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ORNL-2657
Reactors—Power
TID-4500 (14th ed.)

MARITIME REACTOR PROJECT

ANNUAL PROGRESS REPORT

FOR PERIOD ENDING NOVEMBER 30, 1958

OAK RIDGE NATIONAL LABORATORY
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION

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A. L. Boch, Project Coordinator

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SUMMARY

This is the initial report in a series of annual reports on ORNL participation in the program for development of nuclear-powered merchant ships. This program is under the joint direction of the Maritime Reactors Branch of the Division of Reactor Development of the U. S. Atomic Energy Commission and the Nuclear Power Office of the Maritime Administration of the U. S. Department of Commerce. The Laboratory supplies, in addition to review and technical advisory service to the AEC, experimental and research service in support of the developmental program. The supplementary studies, experimental programs, and specialized services conducted during the past year are described in this report and summarized below.

In studies of hazards associated with the operation of the N. S. Savannah, estimates were made of the activity buildup to be expected in the primary cooling system, the coolant purification system, and the waste disposal system. Evaluations of the relative importance of corrosion-product and recoil activities indicated, as was expected, that corrosion-product activity is dominant in pressurized-water reactor systems. The results of the idealized calculations were adjusted on the basis of experience with presently operating pressurized-water systems.

The possibility of an ozone explosion in the liquid-nitrogen-cooled charcoal adsorbers of the gaseous waste collection system was examined. It was concluded that explosions could be avoided by excluding oxygen from low-temperature systems. Similarly the potential hazard of a hydrogen explosion was evaluated. Flash-back arrestors will be provided where appropriate. Since stratification of hydrogen released to the containment vessel could possibly permit the buildup of a flammable concentration in a static atmosphere, forced air circulation will be provided at the top of the control rod housing. Instrumentation will be provided, as required, to measure hydrogen buildup.

Specific criteria for the containment vessel for the N. S. Savannah reactor were developed. The containment vessel is designed to retain the radioactivity that would be released by the maximum credible accident. Features considered in the establishment of the criteria were allowable pressure, temperature, and leakage, missile protection, penetration design, stress level, and dose level.

A limited study is under way in an attempt to develop an optimum waste disposal procedure to be used on nuclear merchant ships of the N. S. Savannah type. Both dockside and sea disposal of the demineralizer resins, radioactive gases, and other radioactive wastes are being studied.

Procedures to be followed to provide effective radioactive protection for the passengers, crew, and any other persons associated with the operation and maintenance of the vessel are being developed. The procedures are to be set forth in a Health Physics Manual that is now being drafted.

A pressurized-water in-pile test loop is being designed and constructed that will be operated in the Oak Ridge Research Reactor. Fuel elements proposed for the N. S. Savannah will be tested in this loop under simulated reactor operating conditions in a radiation field. The loop provides for the simultaneous irradiation of six fuel pins and the out-of-pile testing of control specimens. Examinations of test specimens will include dimensional measurements for swelling or distortion, metallographic examination of sections, analyses of fission gases, removal of the uranium oxide pellets, and examination for cracks, other damage, and burnup. Thermocouple attachments will be examined to determine the effect of irradiation.

Metallurgical studies have been made as necessary in the evaluation and testing of reference design components for the first core of the N. S. Savannah. Experiments were conducted with both Nicrobraz No. 50 and electroless nickel as brazing alloys for joining fuel element tubes and ferrules. Joints prepared with these alloys are brittle and susceptible to hairline cracking but reasonable strengths can nevertheless be obtained. Electroless nickel offers advantages in terms of preplacement of the brazing alloy. Neither alloy appears to have a deleterious effect on the type 304 stainless steel base metal.

Welding studies have indicated that welds free from defects can be prepared with either cup-type or plug-type end closures. These joints can be radiographically inspected for porosity and lack of fusion on a 100% basis by making three exposures of each weld.

Two methods of incorporating the burnable poison, boron, into the fuel elements have been examined. The reference method of using wrought

boron-containing stainless steel tubing has the disadvantage that boron losses occur when brazing in hydrogen atmospheres at temperatures above 1010°C. These losses can be avoided by brazing in vacuum or argon. Attempts to incorporate boron into UO_2 either as ZrB_2 or as a controlled impurity met with little success.

Development studies of pyrometer wires for monitoring the temperature of UO_2 pellets during irradiation were initiated. Experiments have indicated that gold, nickel, vanadium, and probably columbium will be acceptable as the wire material. Platinum showed a marked propensity to react with UO_2 at temperatures well below its known melting point.

Studies of the use of metal heat transfer bonds to replace helium in the fuel elements in future cores of the N. S. Savannah are also in progress. Static corrosion tests indicate that type 304 stainless steel is not attacked by either lead or bismuth after 500 hr at 400°C.

In support of and in cooperation with the physics group of the Babcock & Wilcox Company, calculations are being made of the fine flux variations across fuel tubes, across fuel element cans, near control rods, and near control rod followers. Results obtained thus far are presented. Also a two-group, two-dimensional study of the power distribution in the core for various control rod configurations is being undertaken. Some results are presented for the design core at the operating temperature. The results indicate that the control rods have a relatively localized effect on power distribution and that severe power peaking may occur for particular control rod configurations.

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MARTIME REACTOR PROJECT ANNUAL PROGRESS REPORT

INTRODUCTION

In September 1957 the Laboratory began a program of technical support to the Maritime Reactors Branch of the Division of Reactor Development (AEC) in the development of nuclear-powered merchant ships. This support is coordinated by the Package Reactor Group of the Reactor Projects Division. Starting at a modest level, the ORNL effort was gradually expanded until it now represents a significant, though not a major, Laboratory program.

The main effort of the AEC maritime reactor program is presently directed toward construction of the N. S. Savannah, the first nuclear-powered cargo-passenger vessel. This ship will be 595 ft in length, will have a dry cargo capacity of 10,000 long tons, and will carry a 124-man crew plus 60 passengers. It will be powered by a pressurized-water reactor fueled with slightly enriched UO_2 pellets canned in stainless steel tubes. A steam-turbine drive will deliver 20,000 SHP to a single screw which will propel the ship at a normal cruising speed of 21 knots.

Conceptual design of the ship was done by George G. Sharpe Company of New York. Detailed ship design and construction is being done by New York Shipbuilding Corporation of Camden, New Jersey. Babcock & Wilcox Co. of Lynchburg, Virginia, is furnishing the reactor and propulsion equipment. States Marine Line has been selected as the ship operator. The keel was laid on National Maritime Day (May 22, 1958). Launching is scheduled for May 1959, and the ship is expected to be operational in early 1960.

The Laboratory's support to the Maritime program may be divided into two general areas: (1) review and technical advisory service, and (2) supplementary studies, experimental programs, and specialized services. The work is predominantly directed toward construction of the N. S. Savannah but also extends into other phases of the Maritime program.

The package reactor group acts essentially as a technical extension of the staff of the Maritime Reactors Branch in reviewing reactor designs, specifications, and development programs being carried out by AEC contractors and in providing advice and guidance on technical matters. The assistance of numerous specialists throughout the Laboratory is enlisted in this advisory service. In the past year approximately 200 specifications, drawings, reports, and other documents were reviewed and comments transmitted to MRB. Special attention was given to those areas relating to reactor controls, safeguards considerations, activity buildup in the primary system, reactor physics experiments and calculations, and fuel element design and irradiation programs.

Supplementary studies and experimental programs were undertaken from time to time as the need arose and where it appeared advisable to utilize the specialized facilities and personnel available at the Laboratory. A small group was established in the Metallurgy Division for the investigation of several problems in fuel and core fabrication. The Reactor Projects Division undertook the design and construction of a pressurized-water loop which will be installed in the Oak Ridge Research Reactor. The Health Physics Division is preparing a manual of procedures for the N. S. Savannah shipboard health physics program. The Chemical Technology Division made progress in the development of techniques for sea disposal of radioactive wastes from nuclear merchant ships. The ORNL analog simulator is being used to study the kinetic behavior of the reactor. Studies of control rod programming and other supplementary physics calculations on the Oak Ridge IBM-704 were performed in a cooperative effort with the reactor designer. The Inspection-Engineering Department is providing inspection services for MRB during the fabrication of primary system components and will perform a similar function for fuel elements and core components.

1. SYSTEM HAZARDS STUDIES

The hazards associated with the operation of the N. S. Savannah are being critically reviewed. The preparation of the actual hazards report is the responsibility of the principal contractors – Babcock & Wilcox Co., George G. Sharpe Co., and New York Shipbuilding Corp. – and their evaluation of the problems involved is being reviewed by ORNL for the AEC.

The several independent studies carried out at the Laboratory to establish review criteria are described here. In addition to the work described here and the detailed reviews made of the various systems designs, specifications, and reports, information was supplied to the Maritime Reactors Branch on (1) the shock loading to be expected in various ship accidents, (2) a recommended testing program, (3) estimates of the fission-product gases in the pressurizer, and (4) an evaluation of the dose from stack discharge.

ACTIVITY IN THE PRIMARY WATER AND PURIFICATION SYSTEMS

Estimates were made of the activity buildup to be expected in the primary cooling system, the coolant purification system, and the waste disposal system in order to evaluate the shielding for these systems. The contributions to the coolant activity of the activities resulting from water impurities, corrosion products, fission products, and recoil atoms were calculated. Details of these calculations have been reported.[1]

The specific activity in the coolant resulting from each of the mechanisms was determined and compared with the intrinsic activity in the coolant for two purification system flow rates. An interesting feature of this study was the determination of the relative importance of corrosion-product and recoil activities from in-pile metal specimens as a function of the half-life of the activity. The data are presented in Fig. 1.1. As was expected, corrosion-product activity is dominant in pressurized-water reactor systems. The specific activity in the primary water and total activity in the demineralizer are given in Tables 1.1 and 1.2, respectively, in comparison with similar data obtained by Babcock and Wilcox.[2] Calculations of the shielding required at the demineralizer were based on these activity values.

None of the calculations made to date have included the rate of deposition of nuclides on the surfaces of the primary water system, since data on such deposition are not available. This is the major source of possible discrepancy between the idealized calculations and the actual system. These and other factors influencing the buildup of activity in water systems were evaluated on the basis of experience with presently operating pressurized-water systems.[3]

GAS RELEASE FROM UO_2

Gas release from the UO_2 fuel is of concern both as it affects the pressure buildup within the fuel capsule and as it affects the temperature at which the fuel and capsule will operate. Wide differences of opinion exist between experts in the field with regard to the interpretation of experimental results obtained to date, and the situation is very difficult to analyze because of the number and complexity of the interrelations involved. Omission of any factor can give misleading results. The Westinghouse model for gas release from UO_2(ref 4) has been used to derive a set of linear equations which describes the gas release and temperature relationships in the fuel elements. This information is being coded for the IBM 704.

OZONE EXPLOSIONS

The gaseous waste collection system for the N.S. Savannah employs liquid-nitrogen-cooled charcoal adsorbers to remove fission-product gases from the gas mixture above the liquid in the buffer-sealed surge tank. In view of the several known instances of explosions in cryogenic apparatus employed in radiation fields, the potential occurrence of such an

[1]H. N. Culver, *Cooling Water and Demineralizer Activities in NMSR*, ORNL CF-58-5-83 (May 14, 1958).

[2]Babcock & Wilcox Co., *Determination of the Purification Rate for Normal Operation*, BAW-1015 (rev), June 1958.

[3]H. N. Culver, *Cooling Water Activity Buildup in Pressurized-Water Reactors*, ORNL CF-58-8-2 (Aug. 1, 1958).

[4]J. D. Eichenberg *et al.*, *Effect of Irradiation on Bulk UO_2*, WAPD-183, October 1957.

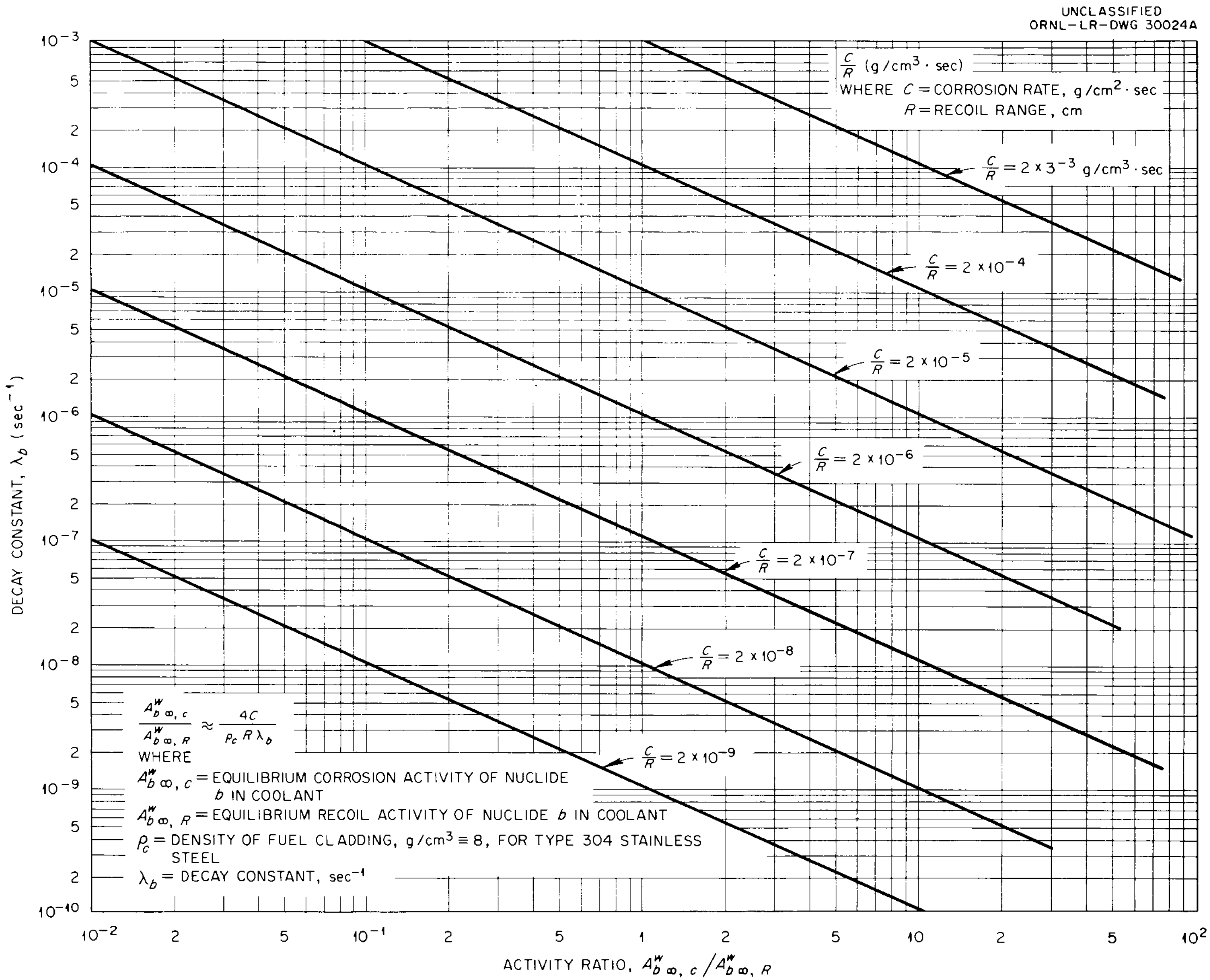

Fig. 1.1. Relative Importance of Corrosion-Product and Recoil Activities in Reactor Coolant Stream as a Function of Half-Life.

