOOLS, COFFINS, SHIPPING CONTAINERS

Fabrication of a mock-up of one-eighth of the CRBR core was completed. Some reworking of the dummy boiler fuel elements was necessary so that they could be readily inserted into the grid plate. The mock-up is shown in Figure 1.15.

Fifteen dummy superheater fuel elements for use in the mock-up were constructed. Since the reference design for the superheater element calls for a helically wrapped wire spacer, a test to determine the best wire diameter was conducted. Elements wrapped with various diameter wire were assembled. The wire diameter was chosen so that an element could be inserted and withdrawn from a tube that was bowed more than the amount of clearance allowed.

It was necessary to develop a technique for wrapping the wire around the elements. A device was designed, with which the tubes can be wound with various diameter wires at various pitches. This device is now being built.

Two types of grappling mechanisms for handling the superheater elements were developed and are undergoing dry tests with the mock-up.

A boiler element handling ball was designed, and a sufficient number were constructed for the core mock-up. They are being assembled on the boiler element mock-ups.

A pole for guiding the boiler elements while loading them into the core was constructed.

A one half ton handling crane was received and will be installed. This crane will be used to place the mock-up in the 40-foot tank for final underwater tests.

Various combinations of underwater lights were tried in the tank. A layout simulating one half of the actual core was placed in the tank to determine their effectiveness. Two 1000-watt underwater lamps appear to provide adequate illumination for refueling and have been installed in the tank.

1.6. LOW ENRICHMENT SUPERHEATER FUEL ELEMENT

1.6.1. Design Status

The research and development program to develop a low enrichment fuel element for the superheater core of the Pathfinder was continued. A description of the program and its objectives was previously given.⁶

1.6.2. Tests and Studies

1.6.2.1. Programmatic Data Survey. The survey was continued with primary interest in information on methods of cold-compacting uranium dioxide fuel bodies, powder characteristics, lubricants, binders, density improving additives, methods of compaction and consolidation, sintering cycles and atmospheres, and alternate fabrication processes. Reports in our library were cataloged and reviewed. Procurement of additional reports from AEC-TISE and other facilities was initiated. The information from this survey, which will be compiled later, is now being used in the fabrication experiments.

1.6.2.2. Ceramic Fabrication Studies. Depleted Uranium Dioxide Powder -- Shattuck Chemical Company. Powder Analysis. A Fisher sub-sieve analyzer is now being used along with metallographic techniques to study

6. Pathfinder Atomic Power Plant, Technical Progress Report, July 1959 - September 1959, ACNP 5924, p. 22, 1959

powder particle size, shape, porosity, etc. The depleted oxide from Shattuck Chemical Company, which is being studied, was found to be of typical ceramic grade characterized by irregular shaped agglomerated crystallites with high surface area and varying particle size. An analysis of powder as-received and after ball milling is given in Table 1.1.

TABLE 1.1
ANALYSIS OF SHATTUCK DEPLETED URANIUM DIOXIDE

Condition	Metallographic Analysis		Sub-Sieve Analysis
	Range ()	Ave. ()	Ave. ()
As-received	1-30	5-7	1.7
Ball Milled 25 Hours	0.5-10	2.5	1.0
Ball Milled 64 Hours	0.5	0.5	0.5

Pelleting Studies. Three series of pellets were formed from 1) as-received powder, 2) powder ball milled for 25 hours, and 3) powder ball milled for 64 hours. The ball milling operation consisted of tumbling 600 grams of oxide and 500 grams of alumina grinding balls in a one-quart jar for the desired length of time.

Five or six different groups of pellets were pressed in each series. The groups differed from one another only in the lubricant and/or binder used for green strength. The lubricants and/or binders used were as follows: 2.0 w/o Sterotex; 0.4 w/o Sterotex; 2.0 polyvinyl alcohol (PVA); 2.0 w/o PVA -- 0.4 w/o Sterotex; 2.0 w/o Carbowax; 2.0 w/o Carbowax -- 0.4 w/o Sterotex.

A group consisted of five to nine pellets, each of which were formed at a different pressure.

The green, unfired density and pellet appearance were the criteria used to select the best binder-lubricant, forming pressure, and powder size. A pellet free from cracks and laminations with sufficient green strength for handling is desired. The highest green density attainable, which results in minimum shrinkage of the pellet during sintering, is preferred.

Pellets of as-received powder were formed at pressures of from 20 to 120 tsi. The best appearing pellets were formed when 2.0 w/o Carbowax -- 0.4 w/o Sterotex was used as the binder. These pellets had green densities from 57.9 to 74.4 per cent of theoretical density. Pellets pressed using only Sterotex or only Carbowax as binder also resulted in good pellets with green densities from 52.1 to 67.3 per cent and from 57.1 to 74 per cent of theoretical density, respectively. Forming pressures from 60 to 100 tsi produced the best pellets with the highest green strength and density.

Pellets of powder ball milled for 25 hours were formed at pressures of from 20 to as high as 90 tsi. The best pellets were obtained when 2.0 w/o Carbowax and 2.0 w/o Carbowax -- 0.4 w/o Sterotex were used as binders. The best pellets were obtained at the higher forming pressures of 70 and 80 tsi. Green densities of from 63.8 to 72.6 per cent of theoretical density were obtained. Pellets formed using Sterotex, PVA, and PVA-Sterotex resulted in pellets with poor green strength and lower green densities.

Pellets of powder ball milled for 64 hours were formed at pressures of from 50 to 110 tsi. The best pellets were obtained when 2.0 w/o Carbowax or 2.0 w/o Carbowax -- 0.4 w/o Sterotex were used as binders. These pellets had green densities from 66.2 to 72.0 per cent of theoretical density. Pellets

made with PVA and/or Sterotex again had poor green strength and lower green densities. The best pellets were obtained at the higher forming pressures.

A second series of pellets of as-received power were formed at pressures of from 50 to 120 tsi. The quality of these pellets agreed well with those in the first run. Minor variations occurred that may be attributed to developments in technology and experimental errors.

From these initial studies in forming pellets of depleted oxide from Shattuck Chemical Company, it appears that the best green pellets can be fabricated using powder ball milled for 25 hours and with Carbowax alone, or Carbowax-Sterotex used as the lubricant-binder. The higher pressures of from 70 to 110 tsi appear to be optimum. Relatively high green densities were obtained with powder ball milled for 25 hours with forming pressures of only 70 to 80 tsi.

Sintering Studies. In sintering studies made thus far, primary emphasis was placed on standardizing the temperature cycle. In early sinters, the pyrometer and power-input control functioned improperly. The sintering cycle consisted in firing one group of pellets for 15 minutes and another group for 75 minutes. The maximum temperature was probably about 100 or 150 degrees less than the 1650 C temperature recorded for these tests.

Two groups of pellets were fired with proper temperature and control. The sintering cycle used consisted in starting the furnace with the pellets in the muffle, heating to 1600 C, and holding at temperature for 30 minutes, and then slowly cooling to about 600 C before removing the pellets from the furnace. The density of these pellets was well below the 93 per cent of theoretical density desired, probably because the time at temperature was too low.

Three groups of pellets were loaded into the boat and stoked into the furnace at about 900 C. The temperature was then raised to 1650 C and held at temperature for 90 minutes. The temperature was lowered to 900 C before removing the pellets from the hot zone. While higher densities were obtained as expected, the thermal shock of heat-up fractured many of the pellets. Many of the pellets had end and edge cracks and scratches on the sides from the thermal shock and forming die.

Six groups of pellets were fired but had to be removed from the hot zone at high temperatures because the furnace malfunctioned. As a result, the pellets involved in these firing cycles were badly fractured and laminated.

Very little firm data was obtained from the sintering studies to date. Pellet density was increased by 15 to 31 per cent for the various sintering cycles. It appears that any of the three types of powder pressed at 60 to 70 tsi and sintered for two or three hours at 1650 C should result in satisfactory fired densities. The tests will be continued to determine the best sintering temperature cycle, for pellets formed at 60 or 70 tsi.

No pellets using PVA and/or Sterotex were successfully sintered. The poor green strength and density makes use of this binder unattractive for further study.

Sintering studies are now in progress to evaluate longer sintering times with slow sample heating and removal from the furnace at 600 to 800 C. A programming controller for the furnace, which would be used to standardize furnace heat-up, time at temperature, and cooling cycles may be obtained. If this equipment is added, a better evaluation of the effects of sintering variables will be possible.

Studies of the Shattuck depleted oxide will be continued to obtain firm data. The study should be completed early in the next period, and a study of another uranium dioxide powder will be initiated.

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2. REACTOR MECHANICAL STUDIES

2.1. VESSELS AND STRUCTURES

2.1.1. Vessel Shell and Nozzles

2.1.1.1. Design Status. Final Design. The final design of the vessel shell is complete except for final results of the full-scale closure test, radiographic tests of the discharge nozzles, and thermal and hydraulic shock analyses. These tests and studies are described below. Final drawings and manufacturing specifications are essentially complete.

Fabrication. All major vessel components are on order. The delivery of the large clad hemispherical sections of the vessel will be several months behind schedule because of the steel strike. Some nozzle forgings were received. The forging for the flue ring was rejected after ultrasonic testing. A replacement forging is due in February. The rejection of the flue-ring forging should not affect the manufacturing schedule since the large hemispherical heads are still the controlling items in the schedule.

All welding processes to be used in the vessel manufacture were qualified in accordance with Allis-Chalmers specifications and the ASME Code requirements. A minimum of two welders per shift were qualified for every process.

2.1.1.2. Tests and Studies. Control Rod Drive Mounting Sleeves.

The control rod drives are mounted on 3-1/2-inch schedule 80 pipes welded into the 2-1/2-inch thick vessel head with full penetration welds. A full-scale radiographic test of the sleeve attachment weld was made to determine the limits of radiographic inspection techniques. A number of ceramic inclusions of 1/6- to 1/8-inch diameter and 5/16-inch length were placed at various positions and orientations within the sleeve attachment weld. Additional inclusions were placed within the sleeve wall. The junction weld was radiographed using the Allis-Chalmers betatron facility.

All inclusions lying outside the projected outer surface of the sleeve were detected. Those lying within this projected surface were not detected. The test indicates that adequate radiographs of nearly all of the weld can be obtained. Only the small fusion area lying within the projected outer surface of the sleeve cannot be radiographed. Due to the lengthy extension of the sleeves both to the outside and inside of the 2 1/2-inch thick head, an improvement in radiographic inspection methods does not seem possible.

Recirculation Loop Discharge Nozzles. The recirculation discharge nozzles will be fabricated of rolled ASTM 212-B plate with a 10 degree taper at the weld preparation to facilitate radiographic inspection. The vessel connection weld must lend itself to unquestionable radiographic inspection. A radiographic proof test is in progress to demonstrate the adequacy of the nozzle design.

