IDO-19012
CEND-82

SL-1 ANNUAL OPERATING REPORT

FEBRUARY 1959 - FEBRUARY 1960

CONTRACT NUMBER AT (10-1)-967
U S ATOMIC ENERGY COMMISSION

NUCLEAR DIVISION
COMBUSTION ENGINEERING, INC.
IDAHO FALLS, IDAHO

DISCLAIMER

IDO-19012
CEND-82

SL-1 ANNUAL OPERATING REPORT

February, 1959 - February, 1960

May 1, 1960

Prepared By:

R. T. Canfield
J. S. Crudele
E. J. Vallario
R. G. Young
W. P. Rausch

U. S. ATOMIC ENERGY COMMISSION
Contract At(10-1)-967

Unclassified
Classification

Evan W Woolford 5/1/60
Authorized Classifier

Approved

W B Allred
W. B. Allred
ABWR Project Manager

COMBUSTION ENGINEERING, INC.

Nuclear Division

Idaho Falls Idaho

REPORT DISTRIBUTION

U. S. Atomic Energy Commission (5)
Washington 25, D. C.
ATTN: Chief, Water Systems Projects Branch
Army Reactors DRD
Mail Station F-311

U. S. Atomic Energy Commission
Washington 25, D. C.
ATTN: Chief, Technical Evaluation Branch
Army Reactors, DRD
Mail Station F-311

U. S. Atomic Energy Commission
Washington 25, D. C.
ATTN: Air Force Liaison Officer
Army Reactors, DRD
Mail Station F-311

U. S. Atomic Energy Commission
Washington 25, D. C.
ATTN: Chief, Water Reactors Branch
Civilian Reactors, DRD
Mail Station F-311

Nuclear Power Field Office
Engineer Research and Development Laboratories
Fort Belvoir, Virginia
ATTN: Chief, Projects Branch

Nuclear Power Field Office
Engineer Research and Development Laboratories
Fort Belvoir, Virginia
ATTN: Chief, Applications Engineering Branch

Nuclear Power Field Office
Engineer Research and Development Laboratories
Fort Belvoir, Virginia
ATTN: Chief, Operations Branch

Nuclear Power Field Office (2)
Engineer Research and Development Laboratories
Fort Belvoir, Virginia
ATTN: Chief, Training Branch

The Martin Company
Baltimore 3, Maryland
ATTN: Resident Engineer, PM-1
Nuclear Division, Mail No. W-754

The Martin Company (2)
Baltimore 3, Maryland
ATTN: Project Manager, PM-1
Nuclear Division, Mail No. W-719

Navy Department (3)
Bureau of Yards and Docks
Washington 25, D. C.
ATTN: Code D-440

Alco Products, Inc.
One Nott Street
Schenectady 5, New York
ATTN: Manager, Engineering Atomic Energy Dept.

Office, Chief of Engineers
Nuclear Power Division
Building T-7
Washington 25, D. C.
ATTN: Chief, Projects Branch

Office, Chief of Engineers
Nuclear Power Division
Building T-7
Washington 25, D. C.
ATTN: Health Physics and Safety Officer

U. S. Atomic Energy Commission (2)
Chicago Operations Office
9800 South Cass Avenue
Argonne, Illinois
ATTN: Director, Programs Division

U. S. Atomic Energy Commission (2)
Idaho Operations Office
P. O. Box 2108
Idaho Falls, Idaho
ATTN: Director, Military Reactors Division

U. S. Atomic Energy Commission (25) & one reproducible
Reference Branch
Technical Information Service Extension
P. O. Box 62
Oak Ridge, Tennessee

Office of Technical Services (10)
Department of Commerce
Washington 25, D. C.

Commanding General
U. S. Army Engineer Research and Development Laboratories
Fort Belvoir, Virginia
ATTN: Contracting Officer

NRTS Technical Library (20)
Phillips Petroleum Company
Box 2067
Idaho Falls, Idaho
ATTN: Report Section

U. S. Atomic Energy Commission (2)
Idaho Operations Office
P. O. Box 2108
Idaho Falls, Idaho
ATTN: Director, Division of Health and Safety

Reactor Projects Division
Oak Ridge National Laboratories
P. O. Box Y
Oak Ridge, Tennessee
ATTN: A. L. Boch, Associate Director

U. S. Army SIPRE
1215 Washington Avenue
Wilmette, Illinois
ATTN: Frederick J. Sanger

U. S. Atomic Energy Commission
Washington 25, D.C.
ATTN: Chief, Evaluation and Planning Branch
Civilian Reactors, DRD
Mail Station F-311

Nuclear Power Field Office
P.O. Box 2558
Idaho Falls, Idaho
ATTN: Captain S. W. Nunnally (3)
Chief, SL-1 Cadre

Combustion Engineering, Inc., Idaho

W. B. Allred (3)
P. R. Duckworth (4)
E. J. Vallario
E. W. Woollacott
ABWR Library (5)

Combustion Engineering, Inc., Windsor

W. H. Zinn (2)
J. B. Anderson
L. E. Anderson (4)
R. H. Gale
H. Geller
B. Gitlow (3)
R. L. Hellens
A. F. Miller, Jr.
J. J. Satchwell
V. D. Schwartz (1)
S. Visner
R. L. Hoover
R. G. Knight
CEND Library (2)

ABSTRACT

This report covers the period from February, 1959, when Combustion Engineering, Inc. assumed operative responsibility of the SL-1 Reactor Plant, to February, 1960. The operations of the year are summarized; the reactor, instrumentation, mechanical, electrical, and facility systems are evaluated; health and safety, and the operational costing program are discussed.

THIS PAGE

WAS INTENTIONALLY

LEFT BLANK

CONTENTS

CONTENTS (Continued)

ILLUSTRATIONS

ILLUSTRATIONS

SUMMARY

The Stationary Low Power Reactor No. 1 is a boiling water reactor demonstration plant for remote military bases. The plant has been in operation at the National Reactor Testing Station for performance evaluation and training of military personnel. The evaluation effort has been conducted to provide performance data for Portable Low Power Reactor plant design.

This report is an assessment of Stationary Low Power Reactor No. 1 plant performance during the year ending February, 1960. The assessment covers plant performance, component reliability, maintenance requirements, health and safety experience, and training of military personnel.

The aggregate of Stationary Low Power Reactor No. 1 operations and maintenance experience leads to the conclusion that boiling water reactors can be operated for continuous and steady generation of power and heat. Reliability of conventional equipment is considered acceptable with few exceptions. In these cases modifications have been made or design alternates are available. No overexposures to radiation and no lost time industrial accidents have been experienced by plant personnel. Routine operation of the Stationary Low Power Reactor No. 1 plant by two man military crews has proved satisfactory.

The Stationary Low Power Reactor No. 1 plant had operated a total of 4,401 hours as of February 2, 1960. Based upon scheduled operating time, the plant was available 83 per cent. During this period, the aluminum alloy core was burned to 363 megawatt days, which is 16 per cent of design core life. Periodic examinations of the aluminum core have revealed satisfactory corrosion resistance as well as no apparent radiation effects. A crud film has been observed on all aluminum core surfaces. Chalk River and Hanford experiments under similar conditions lead to the prediction that a substantial thermal barrier may exist between the metal and water. Attempts to measure this film to date at Stationary Low Power Reactor No. 1 have been unsuccessful. An increase in metal temperature resulting from a film deposit will place the aluminum alloy elements in a temperature range where swelling may be predicted before end of designed core life. Another major irregularity of the fuel elements observed to date has been a significant warping of the boron strips which are tack welded to one side plate on each element. The warpage is believed to be caused by thermal stresses in the clad material. Core operation has not been affected to date and fuel element removal can still be performed.

From initial operation of the reactor to the present, fission product activity was observed in the main steam system. Surface and/or clad contamination have subsequently been ruled out since the activity level is greater than can be attributed to fabrication techniques and no evidence exists of decreasing fission activity from burnout. However, the fission product level has not increased significantly during the period. Based upon a ratio determination of Xe^{138}/Xe^{133} a delay

mechanism exists from time of fission to time of measurement. The delay may be attributable to a small cladding defect. It is significant that with fission product activity in the boiling system, radiation levels have been within tolerances for personnel exposure.

Performance of Stationary Low Power Reactor No. 1 control rod drive mechanisms has been generally satisfactory. The simple rack and pinion mechanism satisfied the operational requirements for the boiling system. Periodic sticking of the rods and other malfunctions can be corrected by minor design improvements.

The use of a flexitallic gasket seal for the reactor vessel head has proved satisfactory.

The transfer of fuel elements has been demonstrated. Removal of an active fuel element from the core to a fuel storage vault was performed with a lead cask and integral winching arrangement. The method for transfer was considered satisfactory, however design deficiencies were uncovered in the transfer cask. These are being corrected.

The Stationary Low Power Reactor No. 1 power plant systems have been operated at normal and abnormal conditions. Because of the test and training requirements, the plant has undergone startup and shutdown conditions during one year which would normally cover many years of normal operation. Even with this type operation, plant components have performed reliably, experiencing only 22 malfunctions which caused unscheduled plant shutdown. Maintenance time required for repairs was found to be reasonable and within the downtime criteria set for remote field installations.

Plant performance has been satisfactory. Steam quality during normal power operation was better than 99.5 per cent and a decontamination factor of 10,000 between reactor water and steam was maintained. The radiactivity level in the main steam piping, turbine, and condenser has been sufficiently low to permit full personnel access to this equipment with no overexposures.

The main air-cooled steam condenser has performed as designed; however, the unit was undersized by approximately 15 per cent. Design data have been obtained for proper sizing of Portable Low Power Reactor condensers, and a Portable Low Power Reactor condenser prototype will be tested at the Stationary Low Power Reactor No. 1 plant during 1960.

The plant evaluation has uncovered several deficiencies which prevent continuous power operation for periods longer than two months. The turbine governor requires replacement of oil every 500 hours. Shutting down the turbine is needed to perform this task. Installation of a continuous oil purification system will remove this problem. Stationary Low Power Reactor No. 1 experience has shown a 1,500 hour maintenance cycle of the canned rotor condensate pump, purification pump, low pressure drain collection pump, and high pressure drain collection pump to be required.

To assure continuous plant operation, paralleling these pumps is necessary. This design philosophy has been used for the feedwater system pumps and can be extended to other system pumps. These changes would permit continuous scheduled operations of Stationary Low Power Reactor No. 1.

The excessive thermal cycling resulting from startup and shutdown of Stationary Low Power Reactor No. 1 has resulted in excessive occurrence of leakage of small valves. Experience has shown asbestos gaskets and standard bolt material unsatisfactory in this service. The use of high strength bolts has substantially improved valve experience. Although steam leaks from valves have resulted in several shutdowns of Stationary Low Power Reactor No. 1, resulting activity levels have been low and no personnel overexposures have occurred.

The Stationary Low Power Reactor No. 1 plant uses an automatic control system in which the main steam pressure is controlled by automatic adjustment of control rods. The system has functioned very successfully at all power levels in the plant and during transient experiments performed.

The Stationary Low Power Reactor No. 1 facility uses native rock for biological shielding. The shield design has proved satisfactory as designed except for insufficient shielding above the purification vault, which was corrected.

Experience on the collection and discharge of contaminated waste indicates that a substantial quantity of low level liquid and solid waste must be dealt with. Low level radioactive gas was vented continuously to the atmosphere and was readily dispersed by air currents. The quantities of waste appear to be easily manageable and have at no period precluded safe operation of the plant.