Table 1.1. Specific Activity in Water During Operation of N.S. Savannah Reactor

Isotope	Half-Life	Activity (μc/ml) Babcock & Wilcox Calculation	ORNL Calculation
Na^{24}	15 h	8.57×10^{-3}	1.11×10^{-2}
Co^{60}	5.3 y	1.19×10^{-2} [(a)]	3.68×10^{-4} [(b)]
Co^{58}	71 d	2.16×10^{-3}	1.26×10^{-3}
Fe^{55}	2.9 y	1.09×10^{-2}	1.49×10^{-2}
Fe^{59}	47 d	5.49×10^{-4}	3.14×10^{-4}
Cr^{51}	27 d	4.00×10^{-2}	2.16×10^{-2}
Ni^{65}	2.56 h	2.97×10^{-4}	1.22×10^{-4}
Mn^{56}	2.58 h	5.42×10^{-2}	1.91×10^{-2}
Mn^{54}	300 d		2.25×10^{-4}
Ta^{182}	112 d	1.67×10^{-2}	(*c*)

(*a*)Calculated for 1 wt % Co in metal.
(*b*)Calculated for 0.007 wt % Co in metal.
(*c*)Ta^{182} will not be present in type 304 stainless steel in detectable amounts.

Table 1.2. Total Activity in Demineralizer Resin for 50-Day Period of Operation of N.S. Savannah Reactor

Isotope	Half-Life	Activity (curies) Babcock & Wilcox Calculation	ORNL Calculation
Na^{24}	15 h	1.05	1.09
Co^{60}	5.3 y	75.3[a]	1.98[b]
Co^{58}	71 d		5.25
Fe^{55}	2.9 y	75.0	79
Fe^{59}	47 d	3.81	1.19
Cr^{51}	27 d	147.5	67
Ni^{65}	2.56 h		2.05×10^{-3}
Mn^{56}	2.58 h	1.15	3.24×10^{-1}
Mn^{54}	300 d		1.15
Ta^{182}	112 d	103	*c*
Total activity on resin		406.81	156.98

[a]Calculated for 1 wt % Co in metal.
[b]Calculated for 0.007 wt % Co in metal.
[c]Ta^{182} will not be present in type 304 stainless steel in detectable amounts.

explosion in the charcoal adsorber system was examined. It was concluded[5] that, since the explosion was due to the rapid reduction of ozone, explosions could be avoided if oxygen and/or ozone were excluded from low-temperature systems wherein they might concentrate by liquefaction. Such provisions would make the proposed circulating-nitrogen-cooled trap for the off-gas system acceptable if the initial oxygen content of the nitrogen were less than 200 ppm. Inasmuch as the off-gas system is to have a hydrogen overpressure, there should be no danger from the accumulation of oxygen from the off-gas.

HYDROGEN HAZARDS

Hydrogen is employed on the N.S. Savannah to provide a constant overpressure in the primary system in order to minimize the formation of oxide corrosion products. Its use raises the question of the potential hazard of a hydrogen explosion in the various parts of the system in which significant concentrations of the gas are possible. Hydrogen will, for example, collect in the gas above the liquid in the primary buffer seal and purification systems, as well as the waste disposal tank and the containment vessel when the off-gases are discharged therein. The hazard associated with the hydrogen supply system is minimized by the use of flash-back arrestors in this system, and all areas wherein this equipment is used are vented so that the leakage of hydrogen through defective fittings or valves will not result in accumulation of hydrogen in the surroundings.

In general, a hydrogen hazard is avoided by maintaining the hydrogen concentration either well above

[5]W. B. Cottrell, *Ozone Explosion in Irradiated Cryogenic Apparatus and Implications upon N.S. Savannah Gaseous Waste Disposal System*, ORNL CF-58-7-97 (July 24, 1958).

or below the flammable range. In the primary system, for example, the oxygen is reduced to water so that the hydrogen concentration is well above the flammable range. This is not possible in the waste disposal tanks which are initially filled with air. However, these tanks are protected by flashback arrestors, and by the absence of any potential ignition sources. On the other hand, the hydrogen vented to the containment vessel during the time the ship is in port or when stack discharge is otherwise impossible, would accumulate to proportions near the flammable limit if all the hydrogen initially present in the cylinders were accumulated in the containment vessel and remained there for a long period of time. Although it may be concluded that the concentration could not build up to a flammable level if the hydrogen were well mixed with the containment vessel environment, the possibility of stratification of the gases is not precluded, and local flammable concentrations could exist at the top of the container. To protect against this situation, it is necessary to provide forced air circulation in the otherwise static ambient at the top of the control rod housing. Instrumentation will, of course, be provided to measure the hydrogen buildup both at that location and in other areas containing hydrogen.

CONTAINMENT VESSEL DESIGN CRITERIA

Consideration of a containment vessel for the N.S. Savannah reactor led to specific criteria for several important features, such as allowable pressure, allowable temperature, allowable leakage, missile protection, penetration design, stress level, and dose level. The initial containment vessel design submitted to ORNL for review provided for the containment of only the fluid that might be released from the primary systems. Since it was the contention of ORNL that the containment vessel should accommodate the fluid released from one secondary system, as well as that from the primary systems, calculations of the containment vessel pressure and temperature for this condition were performed. A pressure of 212 psia and a temperature of 387°F were found. Subsequent similar calculations at Babcock & Wilcox, which included provision for the energy adsorbed by the containment vessel, its contents, and shielding, gave a design pressure of 173 psig.[6]

Preliminary calculations were also made to establish a leakage criterion for the containment vessel. The entire containment system, including the containment vessel and any extension thereof, should be tested to a leaktightness determined by the amount of activity involved and the dilution factors available. The calculations indicated that a leak rate of 1 standard cm^3/sec would give an inhalation dose external to the containment vessel of 1 r/hr.

Inasmuch as the containment vessel is provided to retain the radioactivity which would be released by the maximum credible accident, it is imperative that the vessel remain intact during this accident. Various studies have shown that when steels are at temperatures above the FTP (fracture transition plastic) temperature, failures occur solely in a ductile manner and missiles are not generated when the metal ruptures. Since the NMSR system temperatures are well above the FTP temperature and the strain rates are well below rates of shock wave propagation, missiles will not be generated by the piping or vessels, and therefore over-all missile protection for the containment vessel is not necessary. Missiles could be generated by components such as valves and control rods, however, and the system will have to be carefully examined in order to provide local protection for such missiles.

Penetrations, piping, electrical leads, and service access through the containment vessel walls must be examined with regard to structural integrity, protection against missiles, radiation streaming, and leaktightness under disaster temperatures and pressures. All the penetrations which do not constitute extensions of the containment vessel must be designed to isolate the disaster condition. Penetrations which affect the continuity of the containment vessel must be designed for the existence of disaster conditions therein.

A preliminary study of stress levels in the containment vessel was made for two particular areas of concern: (1) stress concentration in the vessel wall because of penetrations, and (2) thermal stresses in the penetrating pipe and wall area. These studies indicate that the thermal stresses are satisfactorily taken care of but that the penetration stress concentrations in the vessel wall will require additional reinforcement of the wall in the vicinity of these penetrations.

[6] J. E. Lemon and J. C. Ellington, *Containment Pressure Analyses Report*, BAW-1123 (Oct. 10, 1958).

2. WASTE DISPOSAL

Developmental work is under way on an optimum waste disposal procedure to be used on nuclear merchant ships of the N.S. Savannah type. Methods of converting the ion-exchange resins, radioactive gases, and other radioactive wastes to a form suitable for disposal to the environment are being studied. Both dockside and sea disposal are being considered. Since the reactor is a pressurized-water reactor, the principal source of activity is in the primary coolant system. The specific sources of radioactivity are the demineralizer resin beds, the radioactive gases, and the liquid wastes accumulated from the primary coolant blow-down water, laboratory wastes, and shield water. The initial planning of the development program and the preliminary experimental work were described previously.[1]

Four methods of waste disposal are available. First, there could be total containment of all waste aboard the ship while at sea with disposal accomplished at dockside. This method seems undesirable, on the whole, if for no other reasons than that it involves space-consuming storage tanks aboard ship and offers a possible source of contamination of the ship. Second, waste disposal might be accomplished by total release of the activity at sea. It is more likely, however, that limits will be set on sea disposal which will greatly restrict this practice. Sea disposal might be practical, however, if the wastes were concentrated aboard ship by evaporation. The concentrated waste could be placed in containers for disposal at sea or removal at dockside. The waste liquid could be used as makeup fluid for concrete which could be dumped as blocks or in containers. The fourth and the most promising method of waste disposal involves the concentration of the wastes with the use of ion-exchange units. These units would be in addition to those in the primary purification system. With these units it would be possible to eliminate the large volumes associated with the waste by venting the decontaminated effluent stream to the sea. The exhausted resin could be handled by any one of the following methods: (1) contained aboard ship for dockside disposal or regeneration, (2) blown overboard, (3) bagged or canned before disposal to the sea, (4) incorporated in concrete and the concrete disposed at sea either as blocks or in containers, (5) stripped (cleaned) by controlled, slow elution with sea water, followed by regeneration with acid and base. The rate of disposal of the radioactive effluent to the sea could be controlled with the use of monitors so as to meet specified levels for sea disposal. This method is attractive because it permits the re-use of the expensive nuclear-grade resins. This stripping and regeneration procedure could also be applied to the ion-exchange resins used in the primary purification system.

DISPOSAL OF DEMINERALIZER RESIN

The demineralizer bed in the primary purification system will contain sorbed activity and activity associated with the corrosion products filtered by the resin bed. The activity associated with the cooling water and the demineralizer have been estimated and the values are presented in Chap. 1, Tables 1.1 and 1.2, of this report.

The simplest disposal technique would be to discharge the resin bed directly into the ocean. Consequently, the elution effect of sea water on various activities sorbed on a mixed resin bed was investigated. The bulk of the various activities were eluted from the resin in the number of column volumes of sea water indicated in the following: Cs, 7; Nb, 17; Sr, 21; Zr, 62; total rare earths, 62; and I, 120. The data are plotted in Figs. 2.1 and 2.2. Chromium-51 activity was 80% eluted by sea water in 15 column volumes and Ru^{106} was 30% eluted in 16 column volumes (Fig. 2.3). The remaining fractions of the Cr^{51} and Ru^{106} activity eluted very slowly, and it appears that hydrolysis of the chromium and the ruthenium occurred during the elution.

Another method for sea disposal of the exhausted resin bed would be to encase it in cans, cement, ceramics, etc. and deposit it on the ocean floor. Studies were therefore initiated on the behavior of the resin when mixed with Portland cement. Upon mixing of the cement and the resin, gassing was observed. After the cement had set, the resin was found to be homogeneously dispersed in the cement block and there were small gas pockets. The densities of the blocks varied from 1.3 to 1.5

[1] W. J. Neill, *Status of and Suggested Development Program for Merchant Ship Reactor Waste Disposal*, ORNL CF-58-8-88 (Aug. 25, 1958).

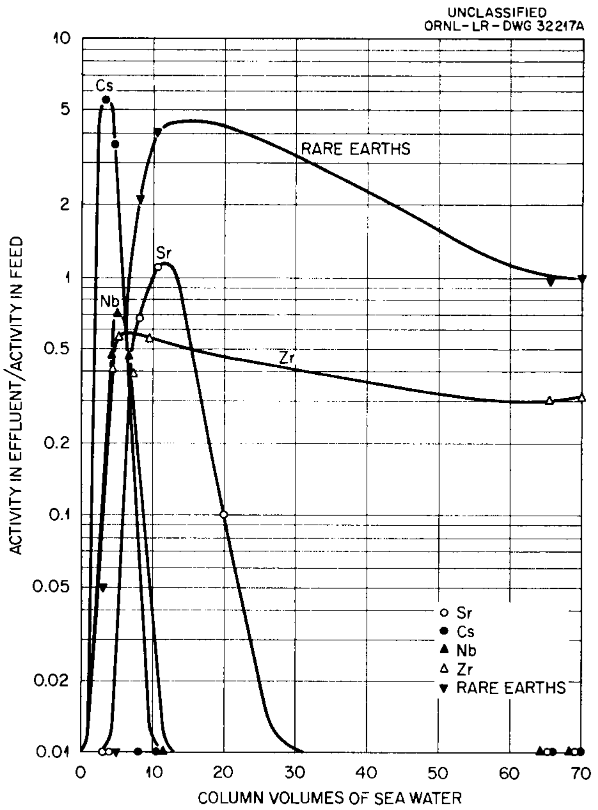

Fig. 2.1. Elution of Fission-Product Activity from a Mixed Resin Bed with Sea Water.

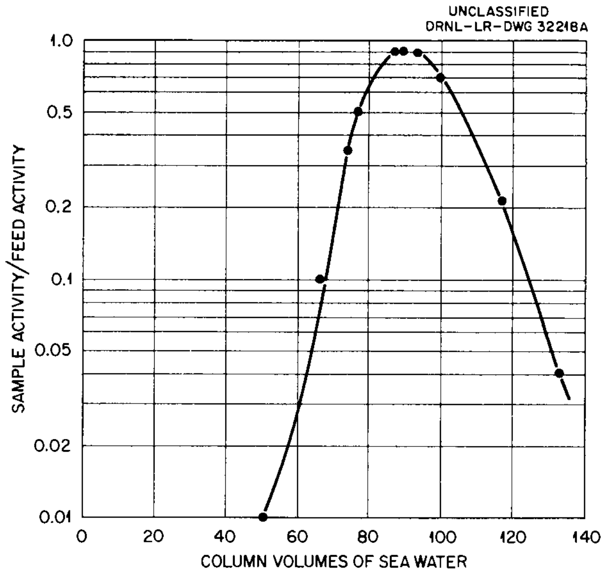

Fig. 2.2. Elution of I^{131} from a Mixed Resin Bed by Sea Water.

g/cm^3 for resin-cement mixtures varying from 100 ml of resin in 115 g of cement to 100 ml of resin in 230 g of cement. This density range is sufficient to ensure deposition of the blocks on the ocean floor. Sea-water leaching experiments with resin-containing cement blocks showed that after 42 days the activity in the sea water was 9.5% of the gross γ and 3% of the gross β of the resin. The sea water had leached 18% of the Cs, 1% of the Sr, 1% of the Ru, and 2% of the total rare earths from the cement-resin mixture. The Sr, Ru, and rare earths activities in the sea water were near background level, thus the values listed for these components are only approximate.

It may be concluded from experimental data that disposal of the resin bed directly to the ocean would result in the rapid elution of the bulk of the activity. If disposal of the resin bed in cement is undertaken, further work should be performed on high-temperature, waterproof cements and suitable containers or coatings to decrease leachability.

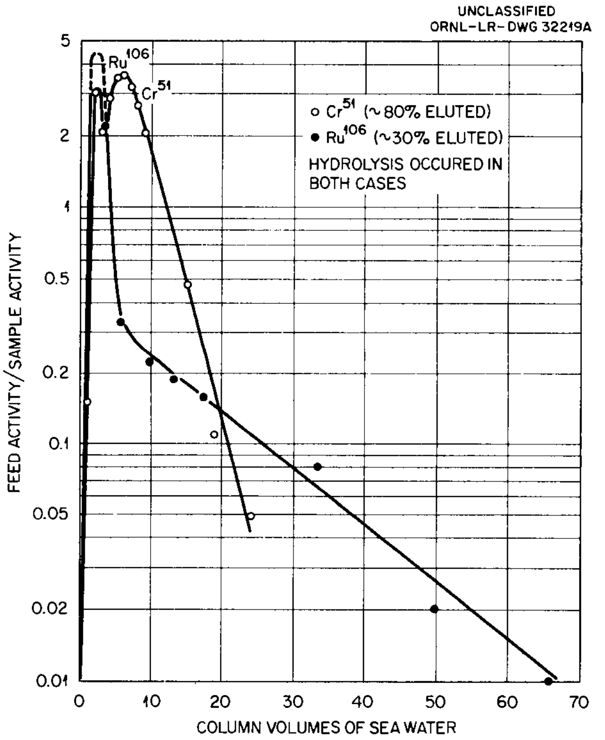

Fig. 2.3. Elution of Cr^{51} and Ru^{106} Activity from a Mixed Resin Bed with Sea Water.