A full-scale carbon steel model of a nozzle and vessel wall segment were fabricated. Ceramic inclusions are being placed at various positions and orientations in the attachment weld and nozzle wall. When the welding

is completed, the nozzle will be radiographed with the betatron facility.

Thermal Shock Studies. A computer program is now available to determine the effect of thermal shocks on the vessel flanges. The analytical results from this program are not yet complete. As soon as results are complete, they will be correlated with experimental data from the full-scale closure test. The maximum vessel heating and cooling rates will be determined.

2.1.2. Bolted Flange Closure

2.1.2.1. Design Status. Conceptual Design. Conceptual design of the bolted-flange closure is complete pending results of the full-scale closure tests described below.

2.1.2.2. Tests. Full-Scale Closure Test. All component items for the full-scale closure test assembly were received. Preparation and assembly of the subassemblies is well underway. The test vessel is about 50 per cent complete.

The preliminary laboratory tests to check-out instrumentation and strain-gage techniques is nearly complete. A method for waterproofing the strain gages that will be located inside the test vessel was successfully tested up to 1000 psig. A method of bringing strain-gage leads out of the vessel was also tested successfully.

The two hydraulic bolt tensioners for the closure tests were received in the laboratory. The tensioners and their associated hydraulic equipment are shown in Figure 2.1

2.1.3. Entrainment Separators and Support Shelf

A second series of moisture de-entrainment tests using a 20-inch

diameter test column is being started. The purpose of this series is:

- 1) to verify the diametrical correction factor indicated by tests⁷ in the 2- and 4-inch diameter test sections,
- 2) to determine accurately the efficiency, pressure loss, and capacity of both the mesh type and centrifugal type mechanical separators, and
- 3) to obtain additional information on the condition of the two-phase interface and to measure the droplet size, if possible.

The test column is now being fabricated and will be installed on the Turbine Test Floor at the West Allis Works.

2.1.4. Boiler Grid Plate.

2.1.4.1. Design Status. Final design of the grid plate is essentially complete. Material for fabrication is on order and is scheduled for delivery during the month of January.

2.1.4.2. Tests. A quarter-scale Plexiglas structural model of the grid plate is being made to conduct a deflection test. Data from the test will be used to experimentally verify the calculated deflection and stresses.

2.1.5. Superheater

2.1.5.1. Mechanical Design. A specification for the superheater tubes was issued and inquiries were made to various suppliers for the fabrication of the tube.

The stress analysis of the superheater manifold flange is complete. The Taylor Forge method for undercut flange was selected. A stainless steel, cover gasket with soft iron core was used between the grid plate

7. J. Wilson and M. McDermott, Pathfinder Atomic Power Plant, Final Report on Moisture De-Entrainment Tests on 2- and 4-inch Diameter Test Sections, ACNP-5921, 1959.

and the manifold flange. The gasket is compressed by forty 1-inch diameter bolts pre-stressed to 15,000 lbs/sq. in. The stresses of the flange are given in Table 2.1.

Table 2.1. Superheater Manifold Flange Stresses

Component	Calculated Stresses (lbs/sq.in.)	Allowable Stresses (lbs/sq.in.)
S_{long}	9600	18,800
$S_{rad.}$	3610	12,500
S_{tang}	1800	12,500

2.1.5.2. Tests. Superheater Moderator Flow Test. A test to determine the pressure drop across the tube sheets is scheduled to be run in January. The test apparatus represents a typical channel of the tube bundles of the superheater. It is composed of two sections of four 3/4-inch pipes, schedule 40, flange-connected. The pipes are disposed on a square pitch. Sharp edged orifice plates with selected diameters will be inserted between the sections to measure the pressure drop for various flow rates and orifice diameters.

Superheater Vibration Test. Vibration test planning is now in progress. The purpose of this test is to determine the spacing of tube sheets, so that vibrations detrimental to the superheater tubes, due to moderator cross and parallel flow, will be minimized.

2.2. RECIRCULATION SYSTEM

2.2.1. Recirculation Pumps

Manufacture of the prototype pump is continuing. Late delivery of the

pump casing and impeller from the steel foundries may delay the completion date and test schedule.

2.2.2. Recirculation Piping Loops

The recirculation piping stress analysis is complete. The results show that the maximum stress in the piping is at the junction between the clad carbon-steel pipe and stainless steel pipe. The joint, besides being stressed by differential axial expansion, internal pressure, and weight, is also subjected to a bending stress caused by differential radial expansion of the two metals. The maximum thermal stress acting in the piping is 13,097 psi. The allowable stress limit is 23,625 psi. The maximum longitudinal stress due to internal pressure, weight, and other sustained external loadings is 6,161 psi. The allowable stress in the hot condition is 15,750 psi.

The recirculation pipe reactions increase and decrease the individual loading on each vessel support column. An increase in the maximum column load of 7.2 per cent will result.

The vibration analysis of the loop piping was re-evaluated. Since the pumps will vibrate with only a small displacement in the lateral direction, it was concluded that the piping should be considered as rigidly fixed at both ends. The vibratory deflection curve would then approximate the static deflection curve for a beam built-in at each end. The natural frequency of the recirculation piping for this case is at least twice the rpm of the pump. Therefore, there should be no resonance difficulties during any phase of operation.

2.2.3. Recirculation Loop Valves

To obtain a more accurate curve of sigma versus valve angle at incipient cavitation, data from tests in which no air was visible were plotted. The test band for the 8-inch i.d. test valve and the calculated sigma for the 20-1/2-inch i.d. recirculation valve at reactor conditions are shown in Figure 2.2. The calculated sigma for the 20-1/2-inch valve is based on three-pump operation with all valves at the same angle. This curve indicates that incipient cavitation will occur in the 20-1/2-inch valve at an angle of approximately 46.5 degrees. The limits of the control range are therefore from 90 to 46.5 degrees, with corresponding flows of 21,167 and 16,000 gpm, respectively, for each Pathfinder recirculation loop.

Testing is complete on the 8-inch i.d. butterfly valve, and the final test report is being written.

A preliminary design proposal for the butterfly valves and operators was received from the Allis-Chalmers York Works.

2.3. STEAM SEPARATORS

A full-scale model of the steam separator is being tested in a test loop. Since forty-five separators will be used in the Pathfinder reactor vessel, the test loop was built to accommodate one-forty-fifth of the total reactor flow. A water-air mixture is being used to simulate the steam-water mixture that will be present in the reactor.

The test loop is shown in Figure 2.3. Water is recirculated by a 12 x 12 type SE Allis-Chalmers pump rated at 4000 gpm and 80 feet of head. The flow is controlled by an 8-inch gate valve. The diagonal pipe section,

which directs flow into the standpipe, contains a 30-kw immersion heater. Air is introduced into the water at a point about halfway up the diagonal pipe and is regulated by a 2-inch gate valve.

The standpipe section has a 19-3/4-inch i.d. The diagonal pipe discharges the air-water mixture through a baffle sleeve into the standpipe. The top of the sleeve simulates the top of the boiler core and separates the inlet section from the downcomer section.

The separator fits into a support plate at the bottom of the standpipe. Therefore all recirculating water must pass through the inlet nozzle of the separator. The separator is supported near its top by two bolt clamps, which are inserted in brackets mounted inside the standpipe. These supports are similar to those that will be used in the reactor.

The standpipe cover has a 2-inch pipe coupling welded into it to allow air to discharge from the simulated steam dome. The cover is also provided with a 2-inch pipe through which air is exhausted from the separator. Ten 6-inch Plexiglas portholes are spaced along the standpipe to allow visual observation of the flow.

As the air-water mixture flows through the separator, a vortex is formed. Separation of air and water takes place in the vortex and the water passes to a diffusion tank. The diffusion tank is slightly inclined so that any air that is carried-under will rise and travel to the downstream end. An air collection drum is at the downstream end. A sight glass on this drum shows the amount of air collected, and a vent on the top of the drum allows the collected air to be periodically released.

The instrumentation for the loop is designed to measure fluid flow rates and pressure drop throughout the system. The water flow rate is measured by means of a sharp-edged orifice located on the upstream side of the 8-inch gate valve. Air flow rates are measured by orifices in the air supply line, the separator exhaust line, and the steam dome discharge pipe. The air carried-under is determined by timing the rate of water displacement as indicated by the sight glass on the collection drum.

Pressure taps are located on the steam separator and in the standpipe so that total pressure drop through the separator can be measured as well as the separator inlet pressure drop. The separator outlet pressure drop can be obtained by subtracting from the total drop or by a direct measurement. Static pressure is measured in the air supply line and in the collection drum. One pressure tap in the standpipe is used to indicate the static water level.

The flow areas in the standpipe are proportioned so that velocities in the standpipe will duplicate those in the reactor, in particular, the outlet flow from the baffle sleeve and through the downcomer region around the separator.

In the reactor, the separators will exhaust steam to the steam dome, which is at the same pressure as the pool surface around the separators. During tests, the air pressure in the separators and simulated steam dome differ by about 1 psi. Tests indicate, however, that this differential has a negligible effect on the efficiency of the separator.

The test procedure used to simulate various reactor steam release rates and recirculation flow rates is as follows:

- 1) The water is heated to 170 F.
- 2) The water flow rate is set to simulate a reactor water recirculation flow rate.
- 3) Data is taken at various air flow rates simulating variations in reactor power.

To simulate various reactor water levels, the following test procedure is followed:

- 1) The static water level in the standpipe is set and maintained at a desired point.
- 2) The water is heated to 170 F, and the water flow is set to a desired rate.
- 3) Data is taken at several air flow rates.
- 4) The test is then repeated at other water levels and recirculation flow rates.

Tests were run to determine the effect of air bubble size and static water level on the amount of air passing into the separator and the amount carried under. A screen with a 1/16-inch mesh was placed over the outlet of the baffle sleeve in some tests to reduce the bubble size.

Test results showed that without the screen, 70 per cent of the air passed into the separator and 30 per cent rose into the steam dome. With the screen in place, about 30 per cent of the air passed into the separator and 70 per cent passed to the steam dome. In either case, the amount of air carried under remained about 0.1 per cent by volume. A percentage of 1.0 is considered allowable.

Tests run at various static water levels showed that the static water level has little effect on the percentage of air separated. The amount of air carried under increased from 0.1 to 0.3 per cent as the water level was decreased to 1 foot above the inlet baffle.

Tests will continue with various sized screens to determine a correlation between bubble size and the percentage of air carried-under. Tests will be made to determine the effect of the water flow rate on the percentage of air carried under. The effect of an inclined perforated plate mounted above the baffle sleeve outlet will be evaluated.