THIS PAGE

WAS INTENTIONALLY

LEFT BLANK

I OPERATIONS

This section summarizes SL-1 operational experience for the year ending February, 1960. Since it is intended as a brief review, the majority of the material has been tabulated. A short history, summary of SL-1 design characteristics, a chronology of SL-1 operations, and operational data summary, training and malfunction summaries are included.

The Stationary Low Power Reactor No. 1 (SL-1) is a small, natural circulation, direct cycle boiling water reactor designed to generate electrical power and space heat for remote arctic installations.

Design and construction of the facility was performed at the National Reactor Testing Station during 1957 and 1958. Criticality was initially achieved on August 11, 1958, and the plant was first operated at power on October 23, 1958. Prior to February 5, 1959, the plant was test operated by Argonne National Laboratory. Since then, the reactor plant has been operated by Combustion Engineering, Inc. as a test, demonstration, and training facility.

A. SL-1 DESIGN CHARACTERISTICS

A summary of the SL-1 design characteristics is as follows:

Plant

Reactor heat output	3 MW (t)
Steam production	9020 lb/hr
Operating pressure	300 psig
Operating temperature	429 ^{o}F
Turbine generator output	300 KW (e)
Steam quality	Saturated
Space heating load	400 KW (t)
Ambient temperature	-60 to +60 ^{o}F
Air cooled condenser capacity	7.5 x 10^{6}BTU/hr

Reactor Core

Cladding alloy for plates (Al - 1% Ni) ALCOA	X-8001
Length of active core	25.8 in.
Equivalent diameter of active core	31.4 in
Number of fuel elements	40
Total thickness of fuel plates (0.050 in. meat, 0.035 in. clad)	0.120 in.
No. of plates per element	9
Average water channel gap	0.310 in.
Fuel in 40 elements, U^{235}	14 Kg
Weight of U^{235} per element (approx)	350 gm
Weight of B^{10} in B-Al strips Attached to 40 fuel elements	22.6 gm

Control Rods

Number of crosses	5
Size of crosses	14-1/4 in.
Length of cadmium section	32 in.
Thickness - cadmium	0.060 in.
Thickness - AL-1% Ni alloy clad	0.080 in.
Scram time	2 seconds
Withdrawal rate (approx. max.)	0.01% k/sec.

Nuclear Data

Core lifetime	3 years
Plant load factor	0.7
Average thermal flux	7.5×10^{12} n/cm^2/sec
Average thermal lifetime	4 to 8×10^{-5} over range of core lifetime
Reactivity changes, %k	
Temperature effect	1.5 to 2.0
Steam voids	1.3 to 2.0
Xe + Sm	3
Xe override	1 to 1.5

Heat Transfer

Average power density in coolant	17.5 Kw/l
Standard flow at 3 MW	9020 lb/hr
Average steam voids in heated channel	9%
Average steam voids in moderator	7%
Feedwater inlet temperature	175 °F
Average boiling length of core	20 in.
Total heat transfer area	475 ft^2
Average heat flux	21,500 BTU/hr (sq. ft.)

B. SL-1 OPERATIONAL CHRONOLOGY

Date	Operation
2/5/59	Combustion Engineering, Inc. assumes control of SL-1 Plant
2/5 - 3/29	Revised operating procedures were prepared
3/30 - 4/1	Cold criticals performed
4/2 - 4/3	Operation demonstration for Combustion Engineering, Inc. Nuclear Safety Committee
4/4 - 4/22	Reactor vessel head gasket developed leak and was replaced as reported in IDO-19001
4/23 - 4/26	Cold hydrostatic testing of head gasket replacement
4/27 - 5/1	Full power test of head gasket replacement
5/2 - 5/13	5-day week, 3-shift operation for training and qualification of Cadre personnel
5/14 - 5/18	Plant secured for repair of plugged gland air ejector
5/19 - 6/2	5-day week, 3-shift operation for training and qualification of Cadre personnel
6/3 - 6/4	Plant secured for repair of main condenser fan motor
6/5 - 7/9	1,000 hour test for evaluation of plant systems
7/20 - 7/27	Operation for Physics testing
7/28 - 8/28	Plant secured for maintenance and modifications
8/29 - 9/20	Operation for test of 2-man crews (3-shift, 5 day week)
9/21 - 9/23	Plant secured for installation of instrumented fuel assembly
9/24 - 10/4	Plant operated for instrumented fuel element test
10/5 - 10/12	Plant secured for removal of instrumented fuel element
10/13 - 11/25	Operation for training of personnel
11/26 - 12/7	Plant secured for maintenance and modification downtime
12/8 - 12/23	Plant operation for test program
12/24/59 - 2/1/60	Annual maintenance period

C. OPERATIONAL DATA SUMMARY

A summary of operational data accumulated to February 2, 1960 is tabulated below. The average values in this table are all above the design plant load factor of 0.7.

TABLE I

	Cumulative Plant Totals
Hours plant operated (1)	4,401
Hours at power (2)	3,971
Reactor-MWD (t) (3)	363
Turbine generator - hours (4)	3,510
Turbine generator - KW (e) (5)	806,100
Simulated heat load-KWH (t) (6)	1,415,100
Fuel burnup - grams U^{235}	458

Notes:

1. The total time the plant was in operation, checkout and startup time, and shutdown time.

2. Total time the reactor was critical.

3. This number is obtained by totaling hourly calculations of reactor power.

4. Taken directly from the turbine hour meter.

5. This number was obtained directly from a KWH meter.

6. The SL-1 reactor is designed to produce 400 KW of thermal energy for space heating. It is calculated every hour from values of coolant flow rates and temperatures.

D. SL-1 TRAINING SUMMARY

The SL-1 plant is used as a training facility by military personnel. Trainees are assigned to each operating crew for on-the-job experience in reactor operation prior to qualifying as Chief Operators or Operators. Table II summarizes the reactor operator training statistics to date.

Training in health physics, industrial safety, first aid, fire protection, and civil defense has also been completed by all operating personnel.

These training activities are a part of the Army Reactor Training Program.

Table II

SL-1 TRAINING SUMMARY FOR 1959

TOTAL IN SL-1 OPERATOR TRAINING

Army	25
Navy	5
Air Force	7
Maritime	18
Total	55

TRAINEES COMPLETING QUALIFYING TEST

Chief Reactor Operators	13
Reactor Operators	16
In Training	2
Other	24

SPECIAL RATINGS OF TRAINEES

Electrical	7
Instrument	9
Mechanical	12
Health Physics	3
Other	24

E. SUMMARY OF MALFUNCTIONS AT SL-1

Tabulated below are the malfunctions that have occurred at SL-1 during this report period. Of the 573 hours total plant downtime, only 143 hours were required for maintenance. The remainder was scheduled downtime, i.e., weekends, and second and third shifts.

TABLE III

MALFUNCTION SUMMARY

Number of Malfunctions	Category	Hours
	Electrical Equipment	63
1	Fan Motor	61
2	Station Auxiliary Breaker	2
	Control System	30
1	Tube in HV Supply	10
1	Tube in High Level Indicator	3
2	Insulation Breakdown HV Supply from Temperature	17
	Mechanical Equipment	473
2	Turbine Governor	7
3	Valves	33
1	Reactor Vessel Gasket	356
2	Control Rod Drives	6
2	Ejectors	71
	Operator Error	7
1	Turbine Throttle Valve not Opened Fully	3
1	Wrong Fuse Pulled	1
1	Unintentional Shorting of Circuit	1
1	Loss of Coolant	1
1	Incorrect Adjustment of TG Governor Linkage	1
22	TOTALS	573

II REACTOR SYSTEM EVALUATION

During the past year of operation of the SL-1 reactor, particular attention was placed on the performance of the aluminum-nickel alloy core. A limited program to investigate core parameters was undertaken and core performance data were recorded for lifetime extrapolation. The discovery of unfavorable aluminum heat transfer characteristics increased interest in the thermal performance of the SL-1 fuel elements. A thermocouple instrumented fuel element test was attempted to establish operating characteristics, but the results were indeterminate. Minute quantities of fission products, detected in reactor water and steam samples, indicated the possibility of a fuel element defect.

The head gasket in the reactor vessel had to be replaced and ensuing difficulties encountered with the vessel studs resulted in almost one month of downtime. The rod drive mechanisms stuck occasionally in free fall operation; the rod seals experienced an excessive wear rate. The difficulties are attributed primarily to high operating temperatures. The shield cooling has been accomplished satisfactorily by natural convection, since the shield cooling requirements were less than predicted.

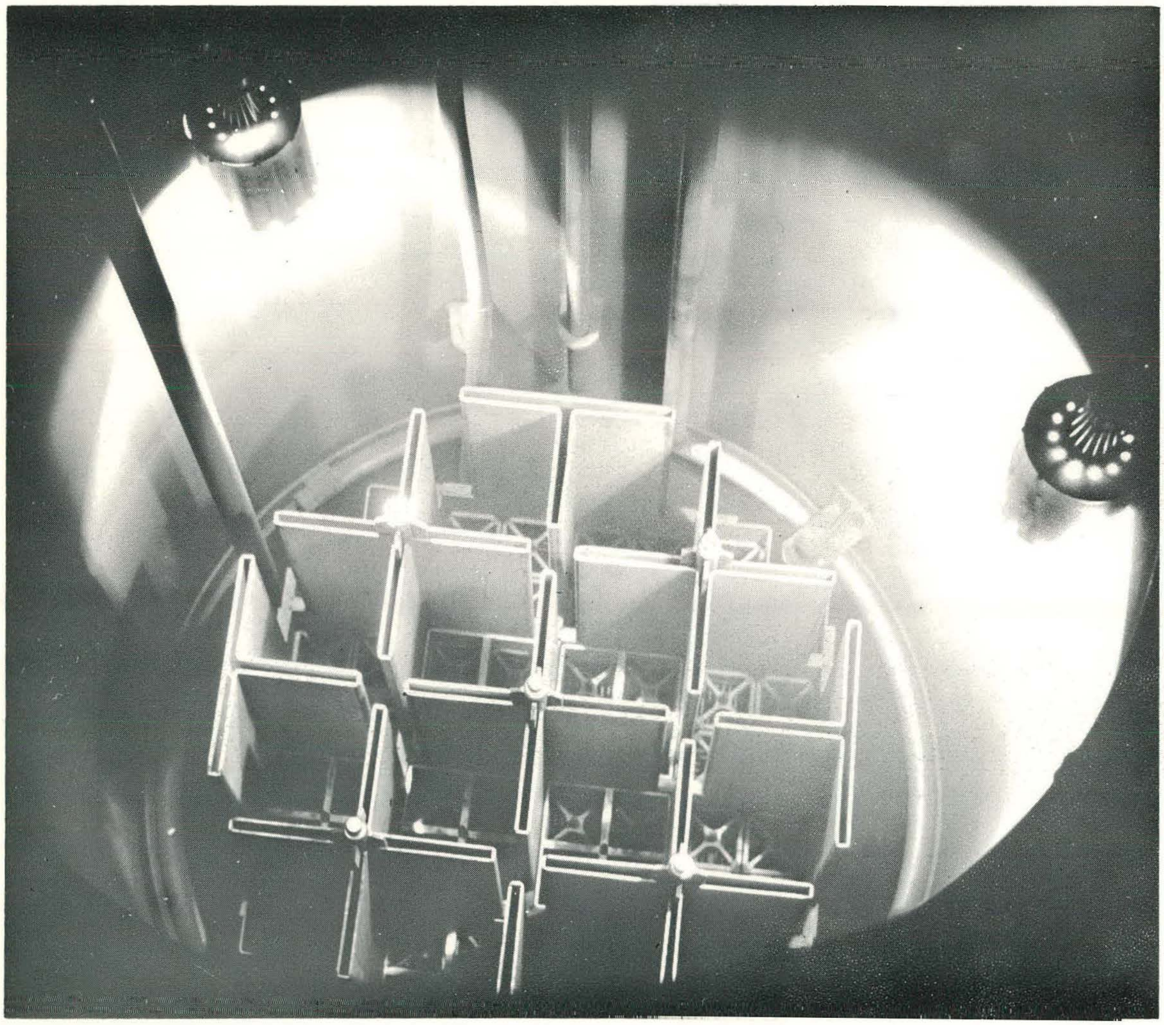
The overall reactor system performance was satisfactory. Recommendations and plans for improvements are incorporated in the individual system evaluations.