DISPOSAL OF LIQUID WASTES

Liquid wastes aboard the nuclear ships will consist of primary coolant blow-down water, drainage from the reactor containment vessel, and laboratory wastes. Several experiments are planned, within the scope of the methods of disposal discussed above, to determine the optimum means of liquid disposal. Although no experimental results are as yet available, it is proposed that these liquid wastes be disposed directly to the ocean, if they meet the standards to be set for ocean disposal. If they do not meet the standards, it is proposed that a cleanup system consisting of a filter along with a mixed resin bed be used to reduce the activity in the liquid waste to the required level before disposal to the ocean.

DISPOSAL OF RADIOACTIVE GASES

The radioactive gases to be considered would be those produced by activation of the oxygen in the coolant water, activation of dissolved gases in the coolant water, and fission-product gases. Since the majority of the radioactive gases to be considered have short half-lives, a holdup technique such as that used in the HRE program, in which the gases are held up in charcoal adsorber beds and eventually released to the environment, has been adopted for the N.S. Savannah.

Studies of the solubility and rate of solution of krypton in sea water would be of value, since the long-lived Kr^{85} activity will exit from charcoal beds. There is a possibility that the solubility of krypton in sea water would allow disposal to the ocean rather than to the atmosphere. Such a disposal system would decrease the possibility of contamination of the immediate atmosphere of the ship.

3. HEALTH PHYSICS MANUAL

The Oak Ridge National Laboratory is preparing a Health Physics Manual for the N. S. Savannah which will provide effective radiation protection measures for the passengers and crew of the vessel. It is intended that these procedures employ the best practices and assume the most modern equipment with sufficient flexibility to establish optimum health physics procedures for routine operation of future nuclear-powered ships, as well as the N. S. Savannah.

The initial procedures will cover all shipboard operations including shipboard maintenance operations and accidents involving radiological hazards. Procedures will subsequently be developed for dockside operations such as fuel reloading, waste disposal, and major maintenance. At this time the outline of the procedures covering all shipboard operation has been prepared and about 25% of the text has been drafted. The first draft is scheduled for completion in February. The outline is presented below.

I. General

  A. Introduction to Health Physics

    1. Explanation and motivation
    2. Statement of problems – detection, control
    3. Units of exposures
    4. Limits applicable
    5. Instrumentation

  B. Scope and Objectives

  C. Responsibilities

II. Description of Reactor and Associated Equipment

  A. Reactor design – fuel, power levels, etc.

  B. Controls and instrumentation

  C. Containment vessel

  D. Shielding

  E. Waste disposal, liquid, gaseous, packages

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  B. Internal Exposure, air and water

  C. Surface Contamination

  D. Waste Disposal

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  F. Emergency Values

IV. Health Physics Program for Exposure Control

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  B. Facilities, Space, and Equipment

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    5. Sampling and Assays
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    1. Dosimetry
    2. Protective Equipment and Laundry
    3. Surveys and Bio-assays
    4. Waste Disposal

  D. Emergencies

VI. Specific Cases

VII. Appendices – Technical and Specific Data

4. PRESSURIZED-WATER IN-PILE TEST LOOP

A pressurized-water in-pile test loop is being designed and constructed for investigating the performance of fuel elements proposed for the N. S. Savannah reactor at simulated reactor operating conditions in a radiation field. The reactor contractor has constructed an in-pile loop which is to be used for a limited test program and then transferred for use on another project. The loop described here is to supplement the contractor's test program for the first merchant ship and to be available for future development programs.

The test loop will provide a facility in which fuel and other materials can be subjected to radiation and other environmental conditions that simulate those of a pressurized-water reactor. The objectives of the test program are (1) prototype testing of fuel pins proposed for use in the first nuclear-powered merchant ship (N. S. Savannah), (2) developmental testing of other fuel element designs, (3) developmental testing of structural and control materials of interest in water-moderated reactors, (4) the accumulation of data for studies of water-chemistry and activity buildup in a pressurized-water system, and (5) the accumulation of data for other studies of a basic or applied nature which require the combined environment of radiation and high-temperature water.

An investigation was made of the relative practicability of loop operation at the MTR and the ETR at the National Reactor Testing Station in Idaho and at the ORR in Oak Ridge. The ORR was chosen for availability of space and degree of control over installation, maintenance, and operation of the test loop. Lattice positions A-1 and A-2 have been chosen for the in-pile location of this experiment. The unperturbed thermal neutron flux at these points in the reactor is 5×10^{13} n/cm^2·sec.

DESIGN CONSIDERATIONS

It was originally thought that copying the reactor contractor's test loop would be the most expedient means of obtaining the loop design but investigations revealed that the loop was too small to utilize fully the flux and space capacity of the ORR. Other test loops which are comparable in size and designed for operating pressures and temperatures in the desired range were therefore studied. Argonne National Laboratory, the General Electric Company and the Westinghouse Corporation have built and operated such test loops. Drawings, performance data, and supplementary information on these loops were obtained.

The ORR loop is being designed to recirculate water at pressures as high as 2250 psi at temperatures as high as 625°F. The initial in-pile section is to operate at the N. S. Savannah reactor conditions of 1750 psi and 500°F, and subsequent tests involving higher pressures and temperatures will require a different in-pile section. The water will be circulated past two sets of test samples, each consisting of six fuel pin specimens. One set will be located in the in-pile portion of the loop and the other in a section of the loop external to the reactor for control purposes. The remainder of the equipment required to maintain recirculation at the desired operating conditions will be located within a cubicle in the basement of the ORR building. The test loop is being constructed of type 347 stainless steel to minimize the effects of corrosion and to eliminate the necessity of heat treatment of welds. The in-pile pressure tube is being made of type 316 stainless steel in order to take advantage of its high allowable stress and also to provide a means of distinguishing between corrosion of in-pile surfaces and activation of material removed from out-of-pile surfaces by corrosion. The design criteria for the loop are given in Table 4.1.

LOOP DESCRIPTIONS

The loop consists of an in-pile test section, a main heat exchanger to remove heat from the water, Westinghouse type 150D canned-rotor pumps for circulating the water, 60 kw of electric heat on piping runs for temperature control, a surge tank, a water makeup system, a water purification system, and an out-of-pile control specimen test section. Process flow diagrams of the system are shown in Figs. 4.1 through 4.4. All process equipment, except the in-pile section, makeup pump, and sampling station, is to be located in a shielded room in the basement of the ORR, as shown in Fig. 4.5.

In-Pile Test Section

The in-pile section, to be located in lattice positions A-1 and A-2 of the ORR, consists of a U tube of $1\frac{1}{2}$-in.-ID pipe. This U tube is to contain up to six fuel pin test specimens, three in each leg. The fuel pin specimens, 0.5 in. in diameter, 18 in. long,

Table 4.1. Design Criteria for Pressurized-Water In-Pile Loop and Test Conditions

	Test Loop Description and Operating Conditions	Fuel Element Description and Test Section Conditions	N. S. Savannah Reactor Description and Operating Conditions
Pressure	2500 psig (design) 2250 psig (maximum operation)	1750 psig	1750 psig
Temperature[a]			
In-pile tube inlet		500°F	
Mean		508°F	508°F
In-pile tube outlet	625°F maximum	516°F	
Fluid Circuit			
In-pile tube (U tube)		1.5 in. ID	
Effective flow area		0.393 in.^{2}/pin	0.185 in.^{2}/pin
Velocity, first pass		10.00 ft/sec	9.66 ft/sec
Flow rate	0 to 90 gpm[b]	40 gpm	
System pressure drop			
at 40 gpm		62 ft	
at 90 gpm		305 ft[c]	
Duration of Tests	Total time unlimited	1 month to 1 year	3 years
Fuel Pins			
UO_2 pellets		0.4265 in. dia, 0.5 in. long	0.4265 in. dia, 0.5 in. long
Stainless steel cladding		0.035 in.	0.035 in.
Cladding outer diameter		0.5 in.	0.5 in.
Active length		16.5 in.	66.0 in.
Over-all length		18.0 in.	
Pins per test		6 (3 in each leg of loop)	
Pin configuration[d]		0.612 in., 60-deg pitch	0.612 in., square pitch

Table 4.1 (continued)

	Test Loop Description and Operating Conditions	Fuel Element Description and Test Section Conditions	N. S. Savannah Reactor Description and Operating Conditions
Heat Generation			
At N. S. Savannah reactor average power density of 7.89 w/g		18 kw	
At N. S. Savannah reactor maximum power density of 34.6 w/g		77 kw	
Gamma heating in in-pile tube at a power density of 10 w/g	67.0 kw	67.0 kw	
Total heat generation	150.0 kw[e]	144.0 kw (maximum)	
Heat exchanger capacity (water-cooled)	150.0 kw[e]		
Electric line heater capacity	60 kw		
Loop piping	$1\frac{1}{2}$ in. IPS sched-80 type 347 stainless steel		
Water Chemistry			
pH range	6.5 to 11.0	7.5 to 8.5	7.5 to 8.5
Hydrogen content	0 to 4 ppm	3.6 ppm	3.6 ppm
Total maximum allowable solids		5 ppm	5 ppm
Maximum oxygen content	0.05 ppm	0.05 ppm	0.05 ppm
Maximum chlorine content	0.1 ppm	0.1 ppm	0.1 ppm
Flow rate through purification system	0 to 1.25%	0.20 to 1.00%	0.20 to 0.25%

[a] Loop temperature may be varied by heat exchanger and electric heater controls.
[b] Flow rate may be varied by main throttle valve control.
[c] System head curve intersects Westinghouse Model 150D head curve at this point.
[d] Test pin dimensions may be varied within limitations of in-pile tube.
[e] Heat exchanger rated at 150.0 kw at an inlet temperature of 300°F.

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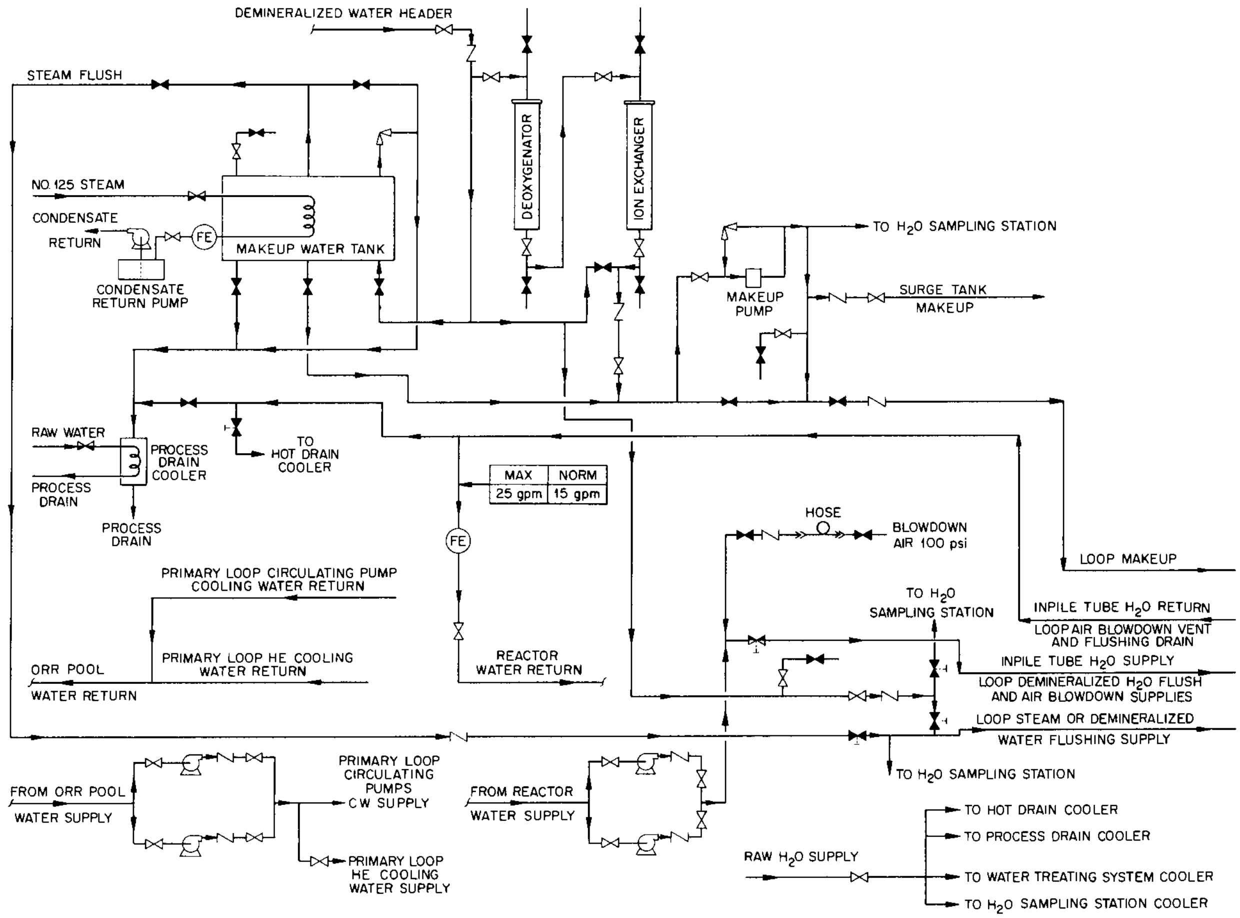

Fig. 4.1. Pressurized-Water In-Pile Loop Auxiliary Systems Flow Diagram.

will be mounted in specially designed holders which will position them in the center of the reactor lattice. Fuel pin specimens having different diameters and lengths may be tested with the use of appropriately designed holders. The cooling water will flow through the U tube and over the specimens, as shown in Figs. 4.6 through 4.9.

The in-pile section of piping will enter the reactor through the refueling flange directly over lattice positions A-1 and A-2. A stainless steel vacuum jacket around the in-pile piping will serve as a thermal insulator. An aluminum sleeve which fits over the vacuum jacket and into a lattice position will be designed to agree with the cross-sectional configuration of an ORR fuel element. Reactor cooling water flow rates will not be altered by the substitution of this in-pile section for reactor fuel elements.

Each leg of the in-pile section will be provided with an O-ring-sealed plug above the reactor to provide for insertion and removal of test specimens. There will be at least 6 ft of reactor pool water over the test specimens during removal, and thus no additional shielding will be required for this operation.

The design pressure and temperature of the in-pile section were set at 2250 psig and 625°F. Combined thermal and pressure stresses will be limited to values specified by the ASME Code.