2.4. CONTROL RODS, GUIDE TUBES AND CONTROL ROD DRIVES

2.4.1. Control Rods

All control rods for the critical facility were manufactured. During the manufacture of these rods, all welds were dye-penetrant tested for cracks and in no case was a rod found with a crack in the weld or adjacent area. Although a prototype rod was found to be within drawing tolerances, subsequently manufactured rods were found to be considerably worse than originally specified. Examination of the boron stainless plates from one of the vendors showed variations in thickness of from 0.190 to 0.272 inch for a nominal 0.250 inch plate. The fixture was designed to accommodate only standard mill tolerances and, therefore, all undersized plates were not held tight during welding. These plates had a tendency to warp.

Stress relieving a straight welded control rod at 1200 F for one hour and air cooling showed no dimensional change.

The design of a split boiler control rod previously reported⁸ was discontinued, because a recent refined shielding calculation indicates that radiation level at the top of the shield pool during rod removals will not be excessive.

2.4.2. Guide Tubes

A prototype Zircaloy 2 guide tube was ordered for an elevated temperature control rod drive test. This test will use a boron stainless control rod moving in this Zircaloy 2 guide tube. The test will be used to determine the effects of misalignment on rod operation, galling characteristics of the two materials, clearance between rod and tube, and straightness requirements for the rod and guide tube.

The reactor boiler guide tubes will be constructed in two pieces. The lower half, which is in the core, will be a Zircaloy-2 cruciform envelope, which will be bolted to the lower grid plate. The upper tubes will be stainless steel cruciform envelopes which will be welded together in a single structure. This structure will be removed during refueling operations.

Tensile tests on 2 w/o boron stainless samples which were micro-brazed together were conducted. Half of the pieces were butt-brazed through the gage length while the remainder were brazed longitudinally through the gage length. During the machining of the specimens to specifications shown in Figure 2.4, all the butt-brazed pieces broke in the joint. Examination showed a very poor braze, which did not penetrate the joint. The specimens which were brazed longitudinally were pulled and the results are given in Table 2.2.

8. Pathfinder Atomic Power Plant, Technical Progress Report, July 1959 - September 1959, ACNP 5924, p.39, 1959.

TABLE 2.2 MECHANICAL PROPERTIES OF 2 w/o BORON STAINLESS STEEL*

Sample No.	Temp F	Tensile Strength psi	Yield Strength psi	Elong. per cent	R.A. per cent
1	70	72,550	40,000	4.0	4.8
2	70	70,100	46,000	3.5	2.1
3	500	62,000	-	3.5	5.0
4	500	60,800	-	3.0	6.4
5	1000	56,400	-	3.0	6.7
6	1000	57,500	-	3.5	5.0

*Test bar material was pulled parallel to the direction of rolling. Before the welded boron stainless samples can be pulled, the grippers have to be reworked. This is now being done, and the test results will be reported in the next quarterly.

2.4.3. Control Rod Drives

2.4.3.1. Design Status. Conceptual Design. A water supply system for the control rod rack housing during scram is being designed.

2.4.3.2. Tests. Control Rod Rack-Housing Heat Transfer Tests. The heat transfer test was completed, and the data obtained is being analyzed. Analysis thus far indicates the following:

1) Any method of supplying water for rack displacement during scram by direct connection between the rack housing and pressure vessel results in a thermal siphon that results in excessive heat loss to the shield pool. Neither of the methods previously proposed, i.e. providing penetrations through the bushing or providing a line that passes reactor steam directly to the top of the rack housing, can be incorporated into the control rod drive.

2) Baffles in the section between the bushing and the drive pinion housing effectively reduces the heat loss to the shield pool. The heat loss decreases to a minimum with increase in the baffle spacing and then again increases. During the test, the minimum heat loss was reached with about 1.5-inch axial spacing.

3) If a baffle is used and if the thermal siphon is eliminated, the heat loss to the shield pool will be substantially below 2.5 kw/drive.

Prototype Control Rod Drive. The following components for the prototype control rod drive were received: all special bearings, rotating shaft seals, gear material with exception of the rack, and the servomotor. An order was placed for the rack, pinion shaft, miter gears, and the Stellite No. 3 rack roller and bushing. The gear box for the servomotor was also ordered. A complete position indication system was ordered. It is of the synchro-type employing a transmitter located in the drive unit and a receiver and 8-inch clock-type dial located on the control panel. A specification for a quick-disconnect coupling to couple the drive to the vessel cover was sent out for quotations.

Tests of the induction-gear drive motor with a prony-brake load of 240 inch-pounds torque in air indicated that it would meet the specifications concerning starting, stopping, and reversal time. These times were set at 0.2, 0.1, and 0.2 seconds, respectively.

The tests were stopped to return the motor to the factory for installation of a magnetic disc brake. This was necessary since the original built-in brake was not reliable. The motor has now been returned and testing will continue shortly. The motor will be tested in both air and

oil using a weight, cable, and pulley arrangement to more closely duplicate the loading conditions.

Detail drawings were completed for the hot test rig and were submitted to the West Allis Shops for fabrication. This rig will be used to test the prototype drive at reactor temperature and pressure. The drive motor, latch, scram mechanism, and position indicator will be evaluated. The test rig will also include a full-sized control rod, upper and lower guide tubes, and four dummy boiler fuel boxes.

A dash pot test specification was prepared, and a test rig is being built. The test will be completed if possible before the final hot test begins.

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3. NUCLEAR ANALYSIS

3.1. REACTOR PHYSICS (STATICS)

3.1.1. Flux Peaking in the Boiler Core

A potential contribution to the nuclear hot spot factor in the boiler core region is the flux peaking between fuel bundles. There is a fuel gap of 1-1/2 inches between the 81- and 64-rod regions.

The Simpl I program using the double PI approximation was used to calculate this flux peaking for the hot reactor with 28 per cent voids. The result is plotted in Figure 3.1. The peaking factors, which are defined as the ratio of the maximum flux in the fuel to the asymptotic value, are 1.41 and 1.18 in the 81- and 64-rod regions, respectively.

To compensate for this power peaking, a fuel pellet of natural enrichment and/or poison material adjacent to the gap will be used.

Simpl I was also used to calculate the peaking at a control rod channel in the 81-rod region with 28 per cent voids. The result is plotted in Figure 3.2. The peaking factor for this case is 1.28.

3.1.2. Flux Peaking in Boiler Irradiation Test Subassembly

A 9-pin boiler test subassembly will be irradiated in the GETR. The presence of an aluminum spacer plug between fuel bundles within the element results in a local flux peak at the edge of these spacers. This peak coupled with an axial flux peak in the reactor results in an excessive

fission rate at this location. Therefore, a method of compensating for the local flux peak was necessary.

A thermally black material can be placed around the fuel element adjacent to the aluminum spacers to shield the adjacent fuel from neutrons thermalized in water outside of the fuel cell. Several hand calculations using diffusion theory were performed to verify that this shutter technique was adequate.

After verifying that a shutter would compensate for the flux peaking, the correct sized shutter was determined by running a series of two-dimensional PDQ cell calculations with various sized shutters in position. In these calculations, one group, thermal, diffusion equations were solved with the source term proportional to the water density within the various regions. Black boundary conditions were applied at the surface of the shutter. Flux traverses from these cell calculations were compared with one dimensional transport calculations with good agreement. In all cases, the transport calculation indicated that the amount of compensation from the shutter predicted from the PDQ calculation was conservative.

The final results indicate that a black absorber with the same length as the aluminum spacers would compensate for the local peaking.

3.1.3. Core Burnup

Control rod channels produce non-uniformities in the fuel lattice. With the control rod withdrawn thermal flux peaking in the channel causes a more rapid burnup of the adjacent fuel. However, because shielding of resonance energy neutrons by neighboring fuel rods is decreased, this adjacent fuel undergoes higher plutonium production than that in the center of the fuel

box. The variation in power peaking with time due to non-uniform production of plutonium and depletion of fissionable fuel was calculated.

The variation in resonance integral due to the shielding of resonance energy neutrons by neighboring fuel rods was calculated for a pin within the center of a fuel cluster and for a fuel pin at the corner of two control rod channels. The greatest variation in resonance integral for this comparison would occur in the upper, 81 fuel rod region, because of the high void content within the fuel cluster. Therefore, the variation for that region was calculated with the equations,

$$RI = A + B\left(\frac{\delta}{r_0}\right)(1-\delta) \quad (3.1)$$

and

$$\delta = \sum C_n \quad (3.2)$$

where,

$$C_n = .1277e^{-1.76\left(\frac{d_0}{r_0} - 2.05\right)} r_0 \sum s \quad (3.3)$$

δ = Dancoff correction due to shielding by neighboring fuel rods.⁹

d = distance between fuel rods

r_0 = radius of fuel rod

From the results of these calculations the plutonium production rates at these two positions were calculated using the equation,

$$\text{Plutonium Production Rate} = \phi_{th} \nu \frac{(1-P)}{(1-\epsilon_{res} B^2)} + \sum_{28}^{28} \phi_{th} \quad (3.3)$$

The first term on the right hand side of Eq. 3.3 represents resonance captures. The probability of resonance capture $(1 - P)$ times the relative

9. Dancoff and Ginsberg, CP-2157.

number of resonance neutrons produced yields the relative number of resonance captures. The same average thermal flux for the fuel cell was used for this term for both positions because the fast flux is relatively constant across the cell. The thermal flux obtained from a two dimensional XY calculation was used for these calculations at the two positions. The increased thermal flux due to the water channel peaking was used for the second term.

The isotopic equations,

$$N^{25}(r,t) = N_0^{25} e^{\sigma_a^{49} \phi_{th} t} \quad (3.4)$$

$$N^{Pu}(r,t) = \frac{\text{total Pu prod.}}{\sigma_a^{49} \phi_{th}} (1 - e^{-\sigma_a^{49} \phi_{th} t}) \quad (3.5)$$

were then solved as a function of time for the two fuel positions. The plutonium production rate used in Eq. 3.5 was that obtained from Eq. 3.3 and was assumed constant in time.

The number densities obtained from these equations were then summed to yield Σ_f as a function of time at the two fuel locations. This gives:

$$\Sigma_f(r,t) = N_{25}(r,t) \sigma_f^{25} + N_{Pu}(r,t) \sigma_f^{Pu} \quad (3.6)$$

which is plotted in Figure (3.3). Results indicate that the power peaking adjacent to the control rod channel decreases with fuel burnup when the control rod is withdrawn. Further calculations will be performed to keep this peaking less than the initial value as control rod insertions create more non-uniformities in fuel burnup.