A. REACTOR CORE

1. Description

The SL-1 core was fabricated from an aluminum-nickel alloy (Alcoa X-8001). A general view of the core is shown in Figure 1. Specific core data are tabulated in Section I of this report. The core structure is divided into 16 boxes. The four corner boxes were designed to hold three fuel elements each; the remaining 12 boxes to contain four fuel elements each. The maximum core capacity is 60 elements. Presently, the SL-1 reference core loading consists of 40 elements arranged to approximate a right circular cylinder. The outside elements are supported in the boxes by dummy elements and a SbBe source is located outside the core radius adjacent to control rod No. 5.

The sides of the core boxes serve as shrouds and define the control rod channels, including five full-cross channels and four T-channels. The T-control rods are not used in the 40 element reference loading.



SL-1 ALUMINUM - NICKEL ALLOY REFERENCE CORE

FIGURE 1

The fuel elements consist of nine 0.120 inch thick flanged fuel plates assembled to the side plates by spot welding. A fuel plate consists of an 0.50 inch thick center portion (meat) of aluminum-nickel-uranium alloy, clad with 0.035 inch thick aluminum-nickel. The "meat" is 3.5 inches wide by 25.8 inches long. The finished clad plate is 3.710 inches wide without the flanges, and 27.8 inches long.

To prolong core lifetime in the 40 element reference loading one aluminum-nickel-boron strip 25.8 inches long, 3.875 inches wide, and 0.025 inches thick containing approximately 0.5 gm B^{10} is spot welded to the lower half of the fuel element on the side plate, opposite the full length poison strip.

The fuel element spacing is maintained by stainless steel springs which are fastened on four sides at the top of the element. Fuel element handling is accomplished by a gripper mechanism which attaches onto stainless steel knobs threaded and pinned into the upper end of the fuel elements.

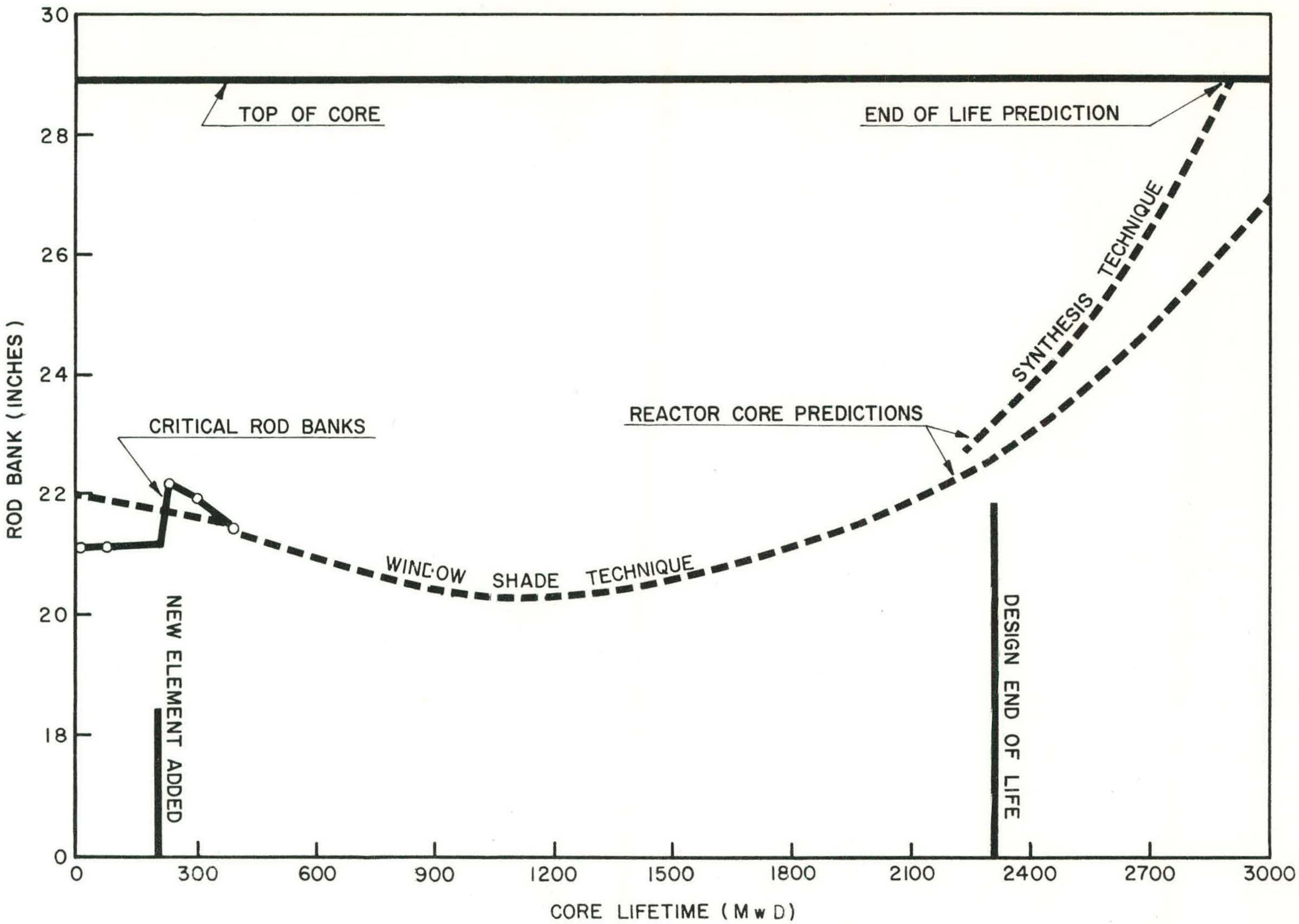
Hold-down devices were designed for the original core to keep the fuel elements from creeping upward during operation. However, a design analysis of the hydraulic forces at 3 MW(t) power operation revealed that the hold-down devices are not required, as indicated in IDO-19003.

2. Discussion

During 1959 the reactor was operated the equivalent of 148.9 days at an average of 2.16 MW(t). The accumulated average core burnup totals 3.3 per cent corresponding to a 16 per cent reduction in core life. The reactor was operated with the control rods in an evenly adjusted bank to keep the burnup distribution constant. In October, 1959, after 210 MWD of reactor operation, a center fuel element was removed for inspection and measurement in a hot laboratory facility and a new fuel element was installed. This loading change distorted the core flux patter, slightly altering the burnup distribution in the core. Figure 2 shows the recorded rod bank positions for equilibrium xenon conditions at 2.57 MW(t) with respect to reactor operation expressed in megawatt-days. Reactor core lifetime predictions are plotted on the same graph for comparison.

a. Fuel Element Temperature

Although the core performance has been satisfactory to date, there is concern over the heat transfer characteristics of the aluminum fuel elements. An analysis of experimental tests performed at Chalk River indicates that the X-8001 Al-Ni alloy film coefficient increases after a relatively short immersion in water. Such a phenomenon occurring in the SL-1 fuel elements would cause the fuel meat temperature to rise above design limits, possibly resulting in thermal damage to the fuel elements. In view of this consideration the reactor operation has been watched very closely for indications of fuel element rupture. A thermocouple instrumented fuel element test was attempted to verify calculated fuel center line temperatures under operating conditions. The test failed because the thermocouples registered reactor water temperature. A future test will be conducted when a method has been developed. As an



CRITICAL ROD BANK POSITION WITH EQUILIBRIUM XENON CONCENTRATION AT 2.56 Mw(t)

FIGURE 2

intermediate step, a controlled laboratory experiment has been proposed to establish the degree of film formation on aluminum surfaces under SL-1 operating conditions. In addition visual inspections of a representative group of fuel elements were made in the reactor vessel using four power binoculars. There were no detectable irregularities discovered on the fuel plates other than a slight warpage of the aluminum-boron side as shown in Figure 3. Estimated separation between the fuel element and the boron was approximately 80 mils. The detection of minute quantities of fission products in the reactor water and steam indicate that a cladding defect may exist in the present core. Constant surveillance over the fission product activity has shown that the condition has not intensified.

b. Control Rod Worth

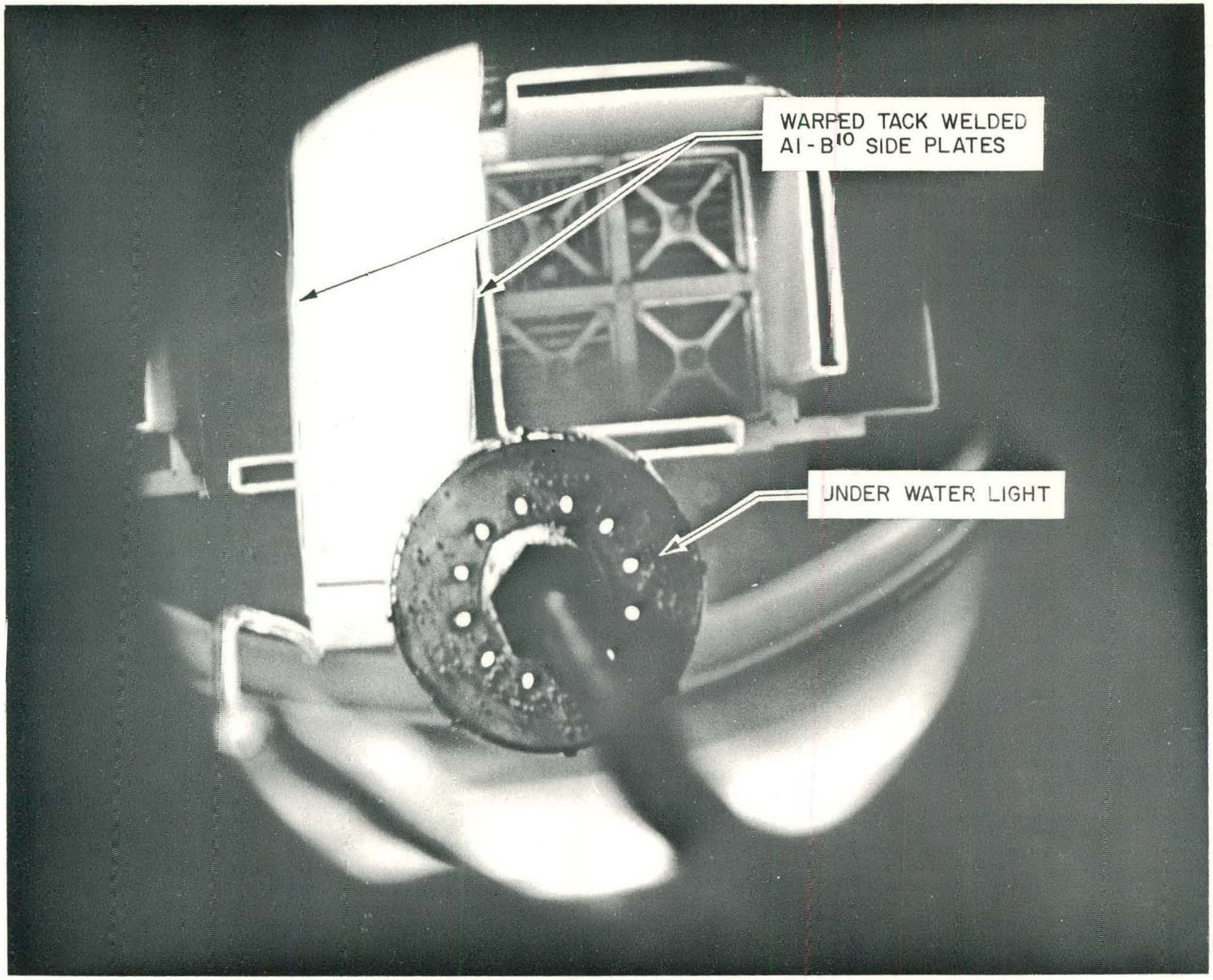
The cold critical and initial core evaluation tests were performed by Argonne National Laboratory. The Combustion Engineering, Inc. test program during the past year was limited to a few tests to verify core operating characteristics. Several differential control rod worth evaluations were made, but the data are presently insufficient for determining reactivity values for core parameters. Thus, the experimental data obtained in the core performance tests can be evaluated only in terms of relative control rod positions. It should be noted that control rod positions are relative to a zero position 3-1/8 inches below the nominal lower dimension of the fuel "meat". The reactivity effect of reactor water temperature was determined by regulating the control rod bank to maintain criticality as the reactor cooled from operating temperature to shutdown conditions. The critical rod bank position was recorded at less than 1,000 watts reactor power for intermediate water temperatures from 420 ^{o}F to 100 ^{o}F. The total rod bank change was 4.5 inches for a water temperature change of 320 ^{o}F.