Control Test Section

Control samples will be exposed to an environment similar to that of the in-pile specimens, except for

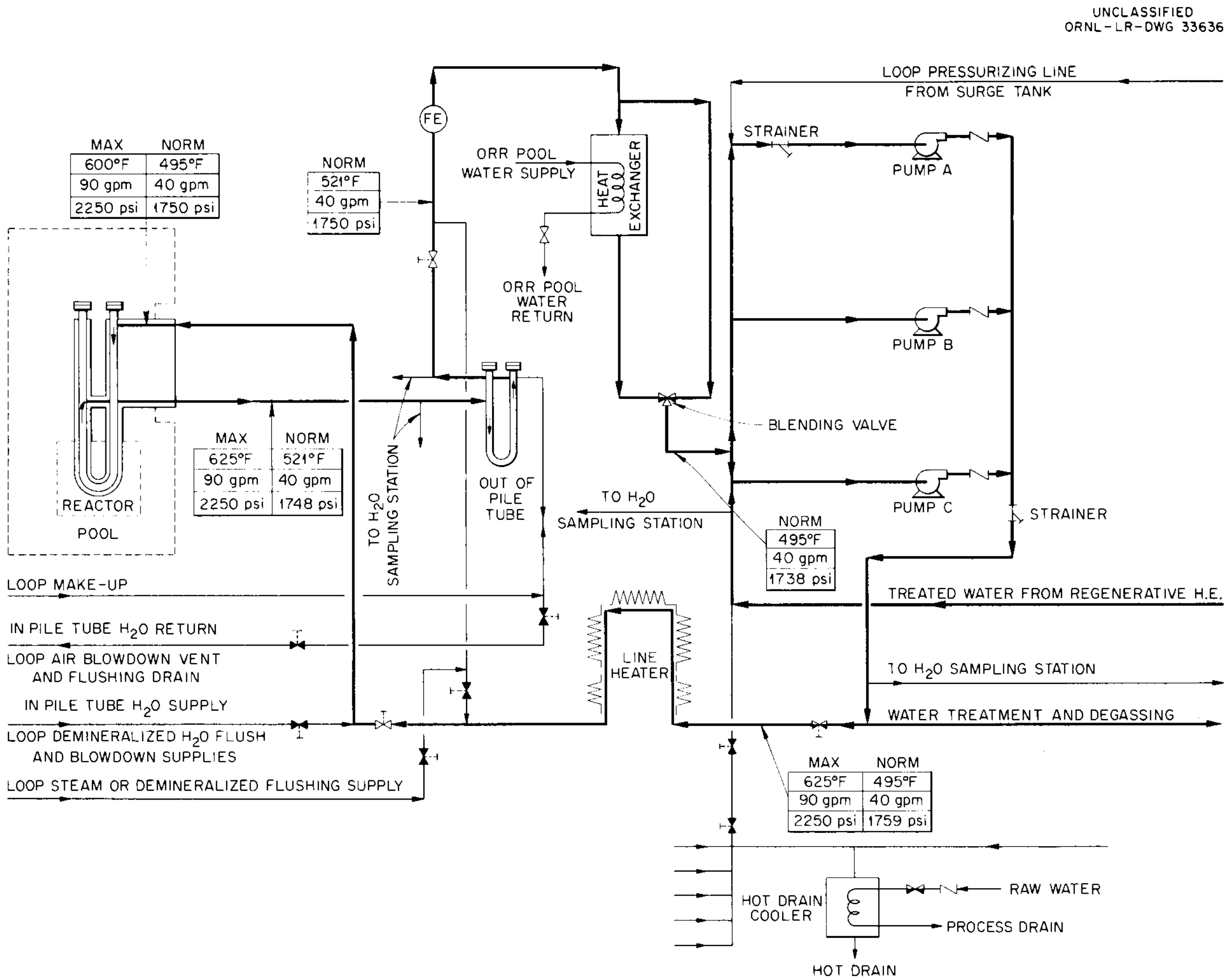

Fig. 4.2. Pressurized-Water In-Pile Loop Primary Loop Flow Diagram.

the absence of neutron and gamma irradiation, and there will be a different temperature differential between the element surface and the water. The out-of-pile test section will be identical to the in-pile section except that it will be horizontal instead of vertical. This section will be located inside the cubicle.

Heating and Cooling Systems

A water-cooled heat exchanger will be used to remove the heat generated within the system by fission and gamma heating and for emergency cooling. Capacity of the heat exchanger will be based on a heat transfer rate of 500,000 Btu/hr with the system operating at a reduced temperature of 300°F. The unit will therefore have excess capacity at an operating temperature of 625°F.

A direct water-to-water heat exchanger was selected for use in this test loop in order to reduce the cubicle space requirements and to simplify the controls required. Two Griscom-Russell Company No. C-72 heat exchanger units will be used.

The three primary system water pumps will be water-cooled canned-rotor Westinghouse Model A-150D pumps. These pumps are to be in parallel in the system, with one normally operating and the other two in standby condition.

The line heater section will be an especially machined section of pipe with clam-shell-type electrical heaters clamped to its surface. These heaters will furnish up to 54 kw of heat to the system. The line heater capacity of the system will be 60 kw, but the heater output is limited to 54 kw in order to maintain a reserve capacity of

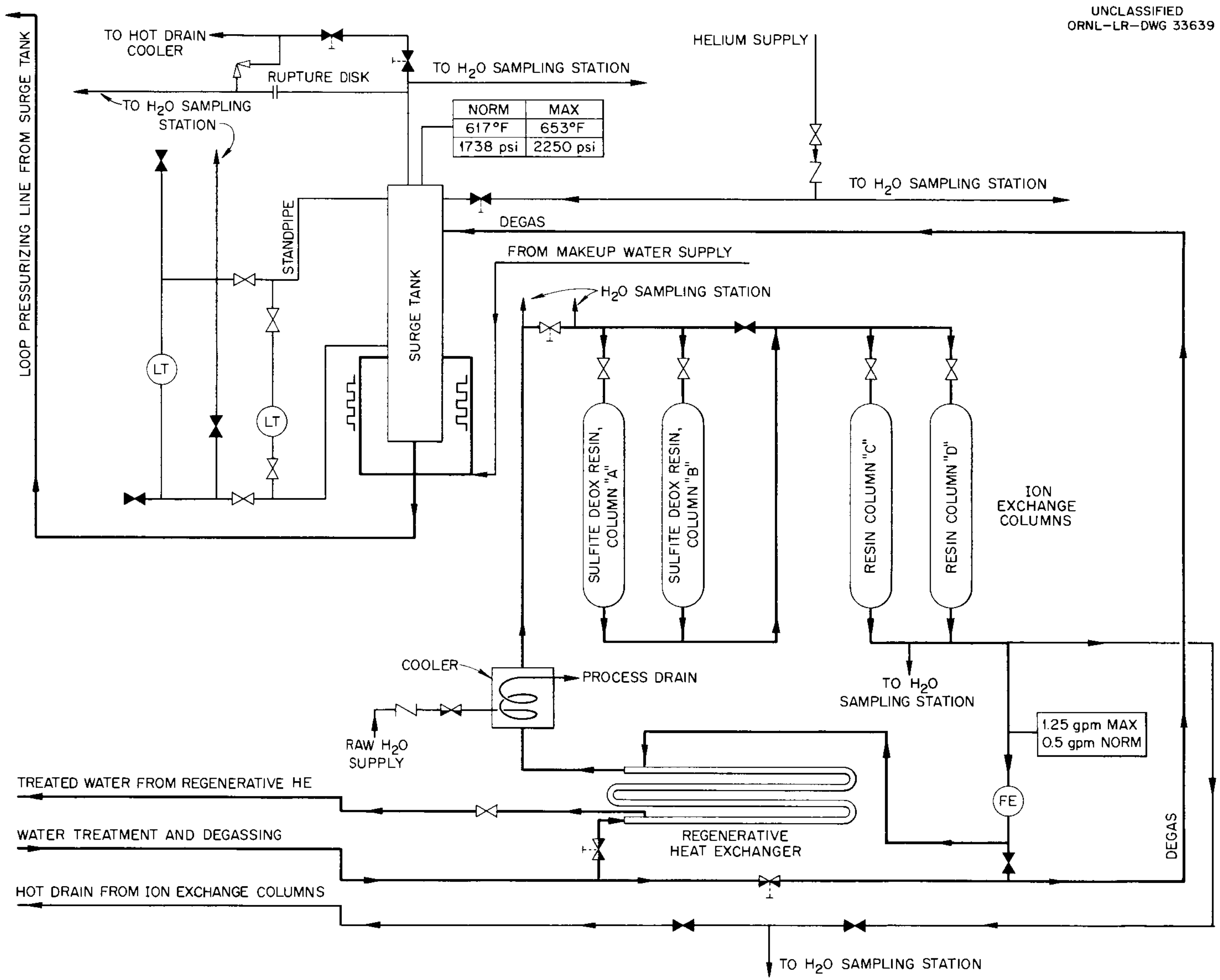

Fig. 4.3. Pressurized-Water In-Pile Loop Pressurizer and H_2O Treatment Systems Flow Diagram.

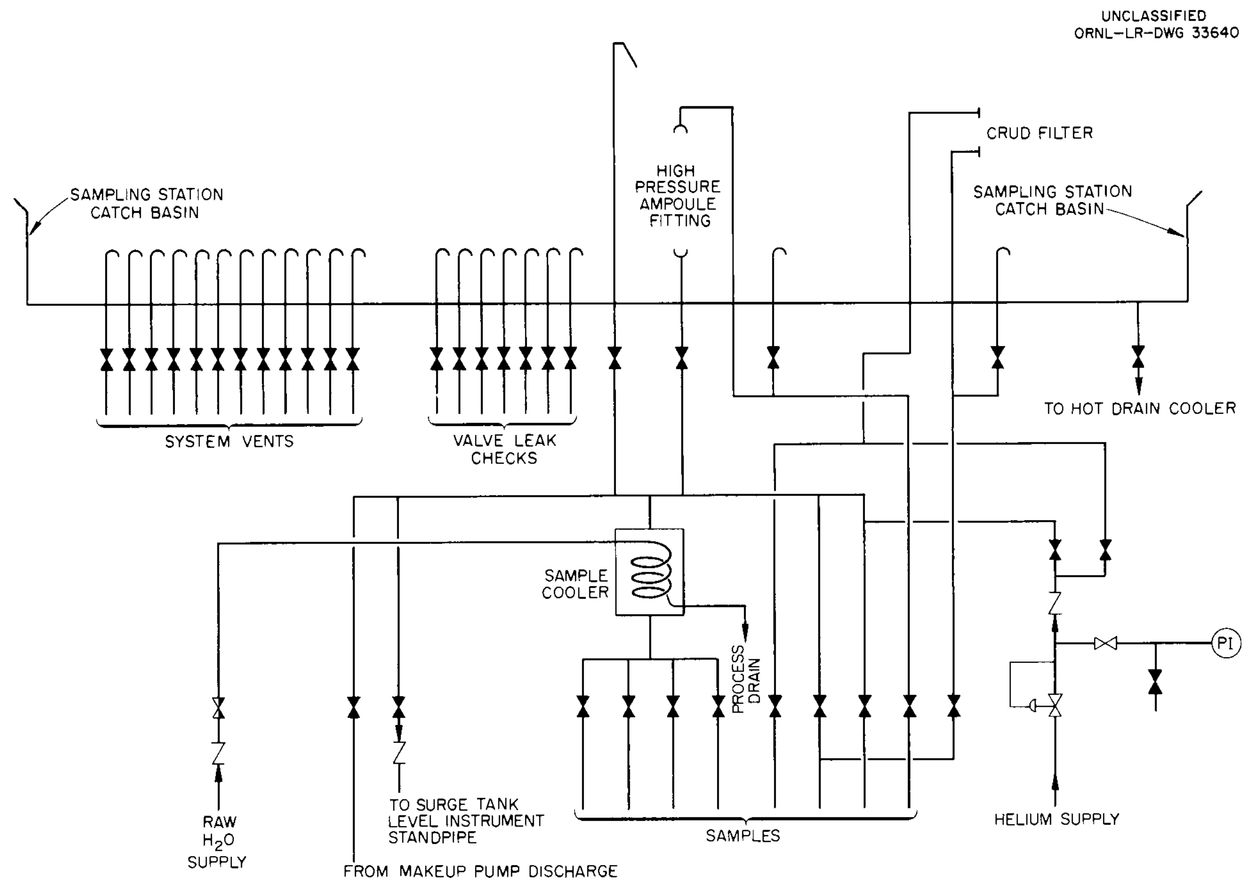

Fig. 4.4. Pressurized-Water In-Pile Loop Water Sampling Station Flow Diagram.

line heater power in the event of the loss of a section of the line heaters. The heat can be added at a controlled rate in order to maintain the desired recirculating water temperature. The clamp-on type of heater is preferred to the immersion type of heater because a leak in an immersion heater could contaminate the system. The heaters will also serve to maintain loop temperatures when the reactor is shut down and thus minimize thermal cycling and startup time.

A surge tank is required to absorb volume changes during heating or cooling of the system and also to provide and maintain the desired system pressure. The volume of the surge tank will be large enough to handle changes in water volume resulting from the contraction of the water that might occur during emergency cooling of the entire system from operating conditions. The surge tank heaters are to be clamped to four pipe legs which extend from the sides of the tank and re-enter the tank at the bottom. The heaters are to be of the clam-shell type used for the line heaters. This method of heating the surge tank water will be used because high thermal stresses would be encountered in attempting to heat the vessel itself.

The makeup system will provide purified water for loop filling and replace loop losses during operation. Demineralized water, furnished by the ORR reactor utilities system, is to be used as the makeup water. This water will enter the makeup tank, where it will be heated to 300°F and then degassed by blowing off about 10% of the water. After degassing, the water will pass through a deoxygenating system before being added to the loop. The deoxygenating system consists of two resin beds and a conductivity cell. The first bed is a sulfite deoxygenating resin and the second an ion-exchange resin. The condition of the ion-exchange resin bed will be monitored by the conductivity cell. The effectiveness of the deoxygenating resin is to be checked periodically by performing an oxygen analysis of the makeup water. All makeup water added to maintain

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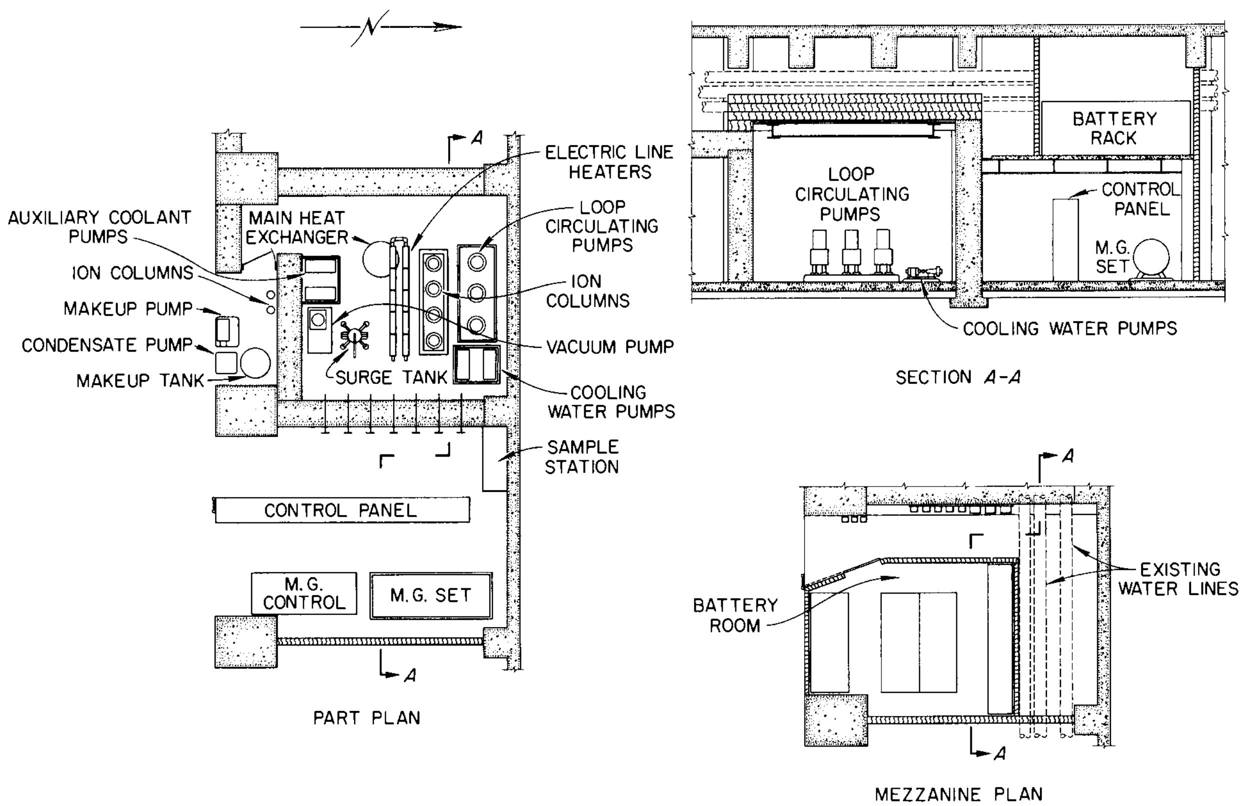

Fig. 4.5. Equipment Arrangement Plan and Section of Pressurized-Water In-Pile Loop.

the proper surge tank level will pass through the deoxygenating system. The deoxygenating system will be bypassed and the water taken directly from the makeup tanks when large amounts of water are required, such as after the loop has been drained. The makeup tank will have sufficient capacity to fill the entire loop with degassed water on the initial startup.