3.1.4. Neutron Streaming in the Superheater

An investigation of the effects of neutron streaming in the superheater steam passages was made. This work was based upon a method developed by Behrens.¹⁰

The following equations, developed in the reference given, were used:

$$\frac{\Lambda^2}{\Lambda_0^2} = 1 + 2\phi + \phi^2 (2r/\phi\lambda) \left[\frac{1}{e^{2r/\phi\lambda} - 1} \right] + \frac{3}{2} Q \frac{r\phi}{\lambda} \quad (3.7)$$

and

$$\frac{\Lambda^2}{\Lambda_0^2} = 1 + 2\phi + \phi^2 \left(\frac{2r}{\phi\lambda} \right) \left[\frac{1}{e^{2r/\phi\lambda} - 1} \right] + \frac{3}{4} Q \frac{r\phi}{\lambda} \quad (3.8)$$

where,

Λ_0 = root mean square distance traveled by a neutron in N collisions in absence of holes.

Λ = rms distance traveled by a neutron in N collisions in presence of holes.

ϕ = the volume ratio, $\frac{\text{holes}}{\text{material}}$

$v\phi$ = the volume of each hole

S = the surface area of each hole

λ = the mean free path of a neutron through the solid material of the reactor

$r = 2v\phi/S$, "hydraulic radius"

Q = ratio of mean square passage length through hole to square of mean passage length.

The Eq. 3.7 & 3.8 apply to the case of parallel, equally spaced and sized cylindrical holes. Eq. 3.7 applies to the direction of orientation of the

10. P. J. Behrens, "The Effect of Holes in a Reacting Material on the Passage of Neutrons, with Special Reference to the Critical Dimensions of a Reactor," AERE T/R 103, 1958.

cylinders, and the Eq. 3.8 to the direction transverse to the cylinders.

Adjusted diffusion coefficients were taken as:

$$D = D_0 \sqrt{\frac{\Lambda^2}{\Lambda_0^2}} \quad (3.9)$$

where D_0 is the diffusion coefficient for the homogenized superheater material.

The following results were obtained:

Energy Group	D (longitudinal)	D (radial)	D (Homog.)
Fast	2.24	2.21	2.21
Thermal	.476	.459	.467

The reason for small changes in D resulting from the steam passages is primarily due to the small size of the superheater elements.

This investigation indicates that the diffusion coefficients obtained by homogenizing the steam passages with the superheater material are adequate.

3.1.5. Partial Superheater Fuel Loadings

Analyses were made of reduced fuel loadings in the superheater. The results, based on one and two dimensional diffusion calculations, are given in Table 3.1.

TABLE 3.1.

Fractional Superheater Loading	Boiler Modification	Reactivity Change Percent $\Delta k/k$	Superheater Power, Full Superheater Loading = 100 per cent	Additional Hot Spot Factor Superheater	Additional Hot Spot Factor Boiler
3/4	None	-2	63	1.07	1
1/2	None	-3.3	36	1.19	1
0	None	-4.7	0	1	1
0	Outer 32 boiler elements spiked to about 3 per cent.	0	0	1	1.22

The resulting steam temperature, superheater and fuel hot spot temperature, superheater power, and turbine generator output are shown in Figure (3.4).

A most significant result in Table 3.1 is the low nuclear importance of the superheater fuel. This may not be immediately obvious because of the central location of the superheater. However, the values of k_{eff} for the boiler and superheater are about 1.28 and 1.08, respectively. This causes the thermal neutron flux and the adjoint fluxes to be depressed in the superheater region. Radial plots of the thermal flux and the fast and thermal adjoint fluxes at a location midway up the core are given in Figure (3.5). These are based on four-group diffusion calculations and are presented as typical of the radial distributions. These curves provide a qualitative explanation for

the low worth of the superheater fuel. It can be seen that an absorber is about four times as effective when located at the flux peak in the boiler as when located at the reactor centerline. The relative worth of a fission neutron is approximately a factor of two for the same locations.

Complete two-dimensional flux and adjoint profiles were used to evaluate the statistical weight and total extrapolated fuel value in the superheater. In the four group model, with all fission neutrons born into the first group and all fissions occurring in group four, these quantities are defined and evaluated as follows.

The statistical weight of the superheater for a thermal absorber is

$$\frac{\int_{\text{superheater}} \phi_4 \phi_4^+ dV}{\int_{\text{Reactor}} \phi_4 \phi_4^+ dV} = 0.117 \quad (3.10)$$

and the extrapolated total fuel value in the superheater is

$$\frac{\Delta k}{k} = \frac{\int_{\text{superheater}} \left[\Sigma_f [\phi_4 \phi_1^+ - \frac{(1-\alpha)}{\nu} \phi_4 \phi_4^+] dV \right]}{\int_{\text{reactor}} \Sigma_f \phi_4 \phi_1^+ dV} = -0.064 \quad (3.11)$$

where,

The integrals are over the volume of the superheater in one case and both superheater and boiler in the other,

and where,

ϕ_4 is the fourth group (thermal) neutron flux

ϕ_1^+ and ϕ_4^+ are the first and fourth group adjoint fluxes.

$\Delta k/k$ is the reactivity change

Σf is the macroscopic fission cross section in group four (corrected for fast fission)

α is the capture to fission ratio in the fuel

ν is the number of neutrons released per fission (equals 2.47).

The statistical weight of the superheater is presented here to show its low value, even in the central location. The extrapolated fuel value is the reactivity loss that would occur if all of the fuel were removed from the superheater based on the worth of the fuel in the fully loaded core. Since the worth of the fuel decreases as some is removed (lowering multiplication in the superheater, thus flux and "importance"), this value, -6.4 per cent, is a definite upper bound on fuel worth. As such, it corroborates the diffusion theory result of -4.7 per cent.

The limiting case of no fuel in the superheater was studied further to establish the capability of operation solely as a boiling water reactor. Various combinations of spiked elements were considered in order to compensate for the reactivity lost with the fuel removed from the superheater. To keep the number of additional elements small and still not increase the hot spot factor seriously, an enrichment of 3 per cent for the outer row of 32 elements was selected.

As indicated in Table 3.1, this compensates for the reactivity loss and only increases the hot spot temperature by a factor of 1.22. The reactivity worth of the boiler control rods in this case is adequate to meet the control requirements.

3.1.6. Alternate Superheater Control Rods

A uniform array of 57 small cylindrical control rods designed to fit into the fuel element tubes was evaluated for the superheater region. The rods were

grouped so as to utilize the four reference control rod drives. The advantages of this design are as follows:

- 1) The hot spot factor due to partial rod insertion (for a desired flux shift) in the superheater is reduced.
- 2) The hot spot factor due to peaking around the control rod guide tube with the rod removed is reduced.
- 3) Thin cylindrical stainless steel guide tubes can be used instead of the thick guide tubes required with the cruciform control rods. This removes considerable parasitic absorption from the superheater which increases the infinite multiplication constant of the hot, clean superheater from 1.08 to 1.14.

To evaluate the effect on control rod worth and flux peaking, two dimensional PDQ calculations were performed. These calculations were made for an XY plane perpendicular to the vertical axis of the core at the location of the maximum superheater fuel temperature. This was about fifteen inches above the bottom of the core. Calculations were made for the core with no rods inserted in either the boiler or superheater cores, and with two rods (28 of the small rods) inserted in the superheater core and none in the boiler core. Calculations were made for both types of control configurations.

Black boundary conditions were applied at the surface of the control rods. A flux traverse through a control rod obtained from the PDQ calculation was compared with a P-3 transport calculation through a control rod with excellent agreement.

Results of the above calculations are summarized in Table 3.2.

TABLE 3.2.

Superheater Core	Hot Voided k_{eff}	Max. Flux in Superheater	Percentage of Original Superheater Power after Control Rod Insertion	Control Worth $\Delta k/k$
No Cruciform rods inserted	1.080	1.14		
No Cylindrical rods inserted	1.089	1.09		
Two Cruciform rods inserted	1.071	1.32	82.2	-.009
28 Cylindrical rods inserted	1.066	1.46	66.9	-.023

3.1.7. Low Enrichment Superheater Element

Nuclear parameters were calculated for the low enrichment superheater element shown in Figure 3.6. This element consists of a group of seven fuel rods centered in a stainless steel sheath. The fuel rods are uranium dioxide pellets 0.214 inches o.d. with a 0.010-inch thick stainless steel clad. The assembly is enclosed in a 0.020-inch thick Zircaloy-2 wall 1.080 inches o.d.

Fast and thermal constants were calculated for several values of enrichment and several fuel-to-moderator ratios. Fast constants were evaluated using the Muft IV code, and thermal constants were calculated with the Sofocate code. Figure 3.7 shows a plot of K_{∞} vs center-to-center element spacing in a rectangular lattice.

3.1.8. Shielding

Streaming of gammas down the straight air duct at the bottom of the shield pool was calculated. The pipe used in the calculation is 3 feet in diameter, 10 feet long, and filled with air. The dose due to streaming at the end of the pipe is 11 R/HR. The dose is such that if air cooling is used for the shield, a straight duct cannot be used at this location.

The streaming of gammas through the recirculation pipe tunnels was calculated for two cases: when the pipe tunnel is just large enough for the recirculation pipes, and when the tunnel is extended down to the pump floor. The dose point is just in front of the pump and on the platform surrounding it. The gamma dose is 10 R/HR for the first case and 70 R/HR for the second case, assuming the reactor is operating at full power. Neutron streaming is much smaller than the above and was not included. This dose due to streaming is approximately the same as the dose from deposition of crud and N^{16} in the recirculation pump and piping.

The latest turbine support design supplied by Pioneer Service & Engineering was evaluated for shielding purposes and found to be adequate. However, a 2 foot concrete wall is required between the water box and the basement floor on the cool side of the condenser and extending for the entire width of the condenser. This wall is necessary to provide shielding from the N^{16} between the condenser tube area and the condensate. This wall need not be permanent and could be made of removable concrete block.

The shadow shielding for the access doorway to the feed water heaters on the basement floor was extended 2 feet in length in order to properly shield the passageway from feed water heater number 1. The center section of the

doorways between the turbine building and the fuel handling building was increased from 1 foot to 2 feet to adequately shield the fuel handling building from feed water heater number 4. There is also a 1 foot concrete slab over the doorway in the turbine building on the mezzanine floor to give adequate protection from the main steam line to a point just inside the fuel handling building. A 1 foot concrete slab over the passageway between the two buildings on the basement level shields the passageway from the purification system line from the reactor. The air ejector line to the stack is looped in the turbine building to get a 60 second delay. This delay provides a sufficient decay of N^{16} so that no shielding will be required for this pipe in the fuel handling building.