The effect of xenon poisoning was measured after 320 MWD of reactor operation in terms of rod bank positions. The xenon decay showed that the xenon peaks at approximately five hours in decay. The maximum xenon effect is negligible as may be expected for a reactor operating with an average thermal flux below 10^{13} $n/(cm^2)(sec)$.

Figure 4 shows the critical rod bank positions with no xenon present and for equilibrium xenon conditions at different power levels measured after 320 MWD of operation. The curves show the effect of steam void and equilibrium xenon concentrations at various power levels from 0.4 MW(t) to 3.0 MW(t).

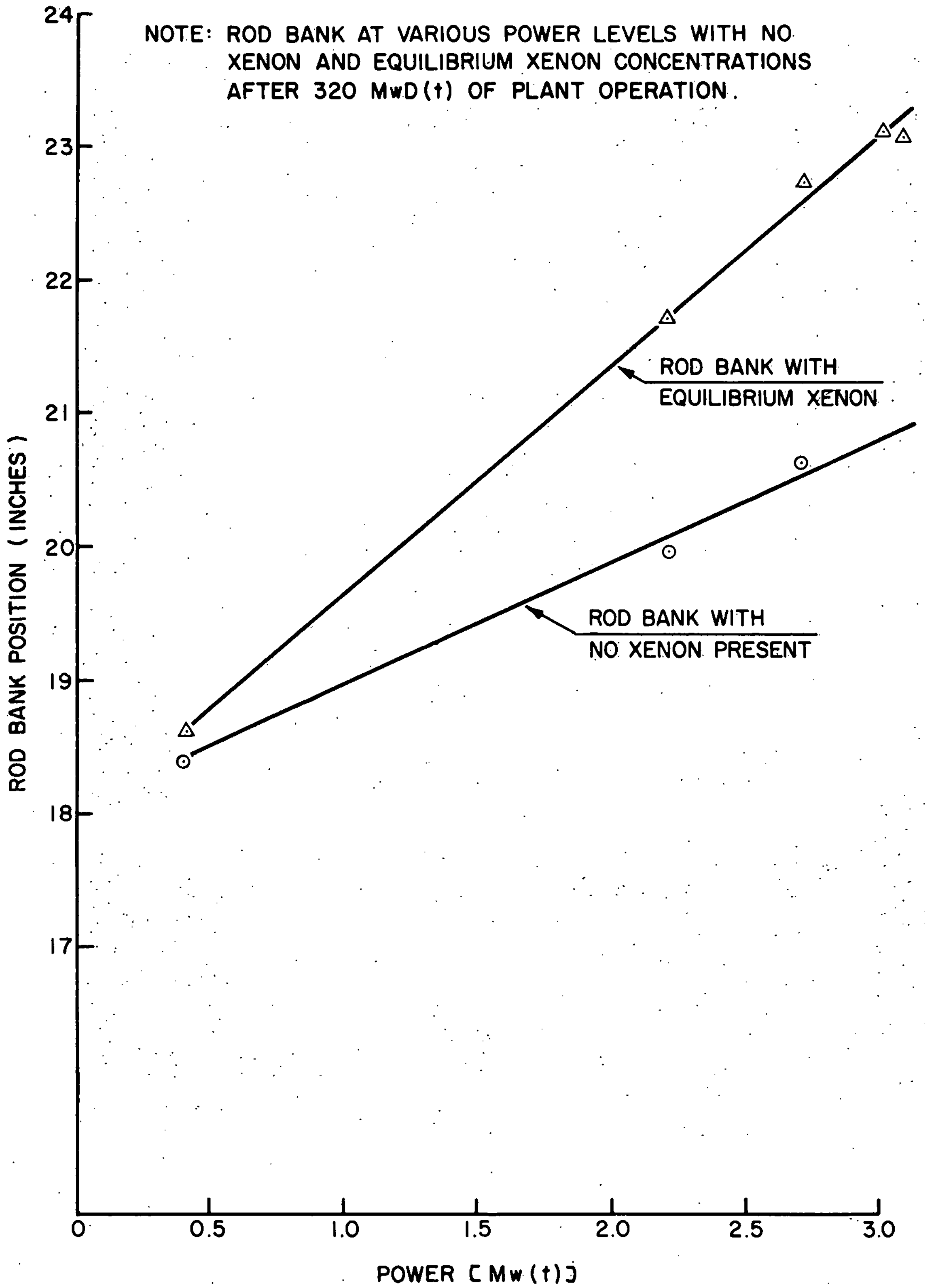
Control rod comparisons were obtained at approximately 1.4 MW(t) by comparing the side rods with the center rod. The results are plotted in Figure 5 showing the relative worth of the side rods with respect to the center rod. The slight asymetrical relationship in the lower core region is apparantly due to flux distortion caused by the moderating beryllium block in the SbBe source which is located adjacent to rod No. 5.

Figure 6 shows the reactivity worth of rod No. 7 as a function of rod positions. The rod was calibrated after 200 MWD of plant operation with the reactor at 120 ^{o}F. The curve indicates that the average independent reactivity worth for a side rod is \$2.60. A comparison of differential rod worths at 18 inches indicates that the center rod is twice as effective as a side rod under cold conditions.