The makeup water will be injected into the system by a Milton Roy Company 18-8 stainless steel Simplex Controlled Volume MDI-53-74-5 pump. This pump is also to be used for pressure testing of the loop.

Water Treatment Systems

A bypass type of ion-exchange system will be provided for the control of water purity in the loop. A portion of the recirculating water is to be diverted from the main system to the ion-exchange resin bed. The water will pass through a regenerative type of cooler and a water-cooled heat exchanger before entering the ion exchanger. The entering water will have been cooled down to a temperature of approximately 95°F in order to prevent damage to the ion-exchange resin. The water leaving the column will pass through the regenerative heater before being returned to the loop. Conductivity cells will be placed before and after the ion-exchange column to measure its efficiency.

A bypass degassifier system is to be provided for controlling the concentration of dissolved gases in the recirculating water. This system will be used mainly during startup and after additions of large quantities of makeup water. A portion of the water will be taken from the loop and fed into a degassifier located in the vapor phase of the surge tank. The gas will be removed from the water by the scrubbing action of the steam as the water passes through the degassifier nozzle. The degassed water will then return to the circulating

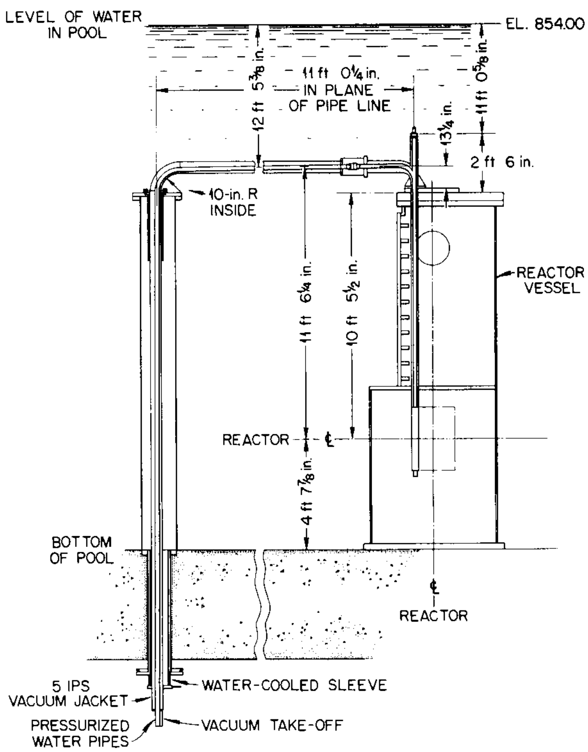

Fig. 4.6. Elevation Drawing of Pressurized-Water In-Pile Test Loop.

system through the pressure equalizing line from the surge tank to the system.

The sample station or system will provide a means for obtaining water samples during operation from various points in the system, such as the pump inlet, in-pile inlet, surge tank bottom, ion-exchange column, etc. All the sample lines which will contain water over 200°F while the loop is in operation will run through a small cooler before entering the sample station. Any test or measurements to be run in the sample station must be done at atmospheric conditions. If high-pressure and high-temperature samples are desired, the sample cooler will be bypassed, and each sample will be taken in a special high-pressure sample bomb. The high-pressure bombs are also to be used for the addition of special materials for controlling the loop-water chemistry. Water is to be circulated through the bomb by using the differential pressure obtained from sample points at the pump suction and discharge. Evacuated sample bombs may be used as required.

In addition to the ion exchanger system, there are to be two full flow strainers to collect foreign particles from the system. One strainer will be located at the inlet to one of the pumps to be used at start-up for collection of extraneous particles and then for standby only. Another strainer will be located in the line downstream of the pump discharge header as protection for the in-pile section in case of pump impeller disintegration.

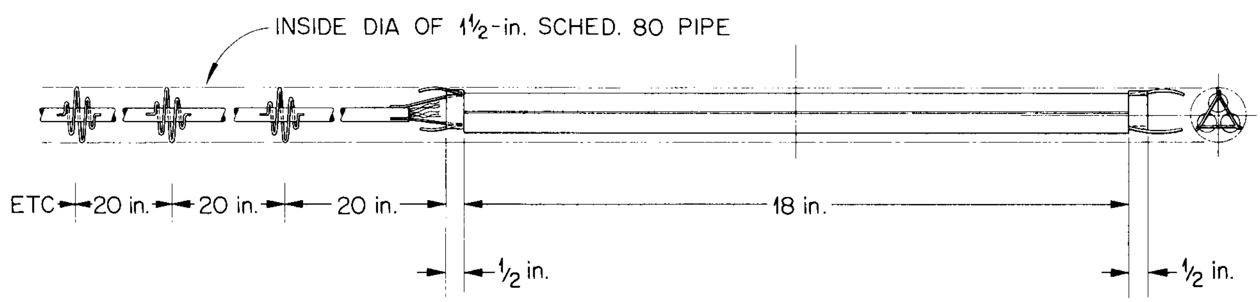

Fig. 4.7. Fuel Pin Specimens.

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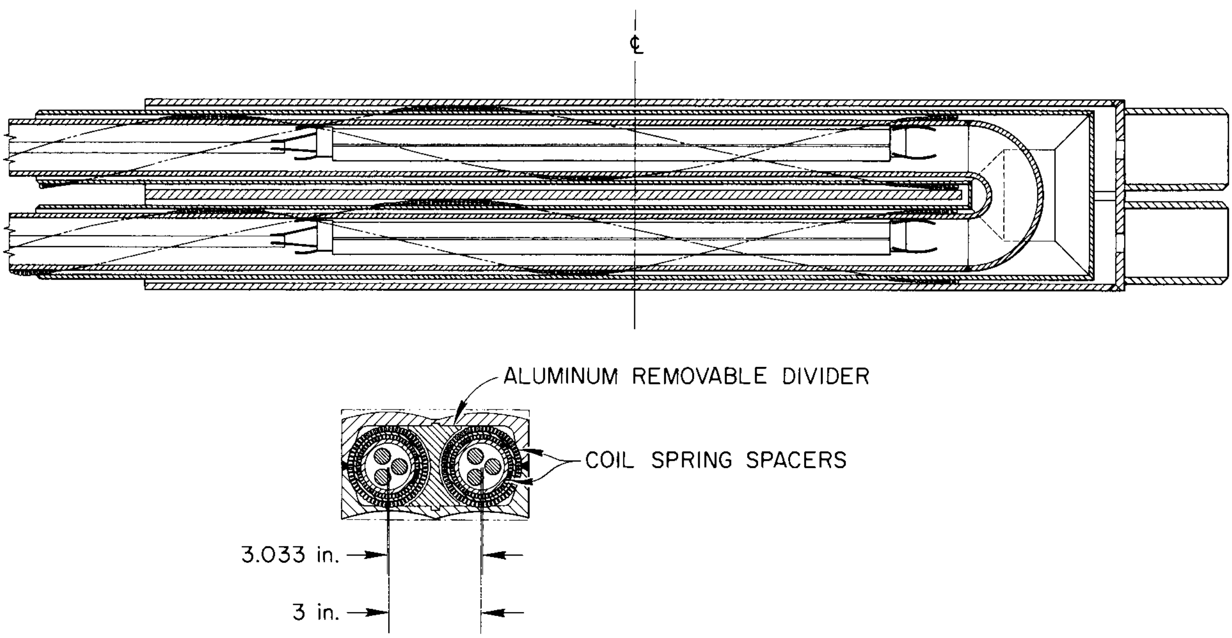

Fig. 4.8. In-Pile Assembly of Pressurized-Water In-Pile Loop.

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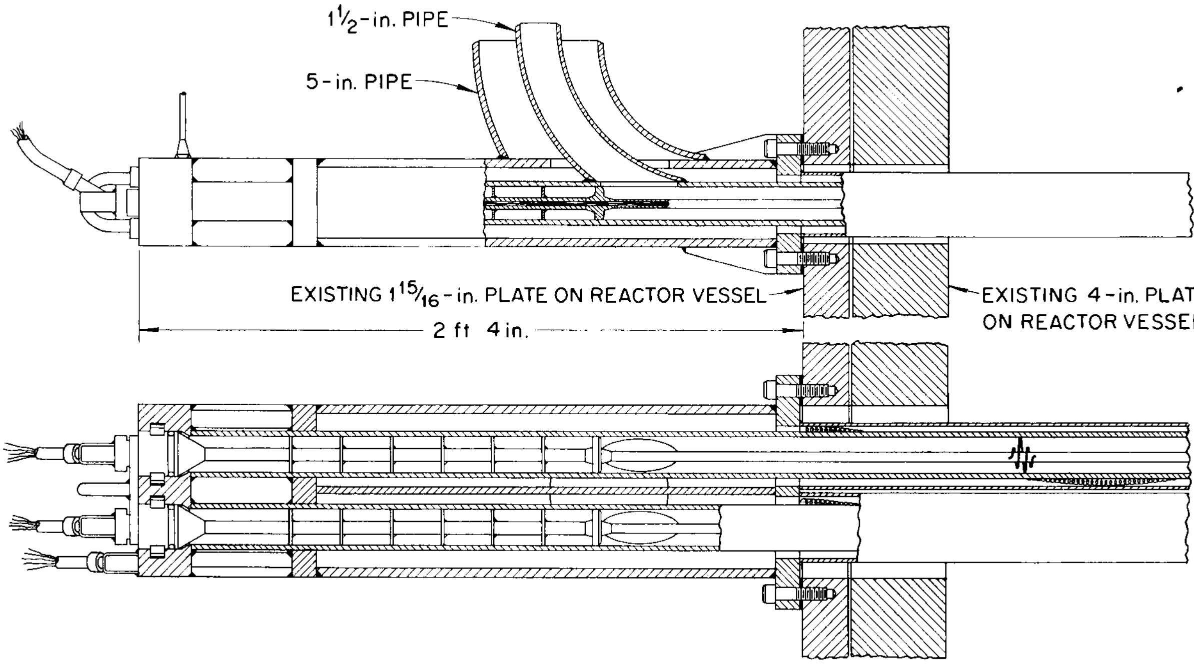

Fig. 4.9. Manifold Above Reactor Vessel for Pressurized-Water In-Pile Test Loop.

INSTRUMENTATION AND CONTROLS

The instrumentation on the loop will be designed to (1) protect the reactor and loop from damage at all times, (2) to provide as fully automatic control of the loop as practical, and (3) to indicate or record all necessary data. The instrument panel boards will contain instruments for the control, measurement, and monitoring of pressure, temperature, flow, surge-tank level, water activity, and saturation. Flow rates will be manually controlled. Consideration must be given to determining the most effective monitoring system to detect failures in the fuel elements under test.

Pressure and Temperature Measurement and Control

The loop pressure is to be maintained by the steam pressure formed in the surge tank. The head of steam is formed by the 24-kw capacity of electrical heaters of the clam-shell type which are strapped to pipe legs on the bottom of the surge tank, as mentioned above. The pressure will be controlled by a pressure controller and a pneumatic-positioner-operated Variac which controls the power to the heaters. The current to the heaters is to be measured by an ammeter. The maximum power is to be limited to 20 kw for safety purposes. The pressures in the surge tank, in-pile section, and pump discharge manifold are to be monitored by pressure transmitters. The in-pile pressure-monitoring system will include dual transmitters and scram circuits to afford maximum protection.

Other pressure gages will monitor the pressure of the process water, demineralized water, flushing steam, and makeup water. There will be differential pressure gages on the panel board with which to measure the pressure drop across the pumps, in-pile section, out-of-pile section, and surge-tank water level.

The loop temperature will be controlled by a water-cooled heat exchanger and electric heaters on the loop piping. Control of the heat exchanger bypass flow rate will provide only an approximate temperature adjustment. More exact temperature control will be maintained by the electric line heaters. In order to utilize the line heaters as the temperature control, an excess of heat will be removed by the heat exchanger. The control of the electrical power to the line heaters will originate from a potentiometer type of indicator controller which receives its input signal from an iron-constantan thermocouple at the outlet of the line heater section. The temperature controller operates a pneumatic-positioner-operated Variac which controls the heater power. This system should maintain the loop temperature with very little variation.

A portion of the line heater capacity (24 kw) is manually controlled and is to be turned on in two steps of 12 kw each. Fine adjustments of power requirements will be maintained by the pneumatic-positioner-operated Variac.

Thermocouples are to be located at all points where it is considered necessary to monitor or record temperatures. The one thermocouple that reads the highest temperature of the in-pile tube is to be connected to a single-point temperature recorder to give a continuous record of the in-pile tube temperature. The recorder will have a scram switch which is set at a suitable value above the operating temperature so that if the in-pile pressure tube temperature rises to a dangerous value, a reactor scram will be initiated.

Safety Features

Circulation of water must be maintained in the loop at all times to cool the test elements and in-pile tube. In order to maintain the required flow with an adequate measure of safety, three pumps are to be provided. The pumps are to be in parallel with one operating and the other two as standby. The flow is to be controlled by a hand-operated flow-control valve and measured with a venturi and flow recorder. Duplicate differential pressure (DP) cells for measuring the flow will be required for safety considerations. Two pressure switches are to be operated from each of the flow signals emitted by the DP cells. One pressure switch is to produce an alarm and turn on the first standby pump at low flow conditions and the other is to produce a reactor scram at a very low flow condition.

The cooling-water flow rates to the system water pumps and to the ion-exchanger system are to be measured by rotameters.

Provisions will also be made in the loop for the switchover to reactor cooling-water circulation in the event that circulation of water through the loop is stopped. Water flow past the sample is required at all times when the reactor is at power to prevent burnout of the sample and the pressure tube. The reactor cooling water is the water that flows through the active lattice and other parts of the reactor that require water cooling. Reactor cooling water use is to be kept to a minimum, since it would have undesirable effects upon the controlled conditions of the test and might also accelerate the corrosion in the in-pile tube. The switch-over to reactor cooling-water operation must be done while the reactor is shut down. The switch is to be made by closing the valves which isolate the in-pile section from the remainder of the loop, opening the process water entry and return valves, and starting one of the two reactor cooling water pumps.

When the loop is operating on reactor cooling water, the flow will be measured by another orifice plate or venturi and DP cell. For safety considerations the system of reactor cooling-water flow measurement, signal transmission, recording, and pressure switch for the scram circuit is similar to that of the main water-flow system, except for the pressure rating of the DP cells. Two reactor cooling water pumps will be installed in parallel, with one pump to serve as a standby.

The main loop pumps are to be connected to a battery-driven motor-generator set and the reactor emergency power system to ensure against interruption or loss of electrical power. The battery-driven motor-generator set is necessary to supply power during the period of time required to put the reactor emergency power system into operation. The 240-v battery system will be located over the operating area and the motor-generator set will be located directly behind the instrument panel.

The line heaters will not be connected to the reactor emergency power, since an interruption of heater power is not considered to be a serious situation. The tank heaters are to be connected to the battery emergency power.

Liquid Level

The surge-tank liquid level is to be indicated and recorded on a DP cell recorder. The signal to the recorder is to be transmitted by two tubing runs from the surge tank. The signal to the high side of the instrument will come from a standpipe connected to the top of the surge tank that is the reference, or cold, leg. The reference leg will be cooled and maintained full at all times to provide a constant water level and density. The signal to the low side of the instrument will come from the bottom of the surge tank and will vary with the level of the water in the tank. A slight correction must be made in the level readings to compensate for density differences between the reference and measuring legs.

Activity

The activity of the main system is to be monitored and recorded. The recorder is to be equipped with an alarm point so that in the event of an excessive increase in water activity, such as might be encountered during a fuel element rupture, the operators will be notified and the loop will be shut down. Provision will also be made for monitoring activity in the purification system, cubicle exhaust vent, cubicle sampling station, and operating area.

Automatic Controls

Automatic control circuits will initiate corrective action to rectify extremely critical abnormalities when they occur and, if necessary, will initiate a reactor scram. The abnormal conditions which will cause an annunciation and the corrective action that will be taken are described below.