A new shielding program 04-2 was obtained. In general the new code is about the same as the 04-0 code with the following additional features:

- 1) Tape units are not required.
- 2) In addition to the cosine source distribution, an exponential source can also be used. These distributions are of the form:

$$S_{(z)} = t_1 \exp [t_2 (Z_S - t_3)] + t_4 \quad (3.12)$$

or

$$S_{(z)} = t_1 \cos t_2 (Z_S - t_3) - t_4 \quad (3.13)$$

These equations can be used in both the radial and axial direction. The source distributions may be described in as many as four ranges between the limits of integration.

- 3) Unequal as well as equal spacing of shells, planes, and lines in the source can be used.

A data check routine was obtained which will detect any bookkeeping errors in the preparation of data for these shielding codes. The 04-2 shielding code will be used for future shielding calculations.

3.2. REACTOR AND SYSTEM DYNAMICS

The dynamics analysis continued with work on the Pathfinder simulator and the analysis of several control schemes. The basic simulator¹¹ structure is essentially complete and correct. Sections of the simulator will be reviewed and changed, however, as more information becomes available with respect to hydraulics and heat transfer, and as design modifications are made.

Further analysis of the CRBR hydrodynamic characteristics was made, and the latest available information is being used for numerical evaluation of constants. In this analysis, the primary system was divided into two nodes; one system being the steam-water mixture in the core and steam dome plus the frothy mixture between, and the second system being the remaining fluid in the separators, lower plenums, and recirculation pipes and pumps.

The differential equations derived herein are non-linear due to the velocity-squared dependency of pressure drop and to the void volume dependency of the core friction factor. For the purposes of hand solutions, the equations are linearized and solved in terms of increments of the variables. For simulation on the analog computer, the essential non-linear terms are kept so that a more general and accurate solution can be obtained.

11. Pathfinder Atomic Power Plant, Technical Progress Report, July 1959 - September 1959, ACNP 5924, Figure 3.12, 1959

The above function will be included in the CRBR transient simulator during the next quarter along with several other changes and improvements.

A change was made in the vessel hydraulics function in the CRBR Transient Simulator. The variation of primary system internal energy during a transient should now be better described thus yielding a more accurate energy to pressure transfer function. The pressure response is directly associated with the recirculation time which is about 12 seconds for full power flow. If enough analog equipment is available, additional improvements will be made by a better simulation of the perfect lags inherent in the differential equations.

The superheater analog in the CRBR Transient Simulator is working well, and the results for a few cases have been hand checked for accuracy. The output information now available is superheater inlet and exit temperatures (bulk average) and hot channel steam and cladding temperatures at the hot spot. The heat flux at the hot spot is also available. The necessary inputs to the superheater analog are reactor pressure, superheater neutron level, and superheater steam flow. The following factors are incorporated:

- 1) second order lag representation of transfer function relating superheater heat generation and heat flux across fuel cladding,
- 2) temperature-dependent specific heat of steam in superheater,
- 3) heat transfer coefficient at hot spot varies with local steam flow rate and steam film temperature.

Due to the non-linearity of the equations involved, several steady-state values of parameters (constants) are necessary as initial conditions. These

are determined by the ASTER code,¹² which determines superheater temperatures and pressure drop for arbitrary input at constant reactor power. It is anticipated that the transient output of the superheater analog can in turn be fed into the ASTER program which will then yield detailed superheater information by solving a series of "steady-state" points.

Several curves taken on the simulator are shown in Figures 3.8, 3.9 and 3.10.

In Figure 3.8 a comparison is shown for steam dome and steam line pressure control. No automatic reactivity control was used. The reactor was operating at full power and a control rod was withdrawn, inserting external reactivity at + 5 per cent/second for 10 seconds. As can be seen, superheater fuel temperature is very sensitive to changes of steam flow rate and power in the superheater.

Figure 3.9 shows a condition in which the reactor was operating at full power and the electrical output demand was changed manually 0.1 per cent/second for 20 seconds. The demand signal was simulated by a manual change of recirculation flow rate in the reactor via the butterfly valves. The response time of the system is 30 seconds for steam line control and 14 seconds for steam dome control with the same pressure controller settings.

Figure 3.10 shows the system response for a turbine trip valve closure. The trip valve was assumed to close in 0.3 seconds while the main bypass valve opening time was also 0.3 seconds. The bypass valve was assumed to start opening 0.1 second after the trip valve started to close causing a high degree of valve mismatch, δS_c .

12. Ibid, p 19.

S_c is the flow coefficient for the turbine admission valve. It is defined by $W_T = KP_2S_c$ where W_T is turbine steam flow rate; P_2 is the turbine inlet pressure; K is a constant.

Other problems investigated were a cold water slug accident determination of feedwater valve opening rate, determination of response to an admission valve change, and comparisons of various controller gain constants for proportional plus reset action.

Little progress was made on the space-time dependent flux study due to lack of time. However, a better understanding of the complexity of the situation was obtained. Three general methods of study seem to be: 1) a series of statics solutions, 2) a nodal analysis employing nodal kinetic equations, and 3) a topological study. The first method seems to be most feasible.

3.3. CRITICAL EXPERIMENTS

The erection of the Allis-Chalmers Critical Facility was completed. Complete preoperational checks and tests were run on the nuclear instrumentation and controls, the control rod drives, and the moderator system. All components of the reactor system were found to be functioning properly with the exception of the snubbing mechanisms on the control rod drives. Although the difficulty did not affect the safety of operation, these mechanisms have since been reworked because of the possibility of physical damage to the control rods and drives.

The delivery of boiler fuel rods for the critical facility was completed. The total complement of fuel rods on hand for the critical experiments is as follows:

<u>Enrichment (per cent)</u>	<u>Number of Rods</u>	<u>Color Code</u>
normal	11,800	gold
1.8	28,400	silver
2.6	3,048	red

The uranium dioxide loading of each fuel rod is approximately 260 grams at a density of 10.42 gr/cc. The total thermal neutron cross section of all impurities is less than the equivalent of 1 ppm of natural boron in uranium dioxide. The cladding material is aluminum alloy X8001. The nominal fuel rod dimensions are:

Fuel pellet diameter	0.338 in.
Total fuel length	17.0 in.
Cladding i.d.	0.341 in.
Cladding o.d.	0.418 in.
Total rod length	17.25 in.

An inspection of the critical facility and a review of the operating procedures and regulations were made by the AEC for purposes of issuance of an operating license. An operating license, CX-15, was issued effective November 3, 1959.

The loading of the 1.8 per cent enrichment boiler fuel into the reactor commenced after issuance of the operating license and initial criticality was achieved on November 11, 1959.

3.3.1. Initial Critical Experiment

Initial criticality was achieved with the fuel element loading shown in Figure 3.11. This loading consisted of 19-7/9 boiler fuel elements with a total U-235 content of 26.4 kilograms. Each fuel element was loaded with

81, 1.8 per cent; fuel rods (9 x 9 array) at each of the four vertical bundle locations. This gave a center to center rod spacing of 0.554 inch, on a square lattice, and a total full height of 72 inches. Of this total height, the three separations between bundles accounted for about 4 inches of non-fuel height.

The excess reactivity associated with this configuration was measured to be 0.09 per cent $\Delta k/k$, corresponding to a reactor period of 84.5 seconds.

The loading of elements proceeded as follows: 2, 2, 2, 2, 2, 2, 2, 2, 1, 1, 5/9, and 2/9, to establish a critical configuration. The monitoring of the approach to critical was done by the neutron sensitive detectors shown on Figure 3.11. In addition to the BF_3 and fission counters and the CIC, a miniature fission chamber was placed in the active lattice of the first element loaded.

Plots of inverse count rate versus the fuel loading are shown in Figures 3.12 and 3.13 for the water up-rods out condition. Figure 3.12 shows the data, as taken from the miniature fission counter, from the 12 element loading to the 19 element loading. The data taken from the linear power channel, the standard fission counter and the miniature fission counter is depicted in Figure 3.13 for the 19 element to the critical loading.

3.3.2. Comparison with Theory

The critical loading shown in Figure 3.11 was arrived at because of the desirability of keeping as good a source fuel and chamber geometry as possible and of placing elements around the safety and regulating rods first. No attempt was made to build the core into a regular geometric shape. After this critical loading had been achieved, a new critical configuration of

more regular geometric shape was constructed. This lattice, as shown in Figure 3.14 consisted of 18-3/9 elements, 5 elements wide by 3-6/9 deep, loaded as described previously. This configuration was just critical, that is, had no excess reactivity, with full water height and all control rods fully withdrawn.

A two group, two dimensional diffusion theory calculation was performed on this configuration using the nuclear constants described in Section 3.1. The effective multiplication constant obtained from this calculation was 0.991, a difference in k of -0.009 from that measured for the configuration.

3.3.3. Experimental Reproducibility

In order to determine the reproducibility of the critical rod position after normal occurrences such as shut downs, scrams, and control rod bridge removal, six reproducibility runs were made. Between each run, the rods were unlatched, the control rod bridge was removed and replaced, and the rods relatched.

The results of these critical runs are summarized in the following table. The reactivity errors involved are, of course, dependent upon the rod worth at the position of measurement.

Critical Rod Position	Differential Worth
55.91 inches	$1.6 \times 10^{-4} \Delta k/\text{inch}$
Probable Error	Probable Error
± 0.12 inch	$\pm 0.2 \times 10^{-4} \Delta k$
Maximum Deviation	Maximum Deviation
± 0.26 inch	$\pm 0.4 \times 10^{-4} \Delta k$

Significant improvement in the reproducibility of critical rod positions, is noted during the same reactor run, that is, in absence of the disturbing effects of unlatching rods and/or moving core components.

3.3.4. Temperature Coefficient of Reactivity

The temperature coefficient of reactivity was measured for the core described in Section 3.3.2, except that the width was increased to a 3-7/9 element thickness to obtain some operating excess reactivity. The range of temperature was 60°F to 165°F, as measured in the moderator. Since the heating of the moderator was a lengthy process, and was done in the reactor tank, the fuel, cladding, and structural components were at the same temperature as the moderator. Constant agitation assured uniform temperatures throughout the core and absence of bubble formation on fuel and structural surfaces.

The results of these measurements are shown in Figure 3.15 where $\Delta k/^\circ\text{F}$ is plotted as a function of temperature. Because the reactivity changes involved are fairly small, and the points shown are the result of reactivity differences between period measurements at different temperatures, there is some degree of scatter to the data. Assuming a linear relationship between reactivity and temperature over this range to be a fair approximation, a least square fit of the data yielded $\Delta k/^\circ\text{F} = -0.42 \times 10^{-6} T + 2.39 \times 10^{-6}$. This gives a zero temperature coefficient at 57°F and a coefficient of $-0.67 \times 10^{-4} \Delta k/k/^\circ\text{F}$ at 165°F for a β_{eff} of 0.007.

A theoretical calculation of the temperature coefficient, in this temperature range and for the specific configuration used in the measurement, is presently underway. These results will be given and compared with the measurements in the next quarterly progress report.