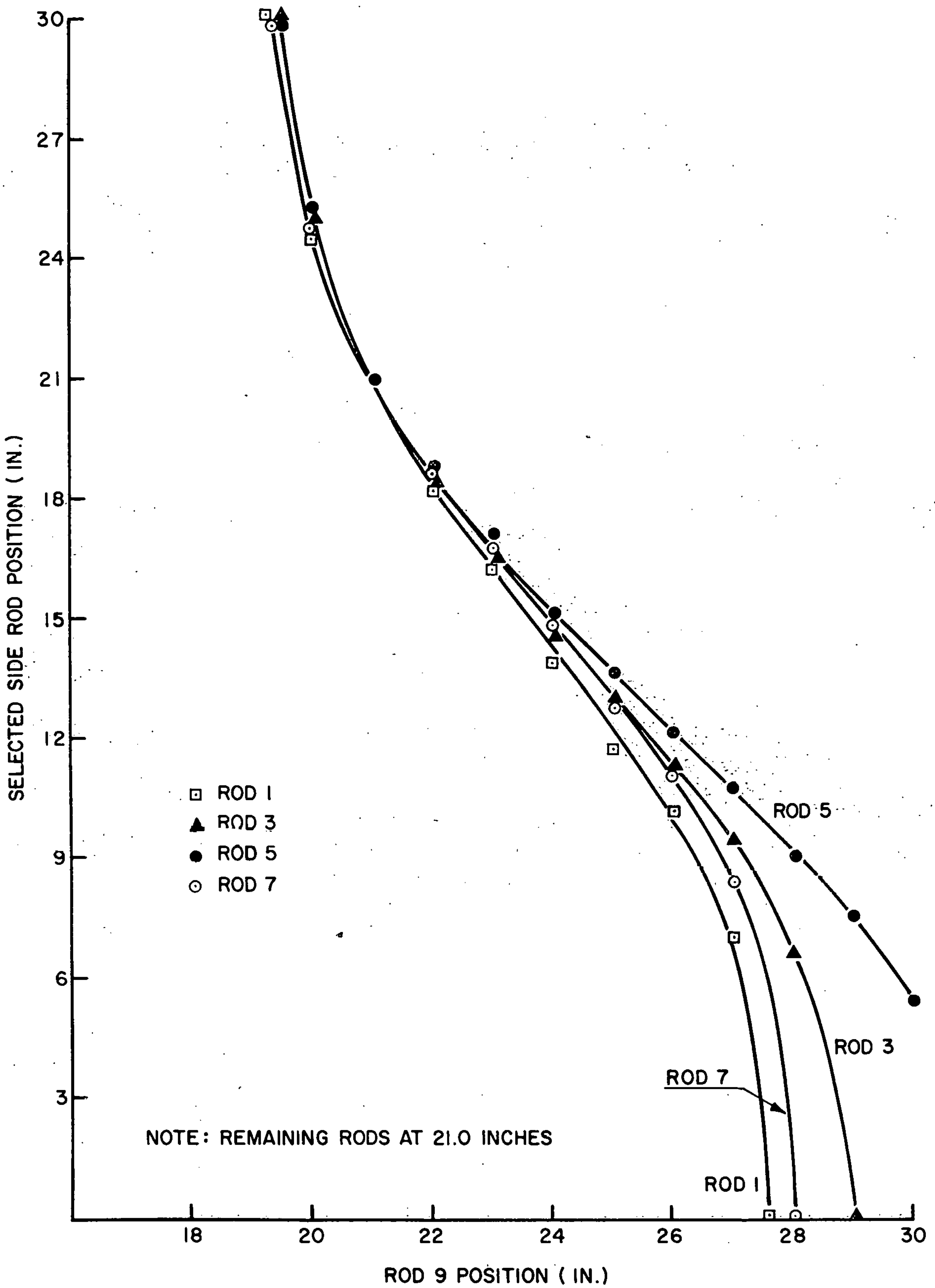
FUEL ELEMENT INSPECTION

FIGURE 3



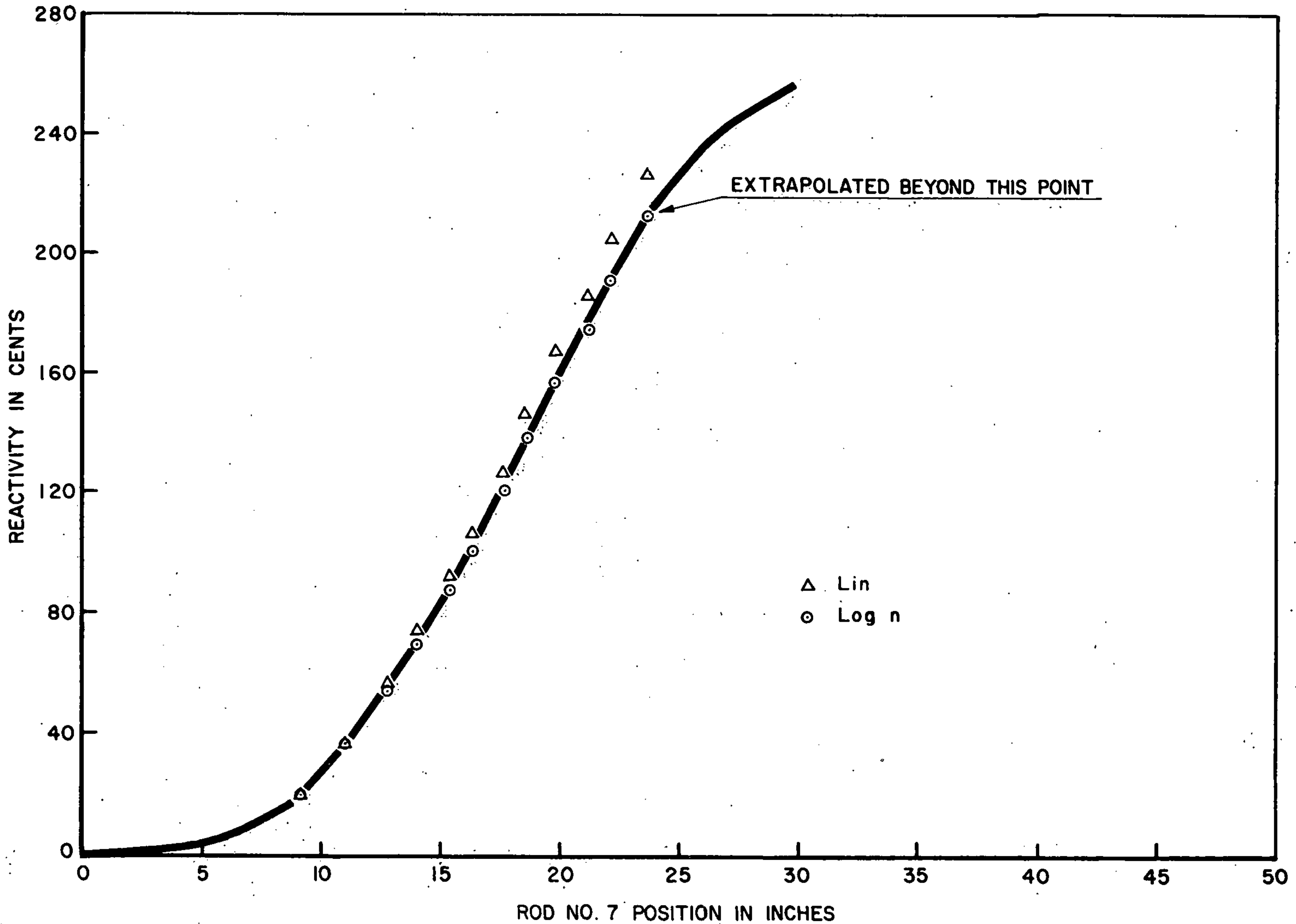
XENON AND VOID EFFECTS

FIGURE 4



CORE SYMMETRY

FIGURE 5



REACTIVITY WORTH OF ROD NO. 7 MEASURED AT 120° F.

FIGURE 6

The importance of the center rod was evidenced when stuck rod conditions were being investigated. It was determined that the reactor could not be shut down under cold conditions, 83 ^{o}F, with the center rod stuck above 16.1 inches and the side rods inserted, nor could full power operation be attained for maximum xenon conditions with the center rod stuck below 15.3 inches and the side rods withdrawn. A summary of the stuck rod conditions for the SL-1 forty element loading appears in Table IV below:

TABLE IV

STUCK ROD CONDITIONS

Reactor Condition	Critical With:	Shutdown With:
Cold (83 ^{o}F) No Xenon	Four side rods in Center and one side rod in	Two side rods out
Hot (300 psig) No power Maximum xenon	Two side rods in	Four side rods out
Full power 3 MW(t) Maximum xenon	One side rod in	

c. Feedwater Temperature Effect

The effect of feedwater temperature on core reactivity was determined by regulating the main vacuum to change feedwater temperature. The test was performed with constant feedwater flow of 5,000 lbs/hr and 7,600 lbs/hr for feedwater temperature between 143 ^{o}F and 175 ^{o}F. The change in core reactivity was too small to be measured by the control rods, thus it is considered negligible.

d. Venting Effect On Reactivity

During reactor startup, dissociation gases are vented every 20 minutes through the reactor vent valve until the air ejectors are turned on, and when the vessel pressure reaches 250 psig. The pressure loss induced by venting causes a reactor transient. The reactor periods observed during 15 second venting operations are tabulated in Table V.

TABLE V

PERIODS RESULTING FROM REACTOR VENTING

Reactor Pressure psig	Resulting Periods Seconds
50	400
100	350
150	300
170	200
200	150
220	70
240	40
260	15
270	7

There is no hazard involved in the venting transient because the self-regulating characteristics of the boiling water reactor stabilize the transient immediately. However, to eliminate the possibility of a spurious short period scram, the venting operation is limited to a 15 second duration and the 10 second period scram relay is by-passed when the vent valve is operated.

e. Core Life

It is presently too early to predict core life time. Although rod bank position for various reactor conditions are recorded every 200 MWD of plant operation for comparison with core lifetime predictions, changes in the core loading and the limited reactor operation make a core lifetime prediction premature at this time. Design core lifetime is 6.3 megawatt years.

During the next year of SL-1 plant operation stainless steel and aluminum-nickel alloy coupons will be placed in the reactor environment to study system and core corrosion, and fission break detection methods will be studied and possibly expanded to include newer methods. The planned power extrapolation will intensify interest in core performance.

B. REACTOR VESSEL

1. Description

The SL-1 reactor pressure vessel, a carbon steel vessel with stainless steel cladding, was designed for 400 psig pressure with a metal temperature of 500 °F. The vessel consists of an ellipsoidal dished bottom, a cylindrical center section with a flange and flat cover plate. Figure 7 shows a cross section view of the reactor installation.

2. Discussion

Except for a head gasket leak which developed on April 2, 1959, performance was good. Upon removal of the head, inspection of the gasket revealed that the outside retaining ring on the outer gasket had jumped the gasket groove in a five degree quadrant. Apparently, the gasket was oversized and never seated properly, and thus prevented the inner gasket from sealing. When the head was removed to replace the gaskets, the head studs were inspected for material failure since there was the possiblity that they had been previously over-stressed. The difficulties in removing several frozen studs resulted in almost one month of down time. The performance of the replacement gaskets has been satisfactory to date. Refer to Report No. IDO-19001, SL-1 Reactor Vessel Head Removal and Gasket Replacement.

Tests showed the SL-1 steam quality to be very good. It was found that steam quality increased from 95.6 per cent at 2,500 lbs/hr to 99.9 per cent at 9,000 lbs/hr main steam flow. Although the plant was designed for a steam separator in the main steam line, the high steam quality did not warrant its installation. An evaluation of the mechanism of water carryover has not been completed to date. The effect of the steam interface velocity, the reactor vessel steam dome size, and the baffle separator in the reactor are being investigated.

3. Conclusions

The reactor head gaskets should not be replaced until a leak develops. Because of the possibility of the replacement gasket leaking, at least two spare sets of gaskets should be kept at the SL-1 site.

To reduce wear to the turbine from moisture carryover during startup, bypass steam flow should be increased to 5,000 lbs/hr before warming the turbine. As the turbine is being loaded a total steam flow of at least 5,000 lbs/hr should be maintained.

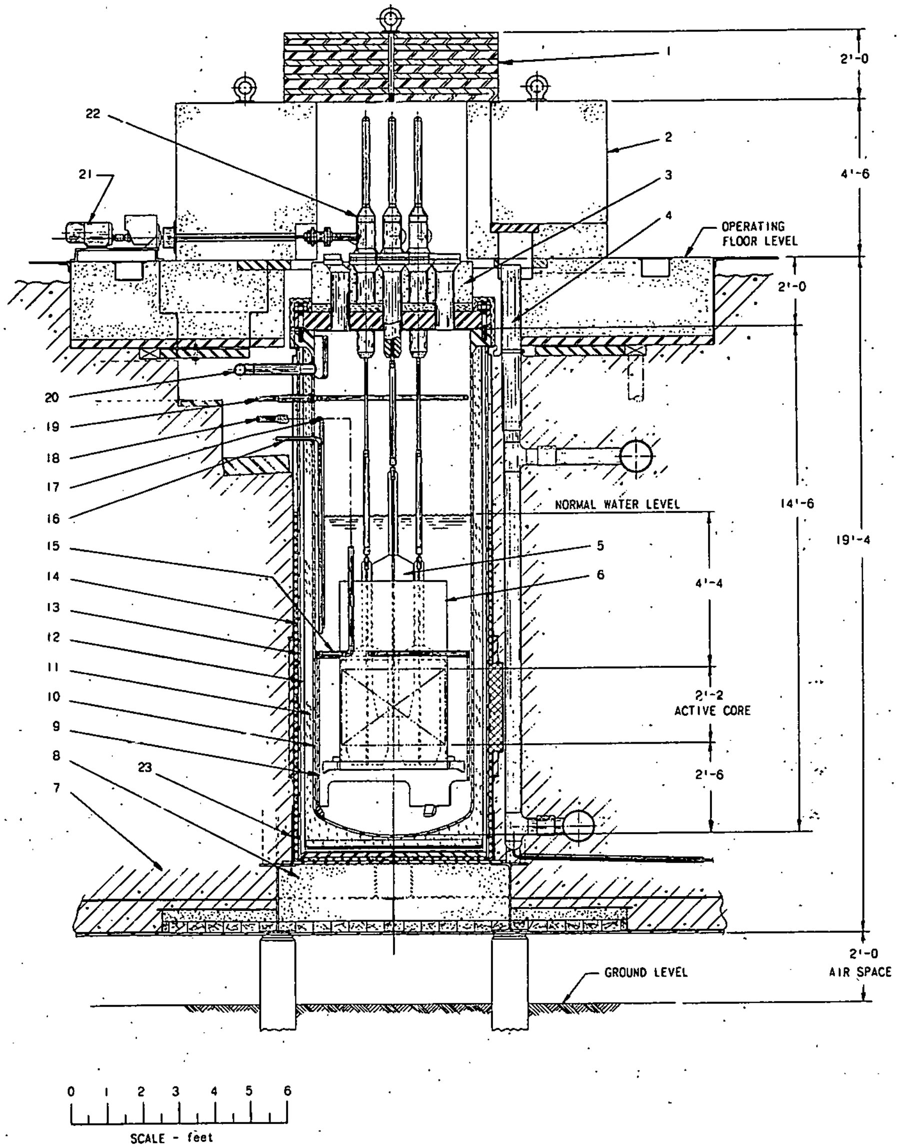
C. CONTROL ROD DRIVES

1. Description

The forty element loading utilizes five cross control rods composed of cadmium sheets with X-8001 aluminum-nickel alloy cladding. Figure 8 shows a full-cross control rod. The active portion of the cross is