1. In the case of low primary system water flow, the operating pump will be stopped and an automatic switchover will be made to the first standby pump. If normal flow does not resume in 2 sec, the reactor will be scrammed. The first standby pump will then be stopped and the second standby pump will be started. If flow does not return to normal within the next 2 sec, the line heaters will be turned off and the heat exchanger will be opened to full capacity.
2. In the case of low pump cooling-water flow, the operating cooling-water pump will be automatically stopped, the standby pump will be started, and the line heater will be shut off. If the flow does not return to normal within 2 sec, the reactor will be scrammed, and the heat exchanger will be turned on to full capacity.
3. In the case of a high ion-column temperature, there will be only an alarm.
4. If steam saturation conditions are approached during a decrease in loop pressure or an increase in loop temperature, the line heater will be shut off. If the condition continues, the heat exchanger will be turned on to full capacity, and an alarm will be sounded. If the loop water continues to approach the steam saturation point by pressure drop or temperature rise, the reactor will be scrammed. Alarm set points located in the pressure and temperature recorders will be manually set for specific operating conditions. During startup and shutdown the alarm set points will require step adjustments by the operator.

5. If the system pressure becomes too high, the surge-tank heaters will be shut off. They will be automatically turned on when the pressure returns to normal.

6. The makeup pump will be turned on and off automatically to maintain the water level. If the level drops below the desired operating level, the surge tank heaters will be automatically turned off. If the level becomes too high, the makeup pump control circuit will be broken to prevent the addition of water to the system by manual controls.

7. An indication of electrical overload on the operating water pump will transfer load to a standby pump.

8. In the case of high water activity, there will be only an alarm.

9. When the temperature of the in-pile tube exceeds a preset point, the reactor will be scrammed.

10. When the in-pile tube is using reactor cooling water, a low reactor cooling water flow rate will actuate an alarm. If the flow rate should continue to drop, a second alarm point will be reached and the reactor will be scrammed. The operating cooling water pump will be automatically stopped, and the standby pump will be started.

Relief Valves

There are to be six relief valves on the system to provide protection for (1) the surge tank in the event of a failure of the automatic control circuit which governs high pressure, (2) the in-pile tube in the event this portion of the system is isolated from the remainder of the system and there is sufficient heat generation in the reactor tube to cause excessive pressure, (3) the makeup tank in the event of failure of the heater controls, (4) the makeup pump to prevent excessive pressures if the pump is operated while the discharge valves are closed, and (5,6) each of the reactor cooling water lines.

OPERATION AND TEST PROCEDURES

The loop operating personnel are to be protected from radiation from the water activity by placing all the external equipment inside a cubicle which has 24-in.-thick barytes concrete walls or their equivalent. The piping connecting the reactor tube to the external system is also to be shielded with 24 in. of barytes concrete or its equivalent. The cubicle will be enclosed and connected to the reactor stack exhaust system so that, if there is a leak or rupture inside the cubicle, the radioactive steam will be vented to the stack instead of contaminating the reactor basement or causing an evacuation of the building because of high air activity.

When a test specimen is to be discharged or removed from the reactor tube, the in-pile piping will be isolated from the remainder of the system by closing isolation valves. The plug-locking device will be removed from the top of the in-pile section and fixtures will be attached to the O-ring-sealed plugs. Test specimens attached to each of these plugs by specimen holders will be withdrawn from the in-pile section by the upward movement of the plugs. The specimens will be shielded by the pool water and no additional shielding will be required. Specimens will be stored in the reactor pool until ready for inspection within the hot cell. Reactor pool water and test loop water will be free to mix during fuel pin charging or discharging operations.

The out-of-pile specimens can be withdrawn in the same manner except that they will be drawn into a length of pipe to prevent spread of contamination. Since the out-of-pile elements will not be irradiated, the only activity present will be that from radioactive corrosion products on the element surfaces.

The fuel-element specimens will be similar to the N. S. Savannah reactor fuel elements, except that the length will be 18 in. instead of 66 in. The specimen outer diameter will be 0.500 in., and the specimens will be spaced at 60-deg intervals on 0.612 in. centers. This configuration was chosen to permit three pins to be tested simultaneously in each leg of the in-pile section. Irradiation tests will be performed with a water velocity of 10 ft/sec across the test specimens in order to duplicate the N. S. Savannah reactor water conditions with the use of the flow control valve.

The instrument panel will be located facing the cubicle wall so that valve handles can be extended through the wall and operated in view of the panel. This arrangement can be accommodated in the space allocated for the experiment.

The loop and its control system are being designed to minimize attention by the operating personnel. To provide for a minimum of coverage, it is important that manual operations be limited to data taking, the monitoring of conditions, and the removal of samples for analysis. It may be desirable or necessary to arrange for backup assistance from reactor operating personnel in the event an emergency situation arises.

Fuel elements removed from the loop will be loaded into casks for removal to a suitable hot cell for examination. This transfer can be accomplished either in the reactor canal or in the ORR canal hot cell. Removal of the specimen holder will be necessary before closing of the cask.

Examinations of the specimens will probably include dimensional measurements for swelling or distortion, metallographic examination of sections, analyses of fission gases, removal of the uranium oxide pellets and examination for cracks, other damage, and burnup. Analyses of induced activity of structural materials or foils may be required occasionally in order to determine radiation doses. Thermocouple attachments will be examined after irradiation. Since the fuel elements specimens will have relatively low enrichment, the cells for opening the capsules and examining the uranium oxide must be capable of handling plutonium.

FACILITY STATUS

The design of the test facility is essentially complete with the exception of some minor details which will not affect the installation schedule. Procurement action has been initiated for the major components, and delivery is expected as scheduled. Two of the three Westinghouse Model A-150D pumps have been delivered. Some minor items, such as the vacuum pump, condensate pump, and some of the pipe and pipe fittings, have also been delivered. Fabrication of the instrumentation and controls panel boards has begun.

Construction work on the ORR building alterations is under way and is to be completed by February 1, 1959. Loop installation is scheduled to begin at the completion of the building alterations.

5. METALLURGICAL STUDIES

JOINING OF FUEL ELEMENT TUBES AND FERRULES

The reference design for the N.S. Savannah fuel elements specifies a 6 by 7 array of 6-ft-long, 0.500-in.-OD, 0.035-in.-wall, type 304 stainless steel fuel tubes spaced by 1-in.-long hollow stainless steel ferrules on an 8-in. pitch. The tubes are filled with slightly enriched UO_2 pellets sintered to 91% of theoretical density. Joining the ferrules and fuel tubes into an integral bundle is accomplished by brazing in hydrogen with Nicrobraz 50, a nickel-base alloy containing 13 wt % Cr and 10 wt % P.

A metallographic study was conducted to determine whether the selected brazing alloy would undercut the tube wall significantly and also whether extensive phosphorus diffusion from the braze metal would occur, with consequent impairment of the stainless steel. To implement this study, tube-to-ferrule joints were prepared with the use of Nicrobraz 50 on type 304 stainless steel. The several brazing conditions described below were used:

Sample No.	Brazing Conditions
1	½ hr at 980°C
2	½ hr at 1010°C
3	½ hr at 1010°C, cool, and repeat heating
4	½ hr at 1040°C
5	½ hr at 1065°C

No appreciable undercutting of the tube walls was observed in any of the samples. Some of the braze-metal fillets contained hairline cracks, however, that revealed the brittle nature of the selected brazing alloy, as illustrated in Fig. 5.1.

Microhardness traverses were made across the brazed joint and adjacent base metal in each of the specimens to determine whether the properties of the stainless steel were affected. The data obtained indicated that the base metal was essentially unaltered by the brazing operation.

The feasibility of making sound joints with electroless nickel as the brazing alloy is also being studied. Electroless nickel is a chemical deposit of nickel containing 7 to 11% P, which has potential advantages in comparison with Nicrobraz No. 50. First, the brazing alloy can be preplaced by plating the ferrules prior to assembly. Also the electroless nickel fillet will be less susceptible to hairline cracking.

Studies to date have dealt primarily with the development of plating methods for rapidly depositing coatings of high phosphorus content in order to obtain an alloy with a minimum liquidus temperature. Small test samples have shown that sound tube-to-ferrule joints can be obtained by using electroless-nickel-plated ferrules and brazing for 15 min at 980°C in hydrogen. As was the case with the Nicrobraz No. 50 joints, no significant undercutting or tube-wall hardening was noted. Fabrication of a prototype 4 by 4 fuel bundle by using electroless-plated ferrules has been initiated. Corrosion tests of electroless-brazed components in high-temperature water are also under way.

A semiquantitative test was devised to determine the relative shear strengths of tube-to-ferrule joints brazed under various conditions. The test apparatus, which is illustrated in Fig. 5.2, is designed to simulate effects which would stem from differences in total expansion of adjacent fuel tubes operating at different temperatures. Results of these studies are summarized in Table 5.1. The data indicate that a repeat of the brazing cycle exerts no deleterious effects on the strength of joints brazed with Nicrobraz No. 50 (tests 2 and 3) and that loads of 900 to 1300 lb are required to initiate failure in N.S. Savannah fuel element reference joints (tests 4 and 5). It may also be seen that joints brazed with both electroless nickel and Nicrobraz No. 50 have nearly equivalent strengths in this type of stress system (tests 5 and 6). Failures in all tests occurred in the braze fillet rather than in the fuel tube; thus if a braze joint should fail, the integrity of the fuel tube will not be affected.

The initial design for the fuel tubes specified that the type 304 stainless steel should contain 300 ppm of natural boron to serve as a burnable poison. Since it was conceivable that deboronization could occur during the brazing cycle required to assemble the fuel tubes into bundles,[1]

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[1]J. C. Shyne and E. R. Morgan, *Metals Prog.* 65(6), 88 (1954).

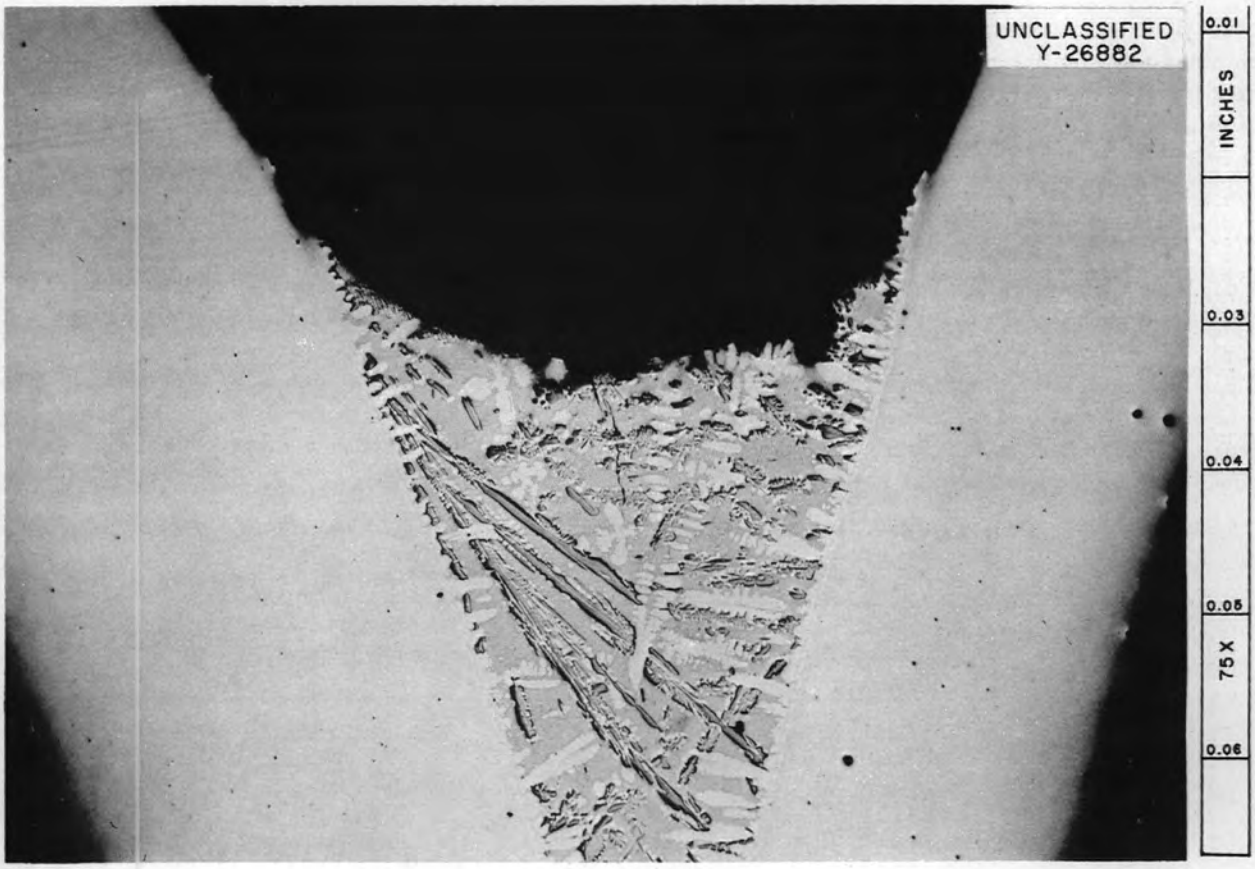

Fig. 5.1. Type 304 Stainless Steel Tube-to-Ferrule Joint Brazed with Nicrobraz No. 50 in H_2 for $\frac{1}{2}$ hr at 1040°C. Note absence of tube-wall undercutting and presence of hairline crack in braze fillet. As-polished.

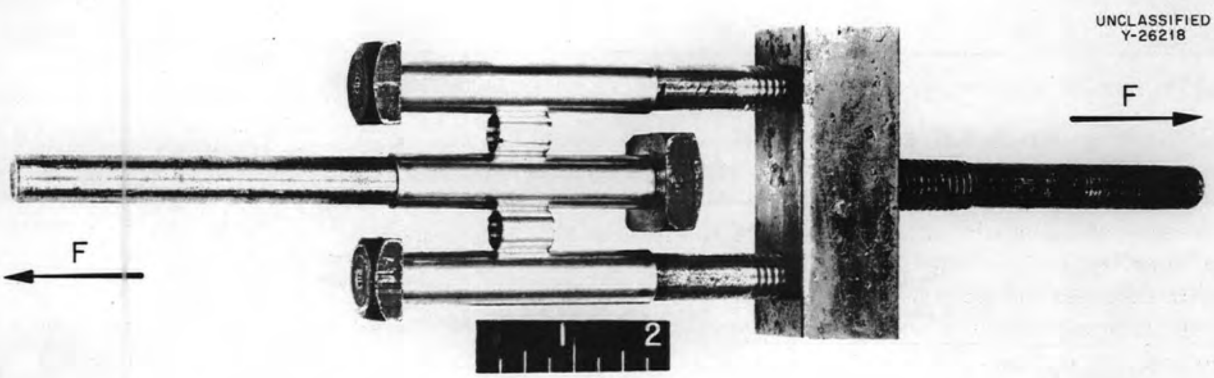

Fig. 5.2. Testing Fixture for Evaluating Strengths of Brazed Joints.

about 30 ft of boronated stainless steel tubing was obtained from the Babcock & Wilcox Company for heat-treating studies. Samples from two 6-ft lengths of this tubing were analyzed at ORNL by wet chemical techniques and each tube was found to contain 240 ppm of boron. The accuracy of the analysis is estimated to be ±10%. Samples from one of these same tubes were also submitted to an independent laboratory (Lucius Pitkin, Inc., New York City) for boron assay. Results obtained on duplicate samples showed 245 ± 8 ppm and 238 ± 6 ppm of boron, in confirmation of the ORNL results.

In order to determine the effect of temperature and brazing environment on the extent of deboronization, samples of tubing were held at 1010, 1065, 1120, and 1175°C for 1 hr in atmospheres of dry hydrogen, wet hydrogen, argon, and vacuum, and then analyzed for boron. The results of this study are summarized in Table 5.2.

The data presented in Table 5.2 indicate that (1) no deboronization occurs at 1010°C in any of the atmospheres investigated, (2) no deboronization occurs in either argon or vacuum in the temperature range from 1010 to 1175°C, (3) significant boron losses occur in hydrogen atmospheres with increasing temperatures above 1010°C, and (4) boron losses in hydrogen atmospheres appear to increase as the moisture content of the atmosphere increases.