3.3.5. Flux Measurements in 9 x 9 Slab Core

Measurements of the neutron flux distribution in the lattice described in Section 3.3.2 were made using bare and cadmium covered U-Al foils and

Cu-Mn wires. The locations at which these traverses were taken is shown in Figure 3.16. An axial traverse was taken near the center of the core and horizontal traverses were taken in the north-south and east-west direction, as shown. Considerable detail of the flux shape was measured near and in water channels, where no fuel was present, such as between fuel bundles, in control rod channels, and in the reflector.

The flux distributions shown have not been corrected for cadmium ratio, so they represent the total neutron spectrum activation flux.

The axial flux distributions are shown in Figures 3.17, 3.18, and 3.19. Figure 3.17 is a total traverse, as measured with U-Al foils, and includes detail in the reflector and between the middle fuel bundles. Additional detail between the central bundles and at the bottom reflector-core interface is shown in Figures 3.18 and 3.19, respectively. These flux measurements were made with Cu-Mn wires.

The flux distributions in the N-S and E-W directions (Figure 3.16) are plotted in Figure 3.20 and 3.21, respectively. Additional detail in the reflector and a control rod channel is given in Figures 3.22 and 3.23, respectively. All of these plots, except as noted, were obtained from U-Al foil data.

The results of these measurements are presently being used to verify diffusion and transport calculations of the lattices used in these measurements. Because of the small geometry and the cold clean lattices used in these measurements, no hot spot information pertinent to the Pathfinder design can be directly inferred from the results.

4. CHEMISTRY

4.1 CORROSION

The effect of dissolved oxygen on corrosion in Pathfinder during normal and unusual operating conditions is being studied. Special attention is being given to steels that are subjected for long periods of time to stagnant oxygenated water. Studies have shown that stainless steels are about twenty times as resistant to such oxygenated environments as low alloy steels.

The test condenser at the Lawrence station was placed in operation in November. Eight aluminum alloys are being tested in treated cooling water. Aside from a few minor equipment modifications, the condenser has operated satisfactorily.

Analytical reports for the test condenser cooling water were received. During approximately two weeks in November, total phosphate and pH were outside the specified limits. Such difficulties are to be expected with the water from the river, because of intermittent contamination from several sources upstream. The resulting variable composition of the water source makes it difficult to maintain a constant composition cooling water. In spite of these difficulties, a marked and constant improvement was shown in maintenance of cooling water composition during December. By the third week of December, phosphate and pH were within specified limits essentially 100 per cent of the time.

4.2. RECOMBINER, RADIOACTIVE GAS REMOVAL AND STORAGE SYSTEM, ION EXCHANGER, AND CHEMICAL PROCESSING

4.2.1. Radioactive Gas Removal System

The present radioactive gas removal system effects a delay time of about 10 minutes and dilutes the off-gas with ventilation air, which is exhausted at a rate of from 25,000 to 60,000 cfm. These means are expected to safely disperse reactor off-gas into the atmosphere.

If large amounts of fission product gases are frequently present in the off-gas, ventilation air may not sufficiently dilute the off-gas. An auxiliary gas removal system would then be necessary.

Two types of auxiliary off-gas disposal systems appear feasible for the Pathfinder and are presently under study. One system would use a charcoal bed to adsorb fission product gas. The second system would use a low-temperature liquefier.

New information received by Pioneer Service and Engineering Co. from Oak Ridge National Laboratory indicates that an auxiliary off-gas decontamination system using a charcoal adsorption system can be designed with present information. Preliminary estimates indicate that a bed containing 7300 pounds of Columbia Grade charcoal operating with a 98 per cent efficient recombiner would be sufficient to provide enough hold-up for stack discharge.

The low temperature liquefaction system will require developmental work before its performance can be properly evaluated. Its future use appears promising, since it should require less space and maintenance than

the charcoal adsorber system. Considerable effort is being expended in developing this type of auxiliary system for the Pathfinder.

A study was conducted to determine the effect of various stack design parameters on the degree of atmospheric dispersion obtainable at the Pathfinder plant. Among the parameters studied were the wind speed at the point of discharge, the variation in ventilation exhaust air flow rate, the stack diameter, and the ventilation fan exhaust pressure. Plume downwash under any combination of atmospheric conditions was studied.

Pioneer Service and Engineering designed the stack considering these factors.

4.2.2. Recombiner

A catalytic recombinder may be installed after the air ejectors in the off-gas line. The dual-Langmuir correlations were applied to determine the approximate size.

Since the gas from the air ejectors may be an explosive mixture of hydrogen and oxygen, approximately 18 pounds/minute of 100 psia reactor steam will be used to dilute the mixture to 2 moles per cent of hydrogen.

To insure that the catalytic bed is dry at all times, the 100 psia reactor steam will be superheated to 325 F.

Calculations indicate a catalyst bed about 6 inches in diameter and 3 inches deep is required to effect nearly 100 per cent recombination. This size would include a design safety factor of about 3. The recombinder shell will be about one foot long allowing additional space for catalyst, if required. The pressure drop across the bed will be about 1/2 psi with the outlet gas temperature at about 600 F.

4.2.3. Liquid Waste Disposal System

Evaluation of the present design indicates that 12,600 gallons/month of low-solids liquid wastes and 8,800 gallons/month of high-solids liquid wastes may be generated as a result of normal plant operation. The high-solids wastes will result from decontamination operations, laboratory drains, filter crud, and regeneration chemicals from the pool clean-up systems. Low-solid liquid wastes will result from all other sumps and drains throughout the plant except those that discharge directly into the drainage ditch.

A preliminary conceptual design of a liquid waste disposal system that would handle these wastes, and a system flow diagram were completed. The low-solids liquid wastes will flow to a 12,000-gallon decay-storage tank, where they will be monitored and sampled. The decay-storage tank will provide a decay time of at least 24 hours and a decontamination factor of 10. From the decay-storage tank, the low-solids wastes will be discharged to the drainage canal if activity is below maximum permissible concentrations (MPC) or will be processed by the waste disposal system demineralizer if the activity is above MPC. Spent resins from the waste disposal system demineralizer will be sluiced into the 12,000 gallon spent resin collection tank for decay and storage purposes.

The demineralizer used in the waste disposal system is the same size and type as is used in the reactor water purification system. The demineralizer will normally operate at 10 gpm. It can provide a decontamination factor of 10^2 in certain cases.

The high-solids liquid wastes will flow to the neutralizing tank, to the evaporator feed storage tank, and to the evaporator. The evaporator will process 85 gallons/hour and will provide a decontamination factor of 10^4 . The type of evaporator that will be used has not yet been selected.

A storage tank will be provided to store high purity water processed by the waste disposal demineralizer and the evaporator. This water, which will be re-used in the plant, will be below MPC and below 2 ppm solids.

A crud collection tank will store precoat filter crud material, evaporator bottoms, and neutralizing tank bottoms.

Interconnections between the waste disposal demineralizer and the evaporator permit recycling when further reduction of activity is required to achieve MPC. The liquid waste disposal system can process reactor water with an activity level of $3 \mu\text{c/ml}$ (nonvolatile) and reduce this level to MPC. Large volumes of high purity wastes can be accommodated by the waste disposal system by using the fuel storage pool as an emergency surge tank. The fuel storage pool demineralizers can process wastes arising from such an emergency at a maximum rate of 300 gpm.

Cooling tower blowdown will be used to dilute liquid wastes in the drainage ditch before discharge to the river. Discharge to the river will be less than the maximum permissible concentrations for non-occupational exposure. Although further dilution of liquid wastes will occur in the river, this dilution is not considered a decontamination factor in the design of the liquid waste disposal system. It will provide a safety factor, however.

The basic material used in the waste disposal system will be low carbon steel. Certain portions that contain corrosive liquids, e.g. evaporator, neutralizing tanks, piping, crud collection tank, evaporator feed storage tank, etc. may be made of 316 stainless steel. A study is being conducted to determine if aluminum piping may be used in the system to transport high purity water.

If an evaporator that is sufficiently reliable, safe, efficient, and flexible cannot be selected, the evaporator may be omitted from the liquid waste disposal system. The high solids wastes would then travel from the evaporator feed storage tank, acting as a surge tank, directly to the crud collection tank.

4.2.4. Solids Waste Disposal

The solid wastes generated in the plant include the following: resins, filter cartridges, fuel elements, fuel element ends, control rods, non-combustible wastes, combustible wastes, and chemical wastes.

Resins will be stored and decayed in the 12,000-gallon resin storage tank. It is estimated that 250 cubic feet of waste resin will be handled yearly. The 12,000-gallon tank will provide storage and decay for several years of operation. The resins will ultimately be shipped to a disposal site.

The filter cartridges will be shipped off site for disposal.

Fuel elements, fuel element ends, and control rods will be stored in the fuel storage pool to decay and then shipped off site for disposal.

The non-combustible wastes will include nuts, bolts, tools, equipment, glassware, etc. If these wastes cannot be easily decontaminated, they will be shipped in drums for off site disposal.

The combustible wastes will include paper, clothing, rags, brushes, mops, etc. The wastes with a high activity level will be shipped in drums for off-site disposal.

The chemical wastes will include evaporator bottoms, neutralizing tank bottoms, and filter crud. These wastes will be stored in the crud collection tank for decay and sampling purposes. The wastes will be shipped in drums for off-site disposal.

Large pieces of equipment can be decontaminated in place or in the decontamination facility. The fuel storage pool can also be used as a decay tank for highly active pieces of equipment.

4.2.5. Cleaning of Reactor Components

Methods and procedures of cleaning, water rinsing, drying, packing, shipping, storage, and inspection were studied. General specifications are being prepared for cleaning Pathfinder reactor components. The specifications are prepared in four parts: 1) preliminary cleaning by the fabricator, 2) protection of materials or components for shipment and storage, 3) on-site cleaning or final cleaning, and 4) circulatory cleaning.

Circulatory cleaning, wherein discrete segments of the Pathfinder system are cleaned by circulating cleaning fluid through them, would be the most direct method of insuring physically and chemically clean components. The selection of cleaning equipment and solutions, however, may present a problem, and further study of this method is required before it can be recommended for the Pathfinder.

Preparation of specifications for individual components and materials has been started and will continue.

4.2.6. Plant Chemistry

Studies were conducted as a part of the plant chemistry program to determine the requirements for sampling points, sampling procedures, tests, and specific chemical analyses. Objective of the program will be to obtain as much useful information on plant operation and control as possible.

4.2.7. Decontamination

Studies are being conducted on the decontamination program for the Pathfinder. These studies include studies of decontamination solutions, coatings, decontamination room layout, and equipment.