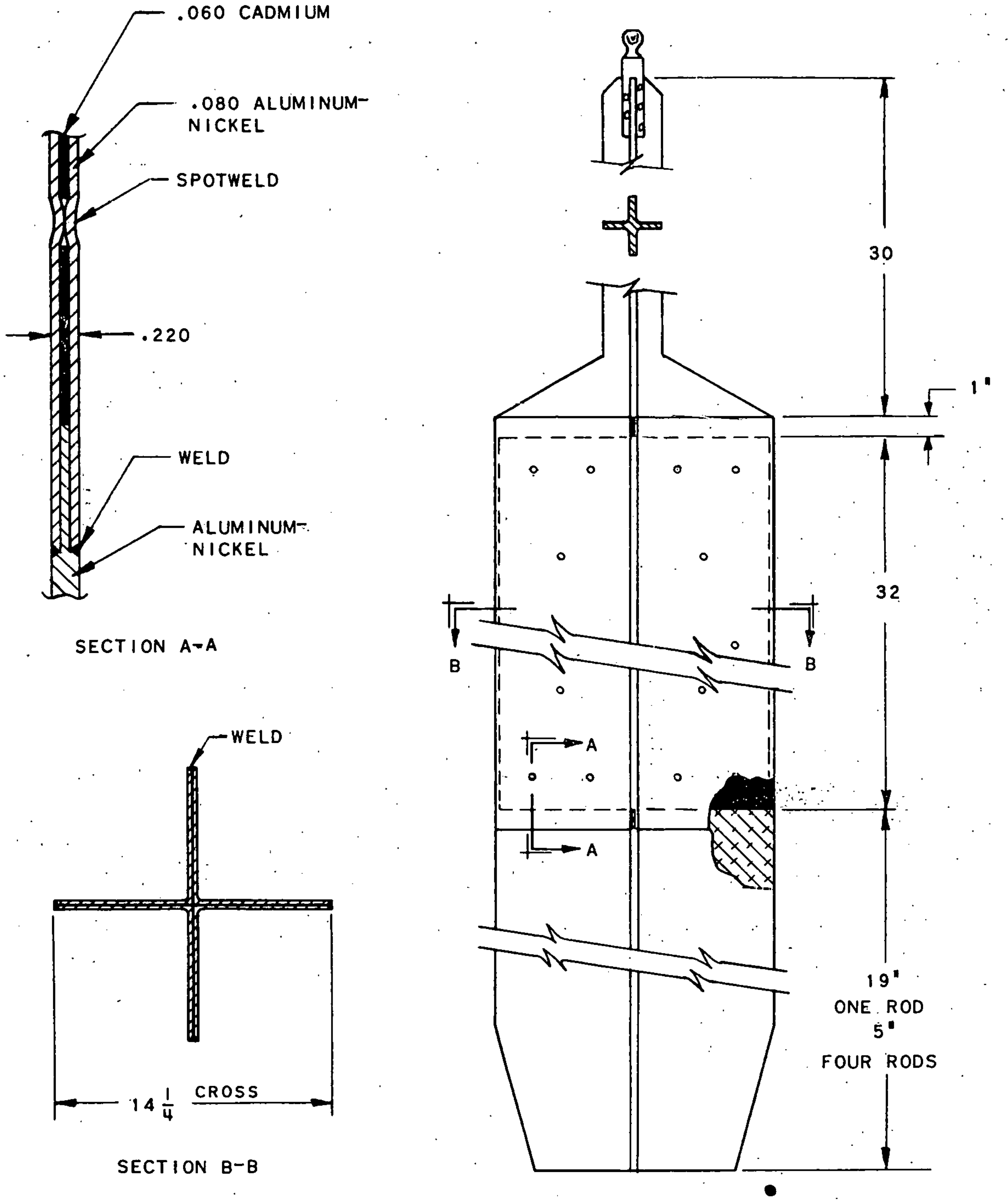
LEGEND

1. LAMINATED TOP SHIELD
2. CONCRETE SHIELD
3. DRY SHIELD MIXTURE
4. INSTRUMENT WELL
5. CONTROL ROD
6. CORE STRUCTURE
7. GRAVEL
8. DRY SHIELD MIXTURE
9. THERMAL SHIELD
10. PRESSURE VESSEL - 4'-6 O.D.
11. INSULATION
12. AIR SPACE
13. SUPPORT CYLINDER
14. LEAD THERMAL SHIELD AND COOLING COILS
15. FEED WATER SPRAY RING
16. SEPARATOR RETURN LINE
17. PURIFICATION PURGE LINE
18. FEED WATER INLET
19. BORON SPRAY RING
20. STEAM LINE
21. CONTROL ROD DRIVE MOTOR
22. CONTROL ROD DRIVE
23. BORAL SHIELD



REACTOR VESSEL VERTICAL SECTION

FIGURE 7



CROSS TYPE CONTROL ROD

FIGURE 8

14 by 14 inches by 0.060 inches thick, and 34 inches long. The cadmium sheets are perforated at intervals by 0.5 inch diameter holes, through which the aluminum-nickel cladding is dimpled and spot welded to provide support. The centrally located rod No. 9 is furnished with a 17 inch follower section made of solid aluminum-nickel plate and the remaining rods have 5 inch followers.

The cadmium in the rods is positioned 3-1/8 inches below the nominal lower fuel dimension when the rods are fully inserted. Stainless steel ball-joint end fittings attached to the upper end of the rods are used to connect the rods to the drive mechanisms. A set of concentric springs located in the rod housings above the reactor head act as shock absorbers and a positive stop during rod drops.

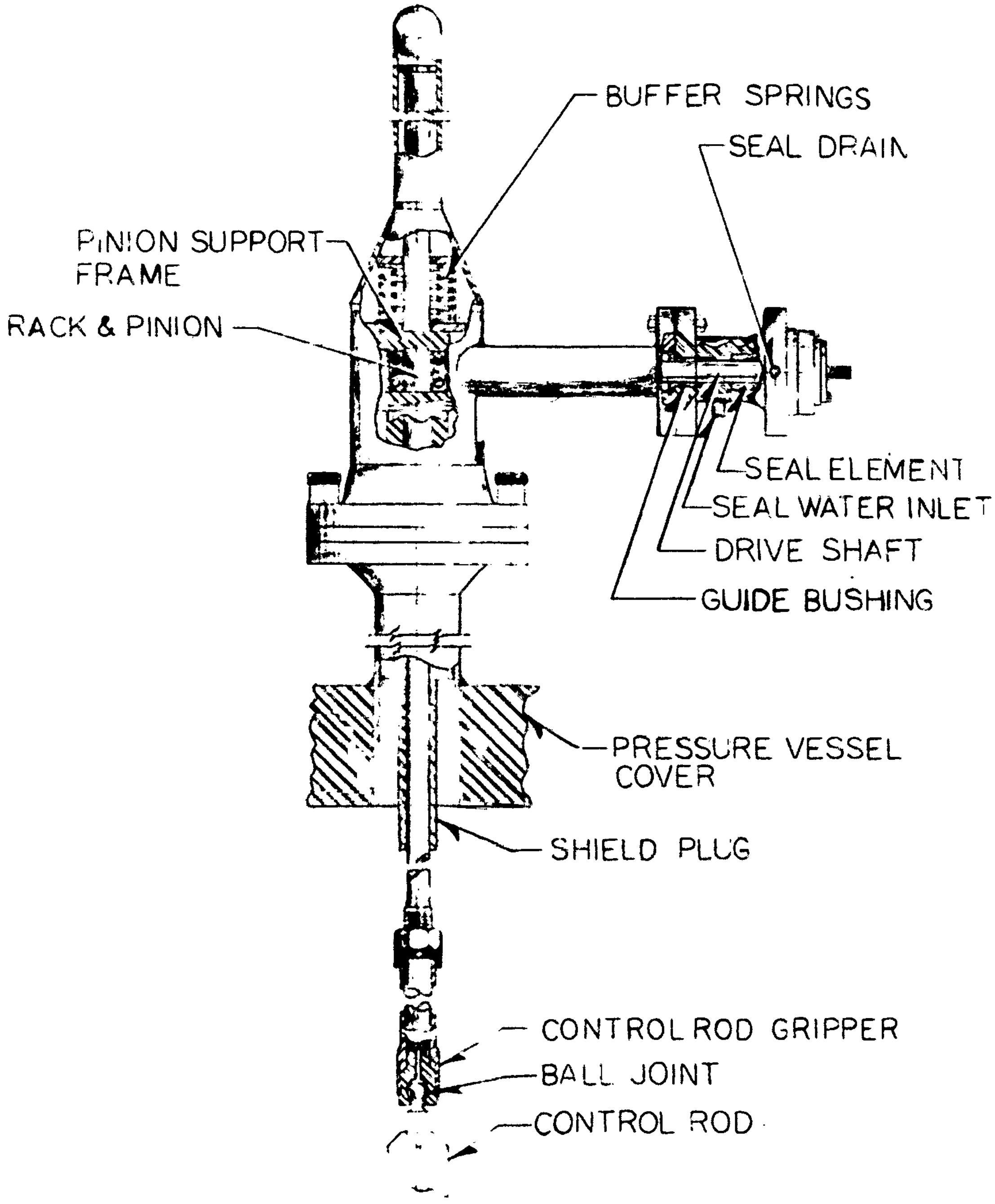
Figure 9 shows the control rod drive mechanism. Vertical linear motion is imparted to the rods by a rack and pinion drive mechanism. The pressure seal on the pinion drive shaft is a positive clearance, labyrinth-type, utilizing high pressure seal water from the feedwater system. Controlled leakage through the seal is bled to the precooler and condensate return tank. A conventional 1/8 hp electric gear motor equipped with a magnetic-disk brake is positively engaged with the pinion shaft by an electro-magnetic clutch. The mechanism is so designed that a scram signal will not only release the magnetic clutch but the drive motor will try to drive the rod down through a unidirectional cam clutch. In the event of a power failure the drive motor current is supplied by the battery powered emergency power supply. Normal rod travel is restricted by mechanical gearing to 2.85 in/min for the side rod and 1.85 in/min for the center rod. The corresponding maximum rate of reactivity addition is 1.5×10^{-4} K/sec.

A synchro-transmitter and cam actuated micro-switches are coupled to the pinion drive shaft through gears to provide rod position indication and actuate position lights and interlock controls. The rods may only be withdrawn individually but two rods may be driven in simultaneously. They may not be ganged in either direction. The center rod has individual control and the side rods driven by one control through a selector switch.

2. Discussion

The rod drives were designed for operation with two negator springs attached to each pinion shaft to limit free-fall shock forces in the mechanism.

The control rod drive performance was satisfactory with minor exceptions. Control rod operation at SL-1 has revealed that the rod drop times increase following reactor shutdown periods. A buildup of particulate matter, or crud, was observed to occur in the water seal rings, pinion bearings, bushings, and rack housing areas. When crud buildup interferes with rod performance and prevents the rod from meeting prescribed rod drop requirements, the condition is temporarily corrected by removing one of the negator springs.



SL-1 CONTROL ROD DRIVE

FIGURE 9

After several months operation, the control rod chrome plated seal shafts were inspected and found scored by the stellite floating seal rings. Figure 10 shows the disassembled seal unit from rod No. 9 after 1,000 hours of operation. The damage is believed to have resulted from high temperatures and crud accumulation in the seals. Four seal shafts were replated as a result of scoring damage.

A five micron filter was installed in the rod seal-coolant influent line in an attempt to screen out particulate matter, but the filter installation has not reduced crud accumulation in the seals.