Table 5.1. Results of Shear Strength Tests of Brazed Tube-to-Ferrule Joints

Tube and ferrule material: type 304 stainless steel

Test No.	Number of Specimens	Ferrule Size	Brazing Cycle	Brazing Alloy	Average Failure Load (lb)
1	6	10-mil wall, ½ in. long	½ hr at 1010°C	Nicrobraz No. 50	240
2	6	15-mil wall, ½ in. long	½ hr at 1010°C	Nicrobraz No. 50	280
3	5	15-mil wall, ½ in. long	½ hr at 1010°C, cool, and repeat heating	Nicrobraz No. 50	260
4	4	20-mil wall, 1 in. long	½ hr at 1010°C	Nicrobraz No. 50	1330
5*	5	20-mil wall, 1 in. long	½ hr at 1010°C	Nicrobraz No. 50	880
6	3	20-mil wall, 1 in. long	¼ hr at 980°C	Electroless nickel	1030

*The type 304 stainless steel tubes used for these specimens contained 240 ppm B.

Table 5.2. Effect of Atmosphere and Temperature on Boron Content of Type 304 Stainless Steel Initially Containing ~240 ppm Boron

Temperature (°C)	Boron Content (ppm) of Type 304 Stainless Steel After 1 hr			
	In Dry H_2 (dew point, −95°F)	In Wet H_2 (dew point, −40°F)	In Vacuum (<1 μ)	In Argon (dew point, +6°F)
1010	270*	234*	274*	254
1065	197	127	260	258*
1120	175*	130*	250*	258*
1175	115	35*	292*	260*

*Duplicate samples.

WELDING OF FUEL TUBE END CLOSURES

End closures in fuel tubes for the N.S. Savannah core are to be manufactured by edge-fusion welding of an inverted cup to the fuel tubes with the use of the inert-arc tungsten-electrode fusion method. The weldability of this joint, as well as that of two plug-type closures, was investigated. The three joints are illustrated in Fig. 5.3.

Welds joining type 304 stainless steel end caps to both boron-free and boronated (240 ppm) type 304 stainless steel were made by inert-arc welding and inspected for defects by dye penetrant, helium leak testing, radiography, and metallography. Welds produced with each closure design were equally sound and no difference was noted in the quality of welds produced with the boronated and boron-free stainless steel. Typical welded end closures are illustrated in Fig. 5.4. Radiographic inspection techniques were devised to provide for 90% weld inspection on a two-exposure basis or 100% inspection on a three-exposure basis, with a sensitivity of 2 to 3%. Radiography, of course, can only be expected to reveal such defects as lack of fusion and porosity.

Fig. 5.3. Fuel Tube End Closure Designs.

TEMPERATURE MONITORING WIRES

One of the most important operational parameters that must be determined to properly evaluate UO_2 pellet-type fuel elements is the maximum temperature to which the fuel is subjected during its tenure in the reactor. Accurate calculation of this temperature from thermal properties of the system is extremely difficult because of several uncertainties. An experimental technique for determining approximate temperatures is to incorporate into the UO_2 pellets metal wires of known and diverse melting points. By metallographic inspection of irradiated pellets containing selected wires, it may be possible to determine whether or not the metal melted from fission heat. Conclusions can then be drawn as to whether the temperature of the UO_2 at a given radial location reached the melting temperature of a particular wire. The proposed culmination of this experimental program is irradiation in the General Electric Co. boiling-water reactor at Vallecitos, California, of a prototype N.S. Savannah fuel bundle containing temperature-monitoring wires.

Gold, nickel, vanadium, platinum, and columbium have been selected for initial determinations of melting point and compatibility with UO_2 at elevated temperatures. The metal wires, 0.020-in. in diameter, will be inserted in the center of the pellet, as well as at radial distances of one-third and two-thirds of the radius to establish a temperature profile.

In order to obtain standards of comparison for post-irradiation inspection, pellets containing each wire have been heated in inert or reducing atmospheres to the observed melting temperature, as well as to 100°C above and 50°C below the melting temperature. Temperatures were measured with an optical pyrometer and appropriate corrections were applied to each measurement. After heating through the required cycle, the surface of the pellet was metallographically polished, and photomicrographs of the UO_2-wire interface were made at magnifications of both 10 and 100 diameters.

Although platinum was initially selected as one of the temperature monitors, it was rejected because of an apparent reaction with UO_2. A marked lowering of the fusion temperature of the wire was observed, even when heating of the wire was carried out in vacuum. It is impossible to state whether this lowered melting point was actually the result of a reaction with UO_2, per se, or with a volatile constituent such as UO_3 which vaporized during

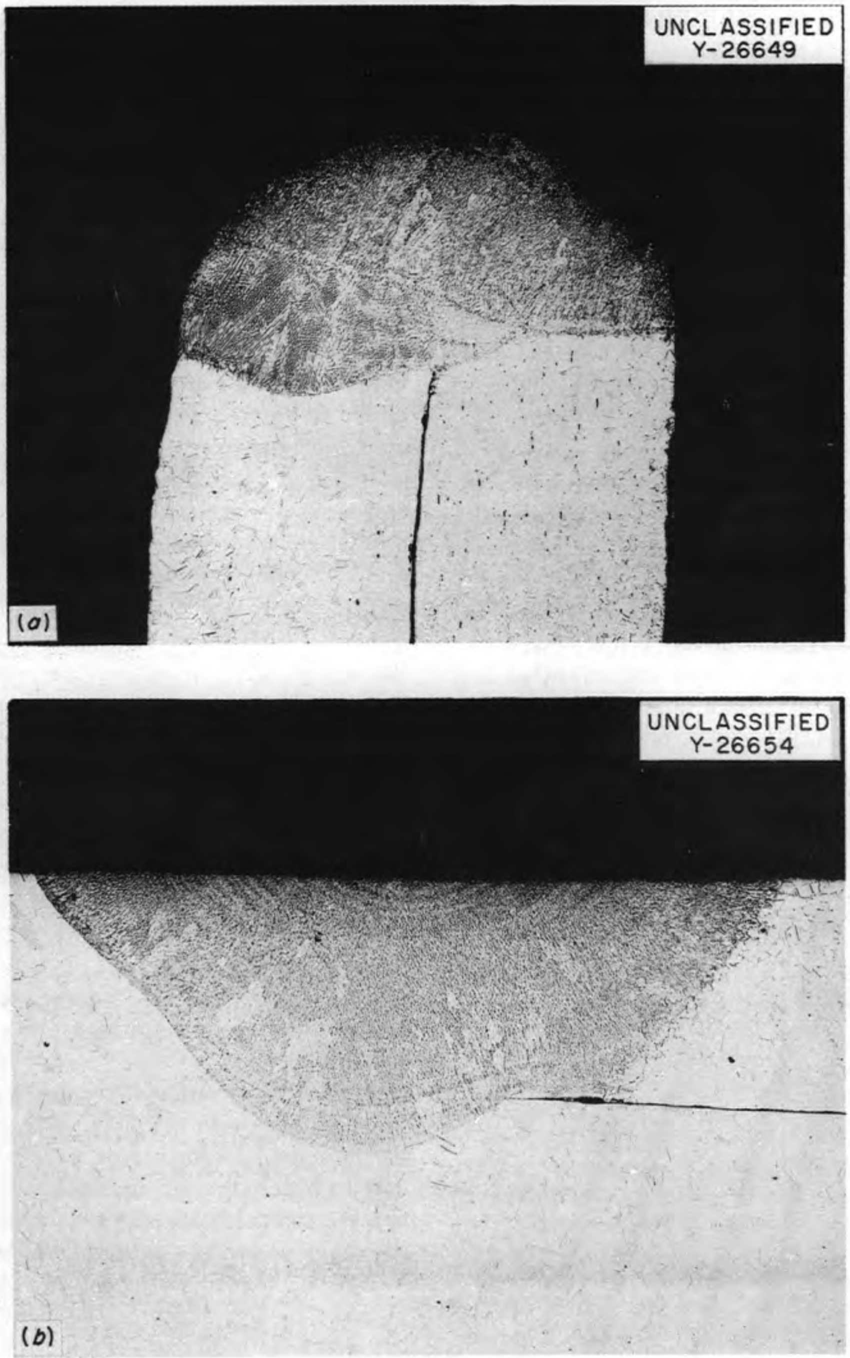

Fig. 5.4. (*a*) Cross Section of Cup-Type Closure in Type 304 Stainless Steel Showing Depth of Penetration and Soundness of Joint. Weld made at 8 amp and torch speed of $4\frac{1}{2}$ in./min using argon cover gas. (*b*) Cross Section of Plug-Type Closure in Type 304 Stainless Steel Showing Depth of Penetration and Soundness of Joint. Weld made at 30 amp and torch speed of 9 in./min using argon cover gas. Etchant: 10% oxalic acid, electrolytic. 50X.

the heating cycle. When UO_2 pellets containing platinum were heated in vacuum, a significant rise in pressure was noted just before the wire melted at about 1400°C. Specimens heated in helium displayed similar melting point anomalies.

No problems were encountered with either gold or nickel wires under the described conditions. Typical nickel wire specimens are illustrated in Fig. 5.5. When UO_2 pellets containing vanadium wires were heated to 100°C above the melting point, the vanadium appeared to wet the UO_2 and to penetrate it intergranularly. Apparently, this does not appreciably alter the properties of the pellets but rather serves to delineate vanadium which had just reached its melting temperature and vanadium which has exceeded the melting temperature by 100°C. Studies with columbium wires, which are still in progress, have been hampered by inability to heat this particular metal to its melting point in a sufficiently pure atmosphere. However, it appears that columbium should be compatible with UO_2 up to at least 2000°C.

INCORPORATION OF BURNABLE POISON IN UO_2 PELLETS

Techniques for fabricating UO_2 pellets containing boron as a burnable poison have also been under investigation. The motivation for these studies was that addition of boron to UO_2 might prove to be a desirable alternative if serious fabrication and chemistry control problems arose with the boronated stainless fuel tubes originally specified for the N.S. Savannah core. A major portion of the effort has been directed toward incorporating 60 ppm of natural boron as ZrB_2 into UO_2. This boride was selected because of its known stability in similar ceramic systems.

The boride was added to the UO_2 and intimately mixed in a rubber-lined ball mill with alumina balls. In the initial studies, 600 ppm of boron was added to facilitate batch preparation and analysis. Pellets pressed from this material were sintered in helium, hydrogen, and a $7N_2$:$3H_2$ mixture at 1700 to 1750°C. Data presented below illustrate the boron retention of sintered pellets that initially contained 600 ppm of boron as ZrB_2:

He	H_2	$7N_2$:$3H_2$
100	5	100
150	5	15
120	<5	200
100	<5	Trace
80	<5	
80	<5	

These data, which were obtained by spectrographic analyses, show that although the boron losses were extremely high in all the sintering atmospheres they were least in helium.

Concurrently with these studies, an investigation was conducted to determine the boron retention in UO_2 to which the burnable poison was chemically added as a controlled impurity during precipitation of ammonium diuranate (ADU). To produce this material, orthoboric acid was dissolved in a uranyl nitrate solution prior to ADU precipitation. Approximately 30% of the added boron was lost during precipitation and subsequent reduction to UO_2. The UO_2, which was ultimately produced, contained 62 ppm of boron. Pellets prepared from this material were sintered at the same temperatures and in the same environments described for the ZrB_2 additions. In every case, the boron content of the sintered pellets was reduced to a level below the limits of detection. Boron-addition studies have been discontinued, since a burnable poison will not be required in the N.S. Savannah fuel elements.

METAL HEAT TRANSFER BONDS IN FUEL ELEMENTS

The feasibility of replacing helium in the gap between the UO_2 pellets and the fuel tube with a low-melting-point metal or alloy is being investigated. If an effective metal substitute can be found, the thermal capabilities of fuel elements of

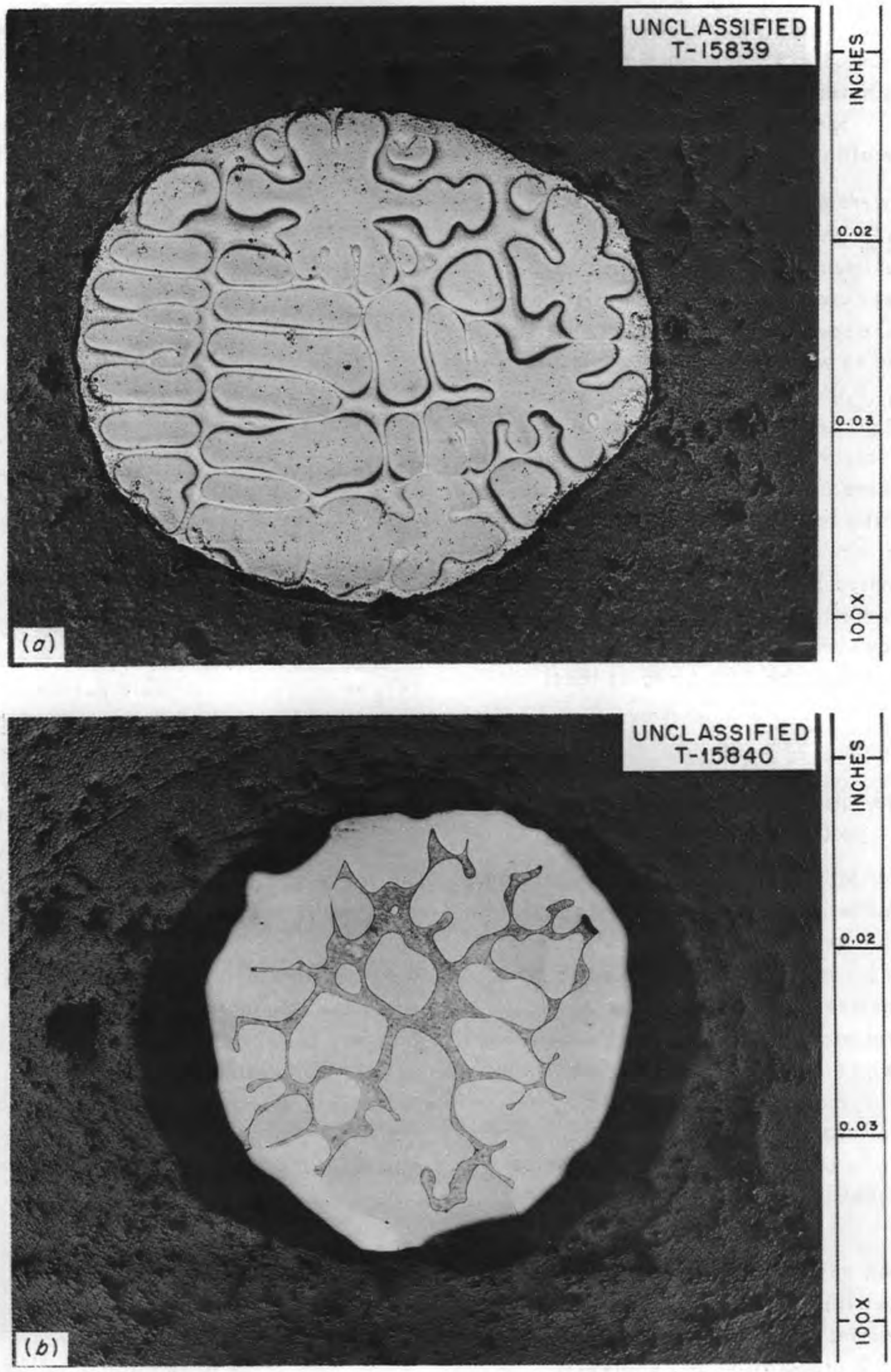

Fig. 5.5. *(a)* **Nickel Pyrometer in UO_2 Pellet Heated to Observed Melting Point (1380°C). Note dendritic structure typical of cast metal.** *(b)* **Nickel Pyrometer in UO_2 Pellet Heated to Just Below the Melting Point and Held for 1 hr. Note signs of incipient failure; wire has not filled cavity. Etchant: cyanidepersulfate.**

the Savannah design can be improved significantly due to the much higher thermal conductivity of the metal bond.