4.2.8. Reactor Water Purification System

4.2.8.1. System Design. Flash Tank. Calculations were made to predict carry-over of moisture and radioactive solids from the flash tank to the condensate system. Results indicate that the flash tank will not require a moisture separator other than an ordinary boiler-type dry pipe. Space in the steam line will be provided for a separator so that one may be installed if plant operation proves it to be necessary.

Final design requirements for the flash tank were completed. The Allis-Chalmers Heat Transfer Department is completing the final design, drawings, and specifications.

A preliminary design for a test loop with which the flash tank may be tested before installation was completed. Cost of the loop is being determined.

Normal Cooler. The normal cooler was modified so that the condensate outlet temperature could be increased from 160 F to 238 F. This condensate is now returned to the feedwater system after, rather than before, the No. 12 feedwater heater.

Shut-Down Pump. Informal proposals for both low-and high-pressure pumps were received. Design details and economics are being evaluated to determine the most suitable pump.

Filters. As a result of filter tests, a filter element pre-coated with a filtering cellulose material was selected.

4.2.8.2. Filter Test Loop. Tests of various filters were completed. Three types of filters were selected for final consideration: 1) replaceable cotton elements, 2) a cation resin bed, 3) an element precoated with a cellulose filtering material. Evaluation of test results indicate the following:

Replaceable cotton element filters retain only a small amount of aluminum corrosion product on each element. A large number of elements would therefore be required, and the elements would have to be frequently changed. Expensive remote handling equipment would also be required.

Although cotton element filters have been used successfully in other reactor systems, these reactor systems are relatively clean, and frequent changing of filter elements is not required. The filter elements may be suitable for use in other parts of the Pathfinder plant where the filter load is of a different nature.

A cation resin bed filter is suitable for the filtration of aluminum corrosion products when the maximum flow rate is about 5 gpm. The low flow rate necessitates the use of fairly large equipment. In addition, careful maintenance would be necessary to prevent corrosion products from being carried through the bed.

The collected aluminum corrosion product may be back-washed from the bed to the waste disposal system. The bed loadup test data and expected amount of corrosion product were evaluated to estimate the amount of back-wash water that would be required. Results indicate that the present liquid waste disposal system would be seriously overloaded by this backwash water, and the system would have to be enlarged.

Further consideration of the initial equipment cost, space requirement, and back-wash water requirement indicates that this type of filter is not satisfactory for use in the Pathfinder reactor water purification system.

An element precoated with a filtering cellulose material is very effective in filtering aluminum corrosion products. Although the capacity of the element is low, similar to cotton filters, the filter media may be back-washed to the waste disposal system. An estimate of the back-wash requirements indicates that the present waste disposal system is adequate. Space requirements are acceptable. The cost of the precoating material is negligible when compared to the cost of cotton cartridges. Operating costs would be lower than with a cotton element filter. The initial cost of a system using precoated elements compares favorably with the other two types. Use of another fuel cladding material rather than aluminum would have little effect on operation with the filter other than possibly extending the time between backwashes.

An element precoated with a filtering material was selected for use in the Pathfinder reactor water purification system. Various filter manufacturers are being consulted to determine the precoat-filter system that would be most suitable for the Pathfinder.

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5. INSTRUMENTATION AND CONTROL

5.1. SPECIFICATIONS

Specifications for the feedwater control, reactor pressure control, nuclear instrumentation, and control panels were prepared and submitted to potential suppliers for bids. All bids are due by February 1, 1960.

5.2. NUCLEAR INSTRUMENTATION

A seven channel system is presently being considered for use in place of the previously proposed nine-channel system. The system as now proposed utilizes instruments of improved reliability. The measurement of neutron flux, instead of using period as the controlling parameter, offers a more stable, but equally safe instrumentation system. The seven-channel system would consist of the following: two BF_3 detectors for the start-up channels, one compensated ion chamber for the log N - period channel, two compensated ion chambers for the operating channels, and two ion chambers for the power and safety channels.

The BF_3 detectors will operate from source level to 10^4 nv. They will provide count rate and period information over this range.

The log N-period channel will provide the system with period protection. The compensated ion chamber has an effective range of 10^{-11} to 10^{-4} amperes.

The operating channels will be used to control the reactor from 10^{-11} amperes to full power. A range selector switch will permit the power to be increased in decade increments at a rate proportional to the period. Control rod withdrawal is prohibited outside the .5 to 110 per cent decade power range. The scram trip will be initiated at 150 per cent of the decade power level. At full power, these channels will become part of the coincidence safety system.

The power or safety channels will operate from 10^{-7} to 10^{-4} amperes. They will be part of the coincidence safety system. The coincidence safety system will require a two out of four signal to initiate the scram trip. The two-out-of-four coincidence can be changed to a two-out-of-three coincidence if one channel is out for servicing.

5.3. CRBR PLANT CONTROL SIMULATION

Meetings were held with turbine control specialists and a control system designer to obtain specific information on control hardware and simulation of hardware dynamics. It was decided to simulate the control system, including valves, in more detail.

Discussions on the relative merits of steam dome (P_1) and steam line (P_2) control yielded the following comments.

COMPARISON COMMENTS

P_1 Control (Steam Dome Control)

1. The area of stability is limited.

P_2 Control (Steam Line Control)

1. Good system stability - no known limits on controller settings.

2. Somewhat better superheater fuel hotspot regulation than with P_2 control i.e., temperature peaks are relatively low. See Figures 3.8 & 3.9.
 3. Relief valve settings are a problem. Relief valves are placed at the superheater exit and since this pressure is allowed to vary with this type of control the pop pressures must either be variable or set too high.
 4. Good exit steam temperature regulation.
 5. Good protection against recirculation pump cavitation.
2. Superheater fuel hotspot regulation not as good. See Figures 3.8 & 3.9.
 3. No relief valve problem comparable to that with P_1 control.
 4. Close control of turbine inlet pressure.

5.4. WATER LEVEL TESTS

Reactor water level tests were completed. Results indicate that commercial instruments using differential pressure are sufficiently accurate to determine the reactor water level. A detailed report of the test is being prepared.

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6. PLANT SAFETY ANALYSIS

Reactor safeguards analyses and studies of plant safety are continuing as plant design progresses. Two supplements to the preliminary safeguards report were submitted to the Division of Licensing and Regulation in November. Subsequently, meetings were held with the Hazards Evaluation Branch and the Advisory Committee on Reactor Safeguards. Review of the safeguards report by the AEC is continuing.

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7. FEASIBILITY STUDIES

7.1. ELIMINATION OF VAPOR CLOSURE

The Internuclear Company successfully used an IBM-704 code to calculate various reactor building pressures as a function of time. The machine gave solutions over a period extending to 45 seconds after an arbitrary accident.

A rough draft of a final report on Elimination of Vapor Closure was submitted and is being reviewed.

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8. STEAM PLANT AND REACTOR AUXILIARY SYSTEMS DESIGN

A design study of the shield and fuel storage pool coolers was concluded and final performance requirements were tabulated. The design requirements will provide for the optimum in system flexibility with minimum sized coolers. The two coolers will meet the following requirements:

- 1) The fuel storage pool cooler will limit pool temperature rise to 105 F under normal loading conditions. The shield pool cooler will limit the shield pool temperature to less than 105 F. A normal fuel loading will consist of 1/3 boiling core and 1 superheater core, both of which have been operated at full power for 12 months.
- 2) The shield pool cooler can be used as a standby cooler for the fuel storage pool if necessary. It will limit pool temperature rise to 140 F or less with a normal fuel loading.
- 3) Both coolers operating together will be capable of limiting the fuel storage pool temperature rise to 140 F or less with a maximum emergency fuel loading. The maximum emergency fuel loading will consist of a whole superheater loading that has been operated at full power for 12 months, and a whole boiler core. Each third of the boiler core will have operated at full power for 12, 8, and 4 months, respectively.

Factors that tend to minimize the required cooler size were considered in the study. Significant factors include: evaporative cooling from the pool surface, increase in capacity of a cooler with pool temperature increase, rapid decrease of fuel element decay heat, and low initial temperature of a pool prior to fuel loading.

DYNAMIC LOOP TESTS #3
AVERAGE CORROSION (mg/cm^2) VS. EXPOSURE TIME (DAYS)
8001 ALUMINUM ALLOY

Fig. 1.1

43-024-051 KRM.

STATIC CORROSION TEST
TEST #10

Fig. 1.2

43-024-053 KRM.

COMPARISON GRAPH FOR
STATIC TESTS ON 8001 AL.
AND X2219 CLAD WITH M400

Fig. 1.3

43-024-052 K.R.M.