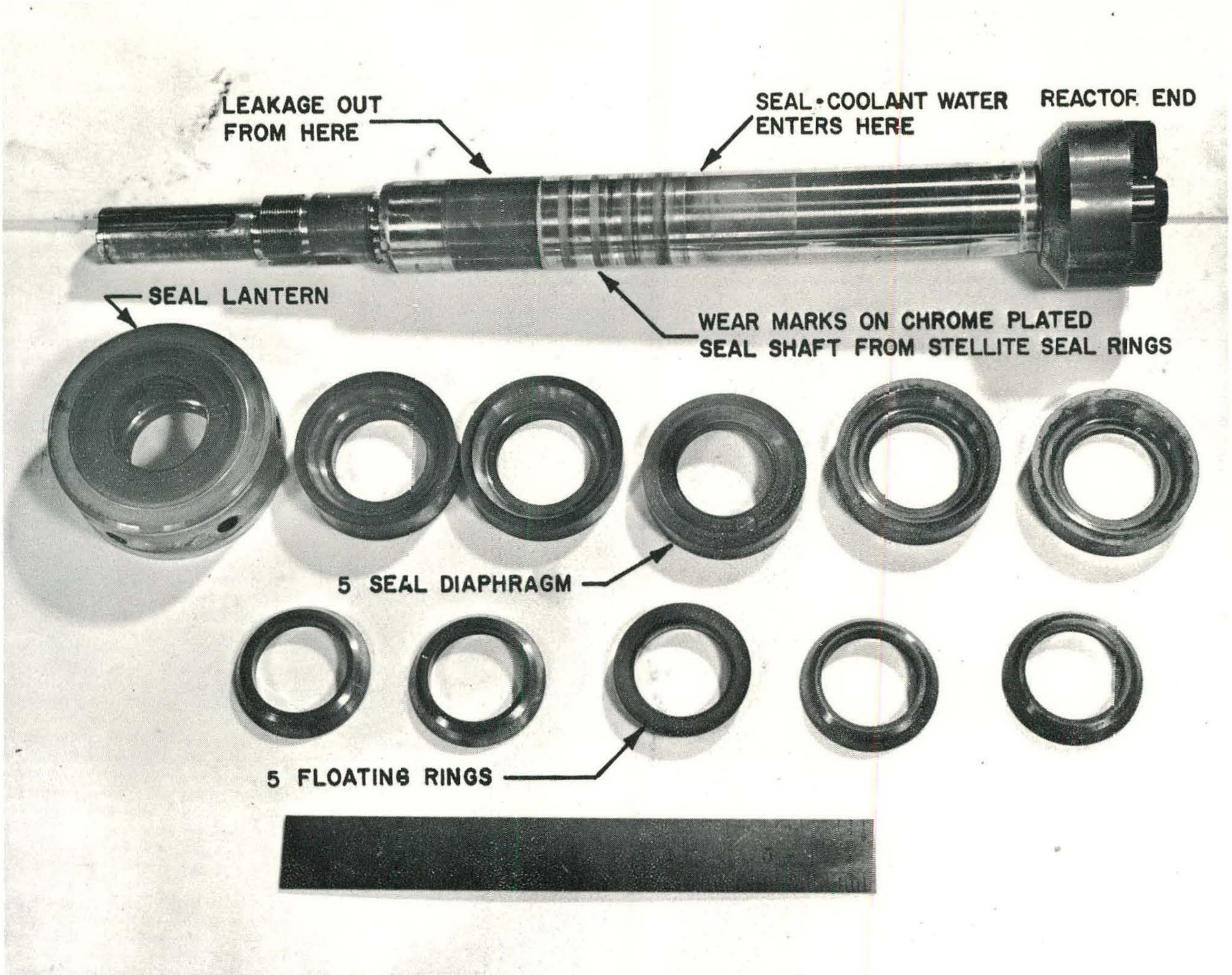
The couplings in the control extension rods were found frozen due to galling. The extension rods consist of three sections: the rack, the extension, and the connector. The threaded couplings between these pieces and the spanner nut, which actuates the ball-joint coupling to the control rod, were galled to the extent that they could not be actuated. The most important of these is the spanner nut, since the rods cannot be disassembled to clear the way for fuel transfer if this nut is frozen. The spanner nut and connecting rod are made from 17-4PL stainless steel. The threaded section of the connecting rod was originally flash chrome plated .00025 inches thick. The plating had evidently worn off and galling occurred between the two stainless steel pieces.

One surface on the couplings of four spare extension rods has been subsequently chrome plated .005 inches thick. The five extension rods presently in use in the reactor have not been replated, since no attempt to remove these extension rods has been made.

Some difficulty was experienced in adjusting the cut off switches for the rod drives following underscram conditions. On several occasions the control rods, when scrammed, would drop one or two inches below the reference zero position and pass the cut off point in the microswitch cam. Thus the drive motor continued driving the rod in through the unidirectional mechanical clutch. The rod would stop on the clampening springs and shear the teeth on the cast iron coupling gear between the gear motor and the clutch assembly. The design shear point in the assembly was originally an aluminum key in the drive motor coupling gear, but experience has shown that the cast steel gears are effective and easier to replace. The span on the cam adjustment was modified on all rod mechanisms to prevent damage from rod drop over travel.

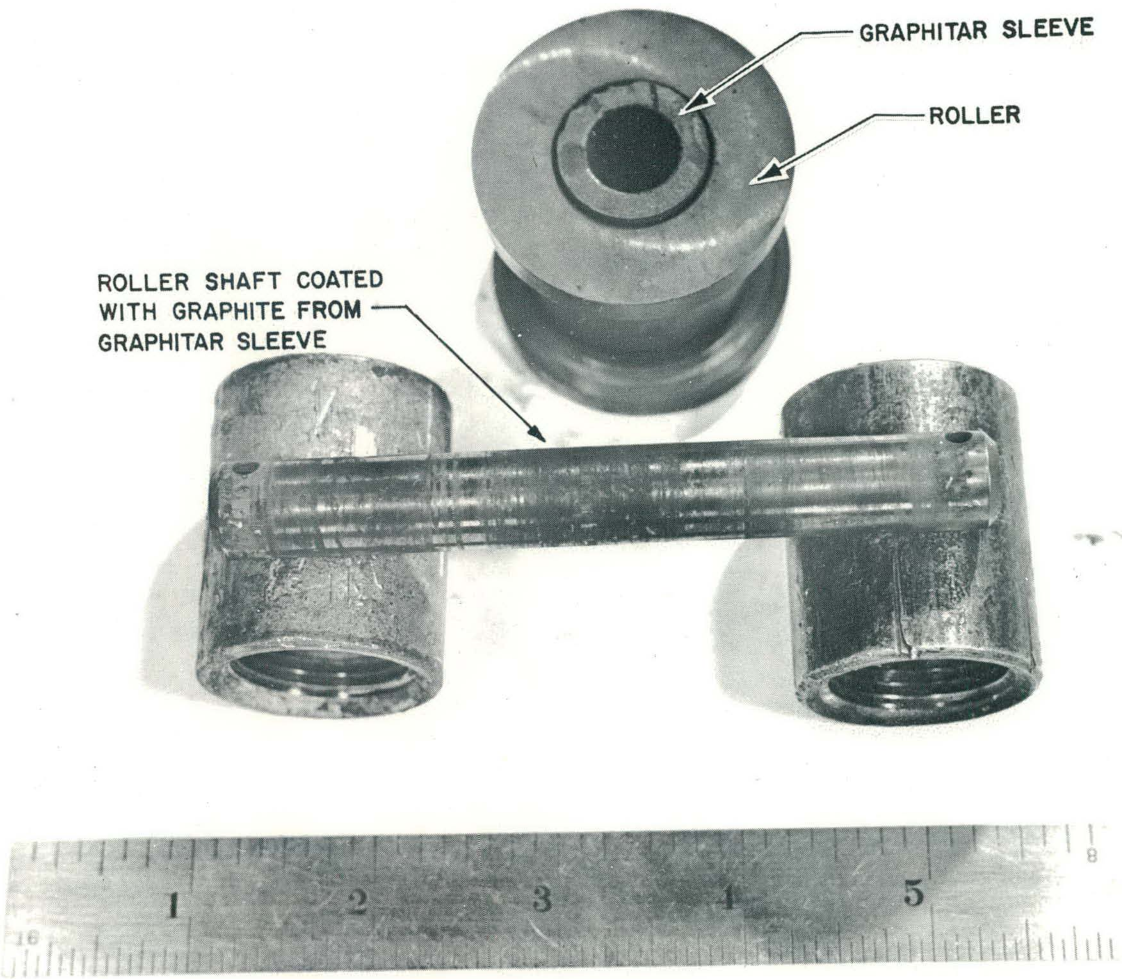
Disassembly and inspection of the rod mechanisms during the annual plant maintenance period revealed that the neoprene "O" rings and sealed bearings in all the rod seal units needed replacements. The failure of these components is attributed to high temperature operation.

Inspection of the graphitar sleeves in the backup rollers for the control rod racks showed that the graphite was plating onto the roller shafts. Figure 11 shows the backup roller removed from control rod No. 9. Binding in the backup roller between the graphitar sleeve and the roller shaft had previously caused the rods to stick. The clearance in



LABYRINTH TYPE ROD SEAL DISASSEMBLY

FIGURE 10



CONTROL ROD RACK BACKUP ROLLER

FIGURE 11

this area was increased and subsequent operation has been without difficulty. However, the graphite plating onto the roller shaft observed in inspection indicated that unless the shafts are cleaned periodically, binding is apt to recur between the backup roller and the roller shaft.

3. Conclusions

Generally, the control rod operation over the past year was satisfactory, but difficulties experienced indicate areas for improvement. First, the high temperature operations of the rod seals should be reduced by improving the air ventilation in the rod housing cavity. The crud accumulation on friction surfaces should be eliminated by improved filtration of seal-coolant water and possible increased flow rates through the seals to flush crud accumulated from the reactor side. Other improvements include finding a suitable substitute for the graphitar sleeve in the backup roller and replating the chrome threads on the extension rods.

D. REACTOR SHIELDING

1. Description

The bulk of the reactor biological shielding is 16 feet of gravel. The ion exchangers, contaminated water storage tank and spent fuel storage wells are embedded within the gravel shield. The outer cylindrical thermal shield is composed of a 1-1/4 inch layer of lead contained between the 7/8 inch thick pressure vessel support tank and a 1-1/4 inch thick cylindrical steel plate at the reactor core level. The bottom thermal shield is composed of a 1-1/4 inch layer of lead contained between the 1-1/2 inch thick pressure vessel support tank bottom and a one inch thick steel plate. The shield cooling tubes are within this 1-1/4 inch lead layer. A boral sheet below the reactor vessel to capture thermal neutrons reduces the possibility of thermal neutron activation of air and possible airborne duct below the reactor building. Additional shielding in the form of steel shot and gravel is contained directly below the reactor. A steel retaining plate supports all the bottom shielding.

During reactor operation the air space below the reactor is inaccessible because of the radiation level. A radiation monitoring survey in this region indicated radiation levels as high as 10R.

The top biological shielding consists of steel plates and concrete and steel shielding blocks. Additional shielding above the control rod drives is provided by two section of steel and masonite plates to reduce the dose rate to less than 10 mr/hr.

2. Discussion

Shielding tests were performed in the shield tank to determine the degree of gravel activation. Gravel samples were taken from the shield plug nearest the reactor in the experimental beam hole and a spectrographic analysis was completed on the samples. The major activity detected was Pa^{333} which results from the (n,γ) reaction of the Th^{232} contained in the gravel.

3. Conclusions

The conclusion of the tests performed and radiation survey data taken at full power operation is that the reactor shielding is adequate except for the area around the purification vault and the simulated heat load heat exchanger.