Lead and bismuth alloys were of immediate interest for this application because of their low melting points and low thermal-neutron absorption cross section. To determine the compatibility of selected molten metal alloys with type 304 stainless steel and UO_2, 500-hr corrosion screening tests were conducted at 345 and 400°C by using the following heat transfer bonds: Pb, Bi, 55 wt % Bi-45 wt % Pb, 0.7 wt % Li-99.3 wt % Pb.

Metallographic studies of the pure-lead and pure-bismuth bonds revealed that neither of these materials attacked type 304 stainless steel at either 345 or 400°C. In tests conducted at 345°C, the UO_2 pellets in bismuth suffered a weight loss of 0.06%, while those in lead lost 0.29%. Evaluation of compatibility tests with the Pb-Bi and Li-Pb eutectic alloys is not yet complete. Tests have been initiated to determine the effects of hot spots producing areas of molten lead in an otherwise solid bond.

TESTING OF COMPONENTS OF THE N.S. SAVANNAH PRIMARY SYSTEM

Scrap steel from the upper and lower closure heads of the N.S. Savannah pressure vessel has been received, and material is currently being removed for drop-weight and Charpy V-impact specimens. A correlation of notch impact energy and nil ductility transition temperature will be made using both sub-size and full-size drop-weight systems.

Samples from various components of the primary system have also been received. These samples are being reduced to a form suitable for wet chemical analyses. The samples will be evaluated by an independent laboratory to determine percentages of tramp elements including cobalt, tantalum, silver, antimony, and zinc, which may contribute to the specific activity of the reactor coolant.

6. PHYSICS

Supplementary nuclear calculations in support of and in cooperation with the physics group of the Babcock & Wilcox Company have been made in order to understand some of the basic nuclear design problems faced by the designers of the N. S. Savannah reactor. Criticality calculations of some of the proposed critical arrangements have also served to provide useful information for the other Maritime Reactor Project review groups.

The reactor core is roughly a right circular cylinder about 5 ft in diameter and 5½ ft high. The core is composed of 32 fuel elements and is controlled by 21 cruciform control rods made of 1.1% B^{10} in stainless steel and clad with stainless steel. Each fuel element is about 9 in. square, and the elements are on a 9.7-in. pitch.

A horizontal section through the core is shown in Fig. 6.1. A fuel element consists of a stainless steel can 9 in. square, about 0.100 in. thick, and about 6 ft high containing 164 fuel pins immersed in the water moderator and coolant. The fuel pins are 0.500-in.-OD, 0.035-in.-wall stainless steel tubes which contain 0.4245-in.-dia UO_2 pellets. The pellets are enriched to about 4.4% in U^{235}, and the active fuel height is about 5½ ft. The fuel pins are spaced on a 0.663-in. pitch to give an

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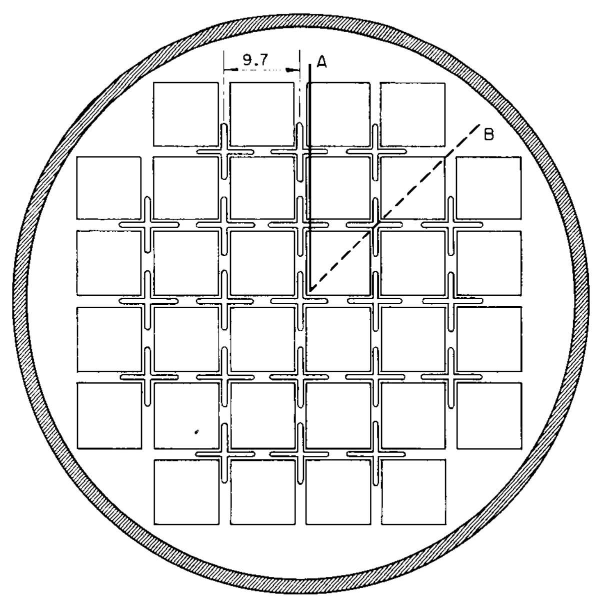

Fig. 6.1. Core Configuration of N. S. Savannah Reactor.

over-all metal-to-water ratio of about 0.6. From this brief description of the core, it is evident that the many existing inhomogeneities present problems in core analyses.

A knowledge of the fine flux variations across fuel tubes, across fuel element cans, near control rods, and near control rod followers is essential to an understanding of this reactor. For example, the temperature variation of the fine flux across fuel element cans has provided the likely explanation of the positive temperature coefficient of reactivity of the reactor up to about 150°F. Above this temperature, the temperature coefficient of reactivity is negative. A comprehensive examination of the fine flux variations across fuel tubes has been performed by using a P-3 spherical harmonic code for the IBM 704.[1] The variation of the flux near a control blade is being investigated by using the SNG program for the IBM 704. Results from both codes have been compared with each other and with flux measurements taken in the critical experiment. The two codes were found to be consistent with each other and with the experimental data.

A comparison of the results from these two codes for the flux distribution in the pin cell[2] is presented in Table 6.1. The results from the SNG code were obtained for both the S-4 and the S-8 approximations. There is a difference of only 0.5% between the P-3 thermal utilization and the S-8 thermal utilization values.

In order to reduce the power peaking to a minimum, it is planned to operate the reactor with some rods completely inserted, possibly a single shim rod, and the remaining rods completely out. With twenty-one control rods, there can be many such arrangements, each with its own characteristic power distribution. A two-group, two-dimensional study of the power distribution in the core for various control rod configurations is being undertaken with the use of the PDQ code for the IBM 704 machine.

A comparison of the two-group, two-dimensional calculations with results from a Babcock & Wilcox critical experiment is presented in Figs. 6.2 and 6.3. In these calculations the core region was

[1] C. M. Copenhaver, *Thermal Flux Disadvantage Factors for Marine Ship Reactor Lattice Cells*, ORNL CF-58-4-47 (April 3, 1958).

[2] A pin cell is one fuel pin plus its associated water moderator.

Table 6.1. Comparison of Flux Distributions for the N. S. Savannah Reactor Pin Cell Calculated by Various Approximations

	Approximations		
	P-3	S-4	S-8
$\bar{\phi}_F/\bar{\phi}_{cell}$, ratio of average flux in fuel pellets to average flux in pin cell	0.816	0.758	0.757
$\bar{\phi}_{SS}/\bar{\phi}_{cell}$, ratio of average flux in stainless steel tubing to average flux in pin cell	0.969	0.924	0.923
$\bar{\phi}_W/\bar{\phi}_{cell}$, ratio of average flux in water moderator to average flux in pin cell	1.218	1.159	1.160

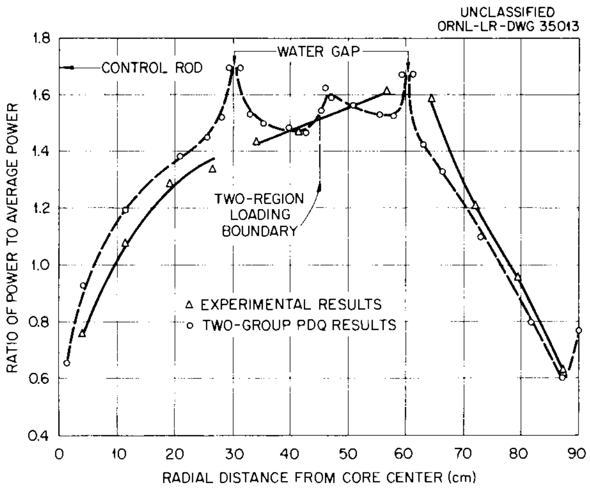

Fig. 6.2. Comparison of Two-Group, Two-Dimensional Calculations with Results from a Babcock & Wilcox Critical Experiment for Power Profile A.

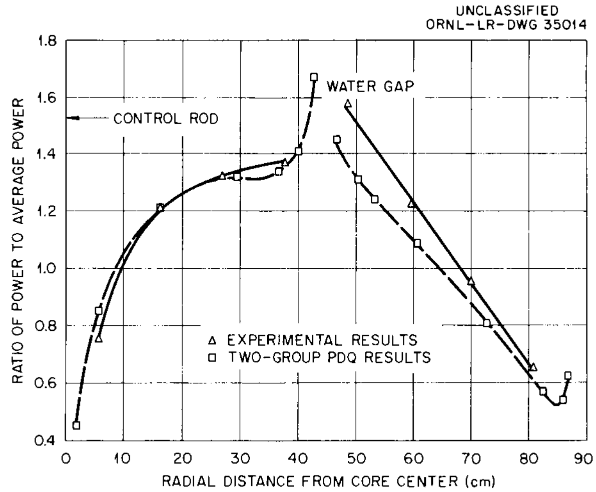

Fig. 6.3. Comparison of Two-Group, Two-Dimensional Calculations with Results from a Babcock & Wilcox Critical Experiment for Power Profile B.

homogeneized into a single region by using the disadvantage factors obtained from the fine flux calculations described above. As indicated on Figs. 6.2 and 6.3, the core contains a central control rod and two fuel regions. Power profiles A and B are as indicated on Fig. 6.1. The calculation is within 15% of the measured results and serves as a combined check on the method of obtaining the two-group constants and on the operation of the PDQ code.

Some results of the power distribution study for the design core at the operating temperature (500°F) are presented in Figs. 6.4 through 6.8. Each figure presents power profiles A and B (Fig. 6.1) for the indicated control rod configuration. The two main features of these results are (1) the relatively localized effect that the control rods have on the power distribution and (2) the severe power peaking that occurs where control rods are out of the core.

A comparison of Figs. 6.4 and 6.5 shows that the insertion of the control rod affects the power distribution very little beyond a radius of about 26 cm. This effect is again demonstrated in Figs. 6.6 and 6.7.

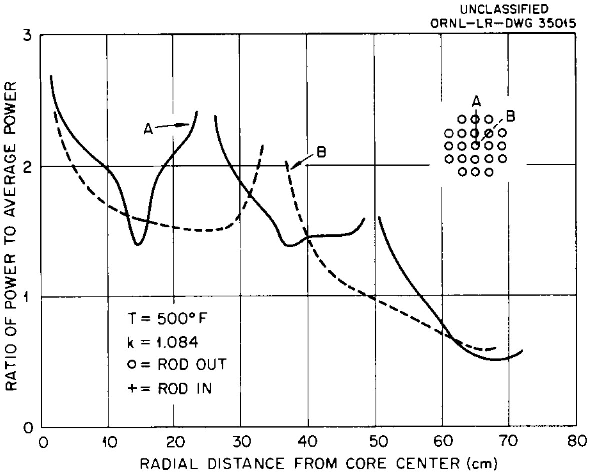

Fig. 6.4. Power Profile for Two-Region Loading, Case No. 4.

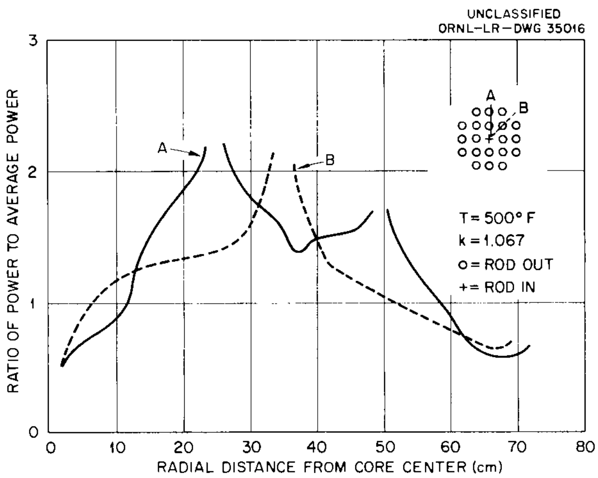

Fig. 6.5. Power Profile for Two-Region Loading, Case No. 6.

Again comparing Figs. 6.4 and 6.5, it may be seen that a very desirable decrease in the peak-to-average power ratio may be obtained by insertion of the central control rod in the configurations of Fig. 6.4. This feature may also be demonstrated by comparing Figs. 6.6 and 6.7.

Figures 6.7 and 6.8 give power profiles for two possible arrangements of five rods. On the basis of minimizing the peak-to-average power, it may be seen that the control rod configuration in Fig. 6.8 is preferable to the one presented in Fig. 6.7. It is hoped that this analysis will culminate in a procedure for operation of the control rods in the N. S. Savannah reactor.

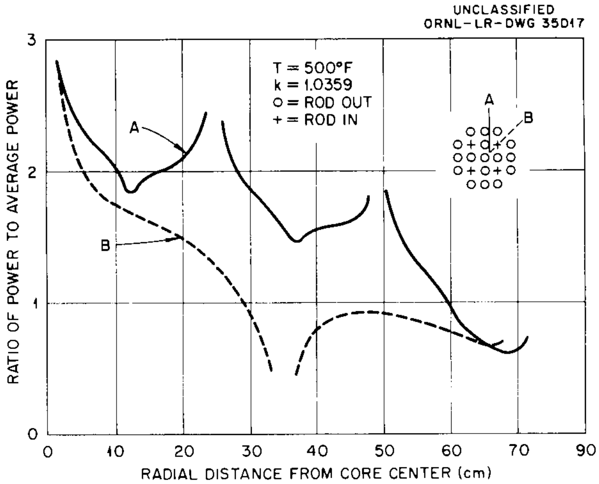

Fig. 6.6. Power Profile for Two-Region Loading, Case No. 10.

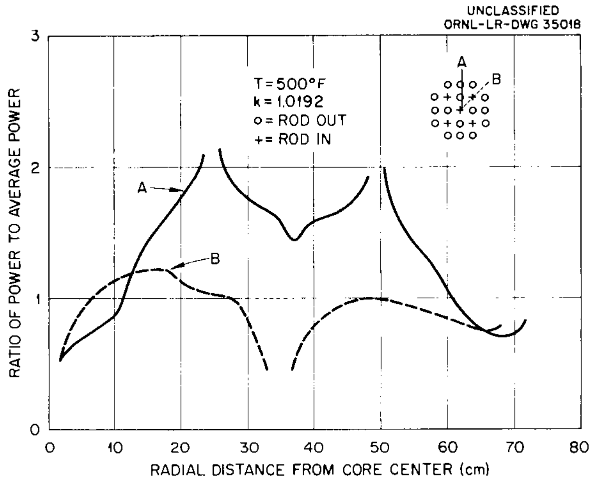

Fig. 6.7. Power Profile for Two-Region Loading, Case No. 8.

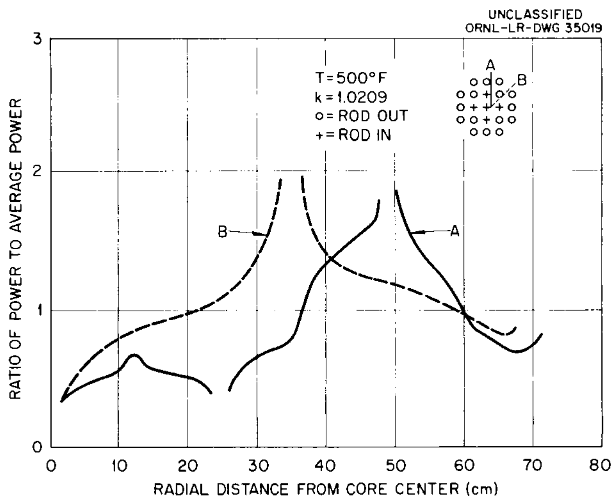

Fig. 6.8. Power Profile for Two-Region Loading, Case No. 9.

INTERNAL DISTRIBUTION

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3. D. S. Billington
4. R. E. Blanco
5. E. P. Blizard
6. A. L. Boch
7. C. J. Borkowski
8. G. E. Boyd
9. E. J. Breeding
10. R. B. Briggs
11. F. R. Bruce
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52. K. Z. Morgan
53. J. P. Murray (Y-12)
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64. A. W. Savolainen
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69. A. H. Snell
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