WELD GEOMETRY FOR AN ALUMINUM
CLAD BOILER FUEL ELEMENT SHOWING
TYPICAL DEFECT

Fig. 1.4

SCHEMATIC OF THE EVACUABLE
WELDING BOX

43-024-123

ONE PIN TEST SECTION
ROD HEAT FLUX VS. WALL
TEMPERATURE-WATER
TEMPERATURE AT 600 PSI
NATURAL CIRCULATION

43-024-054 K.R.M

Fig. 1.6

43-301-395

Fig. 1.7

Fig. 1.8

Seven-Pin Cluster Superheater Fuel Element

REVISIONS		DATE	BY	APP'D	REASON
1	DESIGNED	11-1-52	W. L. BROWN		DESIGNED
2	ENGINEERED	11-1-52	W. L. BROWN		ENGINEERED
3	MANUFACTURED	11-1-52	W. L. BROWN		MANUFACTURED
4	TESTED	11-1-52	W. L. BROWN		TESTED
5	INSPECTED	11-1-52	W. L. BROWN		INSPECTED
6	ASSEMBLED	11-1-52	W. L. BROWN		ASSEMBLED
7	SHIPPED	11-1-52	W. L. BROWN		SHIPPED
8	RECEIVED	11-1-52	W. L. BROWN		RECEIVED
9	DISPOSED	11-1-52	W. L. BROWN		DISPOSED
10	REMOVED	11-1-52	W. L. BROWN		REMOVED
11	REPLACED	11-1-52	W. L. BROWN		REPLACED
12	REPAIRED	11-1-52	W. L. BROWN		REPAIRED
13	REWORKED	11-1-52	W. L. BROWN		REWORKED
14	RETESTED	11-1-52	W. L. BROWN		RETESTED
15	REINSPECTED	11-1-52	W. L. BROWN		REINSPECTED
16	REASSEMBLED	11-1-52	W. L. BROWN		REASSEMBLED
17	RESHIPPED	11-1-52	W. L. BROWN		RESHIPPED
18	REMOVED	11-1-52	W. L. BROWN		REMOVED
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21	REWORKED	11-1-52	W. L. BROWN		REWORKED
22	RETESTED	11-1-52	W. L. BROWN		RETESTED
23	REINSPECTED	11-1-52	W. L. BROWN		REINSPECTED
24	REASSEMBLED	11-1-52	W. L. BROWN		REASSEMBLED
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61	REWORKED	11-1-52	W. L. BROWN		REWORKED
62	RETESTED	11-1-52	W. L. BROWN		RETESTED
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77	REWORKED	11-1-52	W. L. BROWN		REWORKED
78	RETESTED	11-1-52	W. L. BROWN		RETESTED
79	REINSPECTED	11-1-52	W. L. BROWN		REINSPECTED
80	REASSEMBLED	11-1-52	W. L. BROWN		REASSEMBLED
81	RESHIPPED	11-1-52	W. L. BROWN		RESHIPPED
82	REMOVED	11-1-52	W. L. BROWN		REMOVED
83	REPLACED	11-1-52	W. L. BROWN		REPLACED
84	REPAIRED	11-1-52	W. L. BROWN		REPAIRED
85	REWORKED	11-1-52	W. L. BROWN		REWORKED
86	RETESTED	11-1-52	W. L. BROWN		RETESTED
87	REINSPECTED	11-1-52	W. L. BROWN		REINSPECTED
88	REASSEMBLED	11-1-52	W. L. BROWN		REASSEMBLED
89	RESHIPPED	11-1-52	W. L. BROWN		RESHIPPED
90	REMOVED	11-1-52	W. L. BROWN		REMOVED
91	REPLACED	11-1-52	W. L. BROWN		REPLACED
92	REPAIRED	11-1-52	W. L. BROWN		REPAIRED
93	REWORKED	11-1-52	W. L. BROWN		REWORKED
94	RETESTED	11-1-52	W. L. BROWN		RETESTED
95	REINSPECTED	11-1-52	W. L. BROWN		REINSPECTED
96	REASSEMBLED	11-1-52	W. L. BROWN		REASSEMBLED
97	RESHIPPED	11-1-52	W. L. BROWN		RESHIPPED
98	REMOVED	11-1-52	W. L. BROWN		REMOVED
99	REPLACED	11-1-52	W. L. BROWN		REPLACED
100	REPAIRED	11-1-52	W. L. BROWN		REPAIRED

Fig. 1.9

43-401-023

Fig. 1.10

NO.	QTY.	DESCRIPTION	MATERIAL	PART NUMBER	WT.
1	1	TEST SECTION HOUSING ASSEMBLY		43-400-014	1.26
2	1	TEST CLUSTER PUL ELEMENT CONTAINER		43-390-413 1201	
2	3	CLUSTER FLANGE		43-390-015 001	
3	10	100% 7/8" DIA. 3/4" WAVE 1/2" LG.	303 SS	THIS	0.46
4	3	BRAKE 7/8" DIA. 3/4" LG. BAR	303 SS	THIS	0.37
5	1	FLUX	AC-1	THIS	0.06
6	1	SILVER SOLDER	AC-2	THIS	0.07
7	1	WELD ROD 303 SS 1/8" DIA. 1/16" WAVE 1/2" LG.	303 SS	THIS	0.08

Fig. 1.11

Two-Bundle Zircaloy-2 Clad Boiler Fuel Element

Fig. 1.12

Nozzle Galling Test Layout

Fig. 1.13

UNLESS OTHERWISE NOTED: DIMENSIONS ARE IN INCHES & FRACTIONS UNLESS NOTED OTHERWISE				ANGLES	
DIMENSION	UP TO 8 INCH	OVER 8 TO 24 INCH	OVER 24	UP TO 8 INCH	OVER 8 INCH
FRACTIONAL	1/16	1/8	1/4	1/4	1/2
DECIMAL	0.005	0.010	0.015	0.005	0.010
- Confidential - Property of ALLIANCE MANUFACTURING CO., MILWAUKEE, WIS.					
NUCLEAR POWER					
BY ORIGIN	DATE	APP'D	DATE	BY ORIGIN	DATE
100	10/1/68	100	10/1/68	100	10/1/68

**NOZZLE
GALLING
TEST LAYOUT**

43-002092

NOTE #1:-
ALL WELDS TO BE STEAM
TIGHT AT 150 PSI.

NOTE 2:-
HELICALLY WIND NICHROME
HEATING WIRE (IT. 17) AROUND
THE O.O. OF IT. 2.

Superheater Element Seal Test Assembly

Mockup of One-Eighth of Pathfinder Core

Fig. 1.15

Fig. 2.1

Fig. 2.2

43-024-049

ITEM	DESCRIPTION	MATERIAL	QTY	UNIT	QTY	UNIT
1	STEAM SEPARATOR TEST ASSEMBLY		1	ASSEMBLY		
2	STAND PIPE ASSEMBLY		1	ASSEMBLY		
3	DIFFUSION TANK ASSEMBLY		1	ASSEMBLY		
4	1/2" UNF-2A X 1/2" LG. H.V. S.F. HEX. BOLT	STEEL	3000	PCS		
5	1/2" UNF-2B H.V. S.F. HEX. NUT	STEEL	3000	PCS		
6	GASKET 15 O.D. X 11 I.D. X 1/8" THK	NEOPRENE	1	PC		
7	PUMP AND RELATED PIPING		1	ASSEMBLY		
8	1/2" UNF-2A X 1/2" LG. H.V. S.F. HEX. BOLT	STEEL	3000	PCS		
9	1/2" UNF-2B H.V. S.F. HEX. NUT	STEEL	3000	PCS		
10	GASKET 10" O.D. X 8" I.D. X 1/8" THK	NEOPRENE	1	PC		
11	HEATER MANIFOLD ASSEMBLY		1	ASSEMBLY		
12	AIR COLLECT TANK		1	ASSEMBLY		
13	PIPE ASSEMBLY		1	ASSEMBLY		
14	1/2" UNF-2A X 1/2" LG. H.V. S.F. HEX. BOLT	STEEL	3000	PCS		
15	1/2" UNF-2B H.V. S.F. HEX. NUT	STEEL	3000	PCS		
16	GASKET 16" THX X 12" LG.	NEOPRENE	1	PC		
17	STEAM SEPARATOR ASSEMBLY		1	ASSEMBLY		
18	BAFFLE PIPE ASSEMBLY		1	ASSEMBLY		
19	1/2" UNF-2A X 1/2" LG. H.V. S.F. HEX. BOLT	STEEL	3000	PCS		
20	STD. WT. SCREWED PIPE COUPLING		1	PC		
21	2" SCH. 40 X 68 LG. PIPE (THRD BOTH ENDS)	STEEL	1	PC		
22	PACKING		1	PC		
23	1/2" UNF-2A X 1/2" LG. H.V. S.F. HEX. BOLT	STEEL	3000	PCS		
24	GASKET 12" O.D. X 11 I.D. X 1/8" THK	NEOPRENE	1	PC		
25	8-125" FLANGED GATE VALVE CRANE "465"	IRON BODY	1	PC		
26	8 PIPE SUPPORT CRANE "259-G"	EAST IRON	1	PC		
27	MIST ELIMINATOR-METAL TEXTILE CORP. DWG. IC1813		1	PC		
28	MIST ELIMINATOR-METAL TEXTILE CORP. DWG. IC1814		1	PC		

Steam Separator Test Assembly

Fig. 2.3

STEAM SEPARATOR TEST ASSEMBLY	
NUCLEAR POWER	
43-500-649

FLAT TENSILE TEST SPECIMEN

THERMAL FLUX ACROSS FISSION GAP SPACE

THERMAL FLUX ACROSS FUEL ELEMENT & CONTROL ROD CHANNEL

43-024-061 KRM

E_f VERSUS REACTOR OPERATING TIME
FOR DIFFERENT FUEL POSITIONS

Fig. 3.3

43-024-055 K.R.M.

SUPERHEATER FUEL LOADING

vs
PLANT POWER
(SUPERHEATER POWER)
(STEAM TEMP.)
(HOTSPOT TEMP.)

Fig. 3.4

43-023-975 JFW 9-3-59

RADIAL FLUX DISTRIBUTIONS AT THE MIDPLANE OF THE REACTOR

LOW ENRICHMENT
SUPERHEATER ELEMENT

43-024-056 K.R.M.

Fig. 3.6

K_{eff} vs Element Spacing
for Different Enrichments

$T = 489^{\circ}\text{F}$

43-024 - 060 KRM.

Fig. 3.7

Comparison of P_1 and P_2 Control, Control Rod Run-Out

Fig. 3.8

43-024-058 K.R.M.

Comparison of P₁ and P₂ Control, Generator Demand Change

Fig. 3.9

P₂ Control, Turbine Trip Valve Closed and Main Bypass Valve
Began to Open 0.1 Seconds Later

Fig. 3.10

ALL ELEMENTS FULLY LOADED
(9X9 ROD ARRAY PER BOX)
EXCEPT AS NOTED

FIGURE
INITIAL CRITICAL CORE CONFIGURATION

43-024-050 K.R.M.

Fig. 3.12

43-024-119

Fig. 3.13

43-024-120

ALL ELEMENTS FULLY LOADED
(9X9 ROD ARRAY PER BOX)
EXCEPT AS NOTED

CRITICAL CORE CROSS SECTION

43-024-048 H.R.M

Fig. 3.15

43-024-122

*5X3⁶/9 ELEMENT CORE USED DURING
9X9 LATTICE ACTIVATION STUDIES*

43-024-075 T.N.T.

Fig. 3.16

Fig. 3.17

DETAIL OF FLUX PEAK BETWEEN BUNDLE, AXIAL FLUX TRAVERSE NO. 1
 5X3⁶/₉ ELEMENT CASE, 9X9 LATTICE; BARE, 1/4" MN-CU FOILS

DETAIL OF REFLECTOR FLUX PEAK
 AXIAL TRAVERSE #1
 5X3% ELEMENT CORE, 9X9 LATTICE; BARE, 1/4" MN-CU FOILS

Fig. 3.19

43-024-072
 T.N.T.

NORTH-SOUTH RADIAL FLUX TRAVERSE
 5X3⁸/₉ ELEMENT CORE, 9X9 LATTICE; BARE, 1/4" U-AL FOILS

Fig. 3.20

Fig. 3.21

DETAIL OF REFLECTOR FLUX PEAK
EAST-WEST RADIAL FLUX TRAVERSE
5X3⁹/₉ ELEMENT CORE, 9X9 LATTICE
BARE, 1/4" U-AL FOILS

Fig. 3.22

DETAIL OF CONTROL ROD WATER CHANNEL FLUX PEAK
NORTH-SOUTH RADIAL FLUX TRAVERSE

5X3% ELEMENT CORE, 9X9 LATTICE; BARE, $\frac{1}{4}$ " U-AL FOILS

43-024-074
T.N.T.