E. REACTOR VENTILATION SYSTEM

1. Description

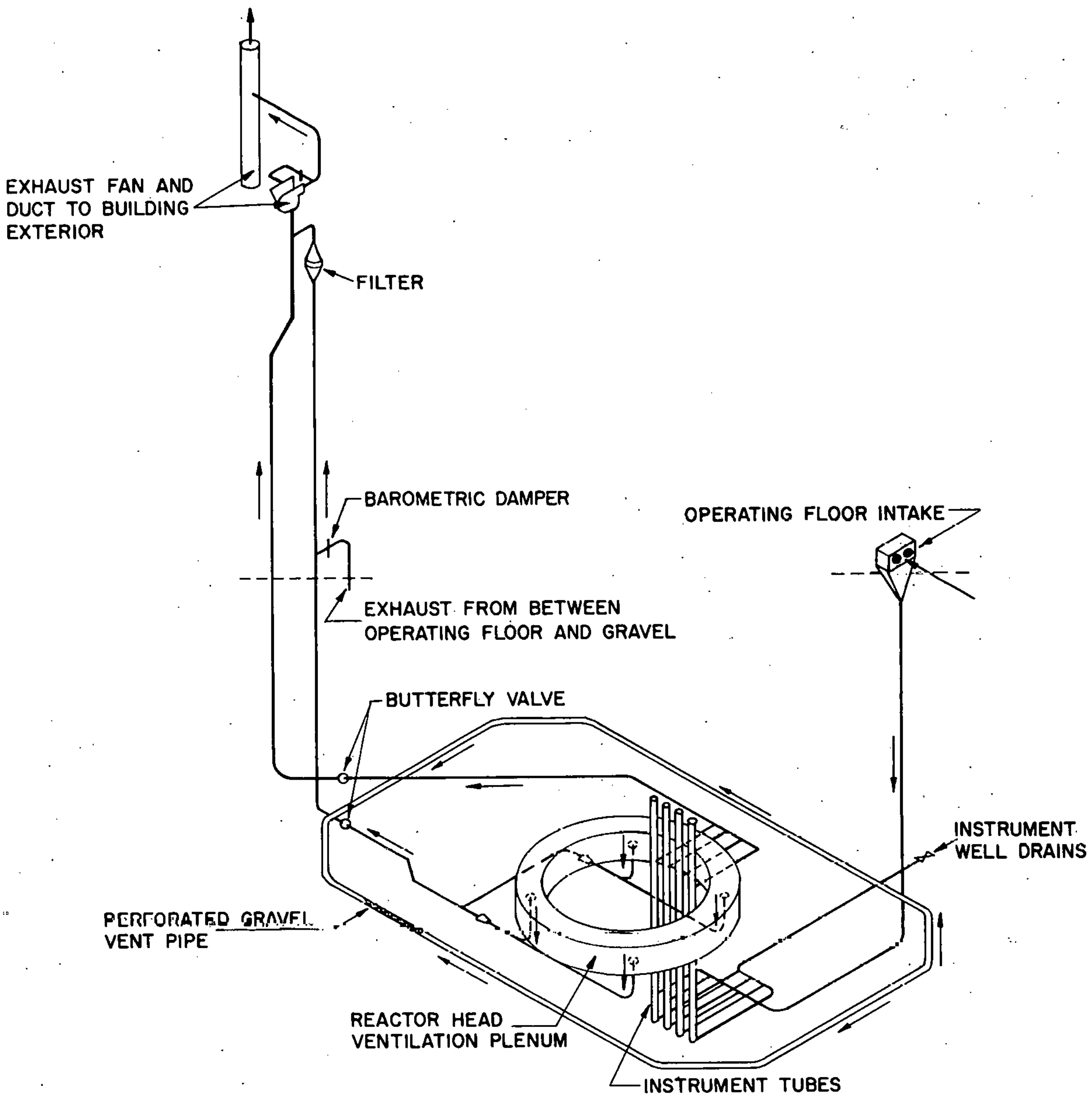
The reactor ventilation system consists of an instrument well fan, which induces flow from four separate areas including: the instrument wells, the reactor plenum, the gravel shielding, and the space between the gravel shielding and the reactor operating floor deck plates. The exhaust from the instrument wells and the reactor plenum join into one duct and flow in each is controlled by manual butterfly valves. The gravel exhaust and exhaust from between the operating floor and gravel join into another duct where the flow is controlled by a barometric damper.

A negative pressure is maintained to prevent irradiated gravel dust from filtering up to the operating floor. The building ventilation system draws hot or cold air from the main condenser ducts, and discharges at a uniform temperature to the operating floor. The operating floor exhausts to the main condenser exhaust duct through a back draft damper. Figure 12 is a line diagram of the reactor ventilating system.

2. Discussion

The operating temperatures in the reactor plenum indicated that the reactor ventilating system was underdesigned. The air temperature above the reactor head was often over 260 °F which interfered with rod seal cooling and instrument cables.

The instrument wells were cooled below 115 °F, even with outside ambient temperatures above 90 °F, but during the annual maintenance program it was discovered that the nuclear instrument power supply coaxial cables were affected by operation at excessive temperatures in the cable conduit near the reactor. Although the nuclear chambers were successfully cooled below the design temperature, 170 °F, the cable conduits were subjected to 200 °F and this high temperature operation resulted in shorted cables from insulation breakdown.



PLANT VENTILATION FLOW DIAGRAM

FIGURE 12

3. Conclusions

Although maintenance requirements for the reactor ventilation systems are simple, good maintenance procedures and an adequate supply of spare parts are important, since the loss of the instrument well fan can result in a condition requiring reactor shutdown.

The reactor ventilation system operated satisfactorily throughout the year although the cooling effect on the reactor plenum is marginal. The plenum ventilation should be supplemented to reduce operating temperatures around the control rods and water level transmitters.

F. SHIELD COOLING SYSTEM

1. Description

The shield cooling system shown in Figure 13 is a closed loop system through which water is circulated to remove heat from the reactor shielding. The system consists of two parallel sets of cooling coils embedded in the lead shielding where the heat is absorbed, a shield coolant/condensate and a shield coolant/air heat exchanger where the heat is dumped, a coolant circulating pump, and the associated piping, valves and instrumentation. The system was designed so that during normal operation a forced circulation circuit would be in service and remove an estimated 40 KW of heat during full power operation. A natural circulation circuit designed to remove 6 KW of heat was incorporated into the system to provide emergency shield cooling in the event of shield coolant pump failure.

2. Discussion

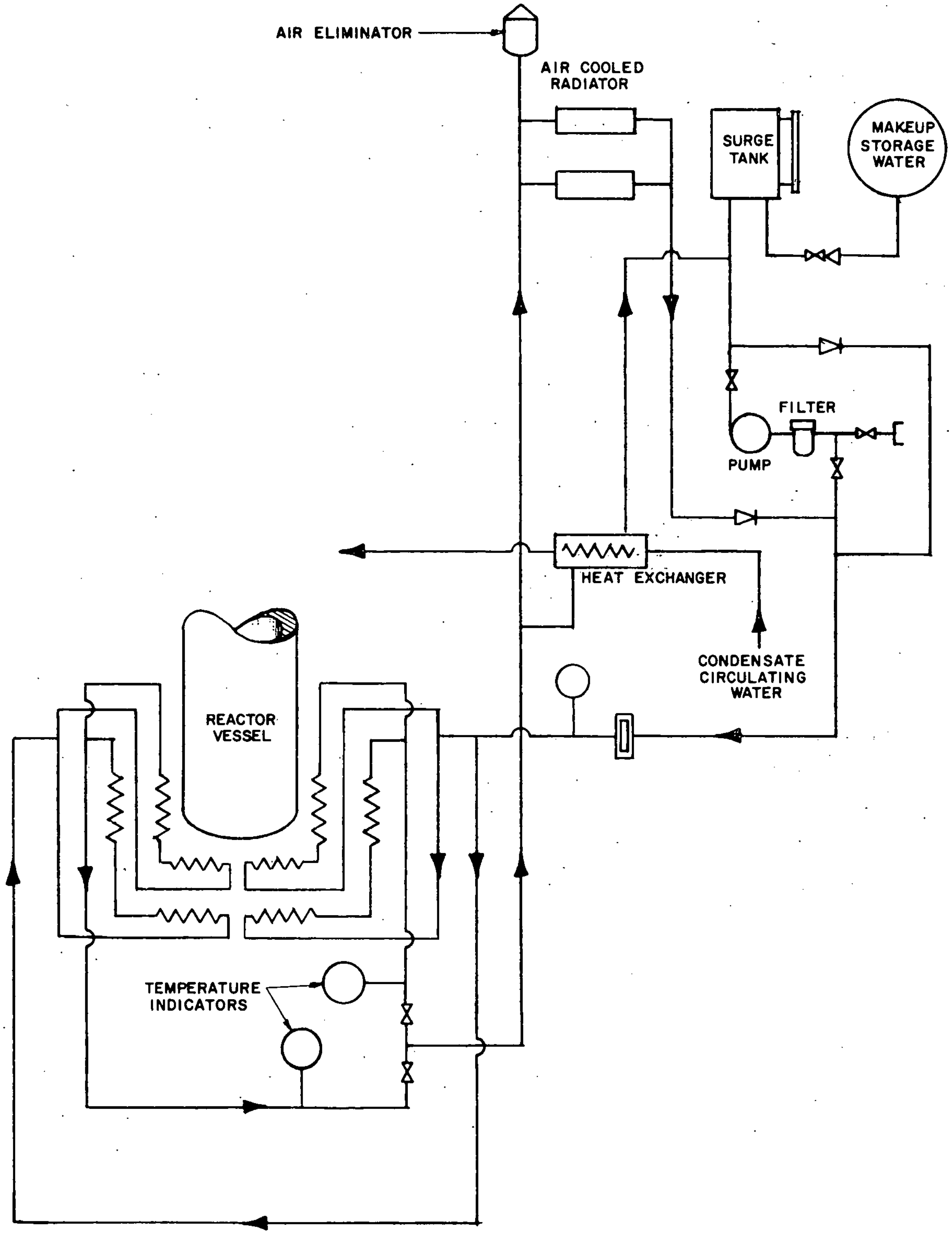
During full power operation, 3 MW [(th)], it was found that the natural circulation circuit provided adequate shield cooling. Although designed to be operated normally, the forced circulation circuit is used only while performing tests on the system.

Tests showed that 12.6 KW of heat was being removed from the shield when operating the reactor at 3MW (t). With natural circulation, approximately 50 per cent of the heat was dissipated to the ambient through pipe losses; the remainder was dissipated by the shield coolant/air heat exchanger. A negligible amount of heat was transferred in the shield coolant/condensate heat exchanger, because of the high condensate temperature.

After a year of operation an accumulation of scale had built up in the system piping. This condition was corrected by flushing the system with demineralized water.

3. Conclusions

During normal plant operation the shield cooling system should be operated by using the natural circulation circuit. Condensate flow to the shield coolant/condensate heat exchanger is not required.



SHIELD COOLING SYSTEM

FIGURE 13

Periodically forced circulation should be used in order to remove scale from the piping. Following the forced circulation operation the filter element should be replaced with a new, or clean element.

G. BORIC ACID INJECTION SYSTEM

Shutdown of the SL-1 can be accomplished by boric acid poisoning. Eighty gallons of saturated boric acid solution, approximately 3.7 per cent by weight, is kept ready for injection into the reactor water by a hydraulic hand pump or by gravity feed if the reactor is not pressured. The use of this solution would reduce core reactivity by more than 10 per cent K.

Piping was modified to include a pressure gauge in order to permit periodic hydrostatic testing of the hand pump without actually injecting the boric acid solution. The entire amount can be pumped into the reactor in approximately 40 minutes. The hand pump is checked in this way at 450 psig on a monthly basis. The boric acid solution is monitored and is changed on a six month basis.

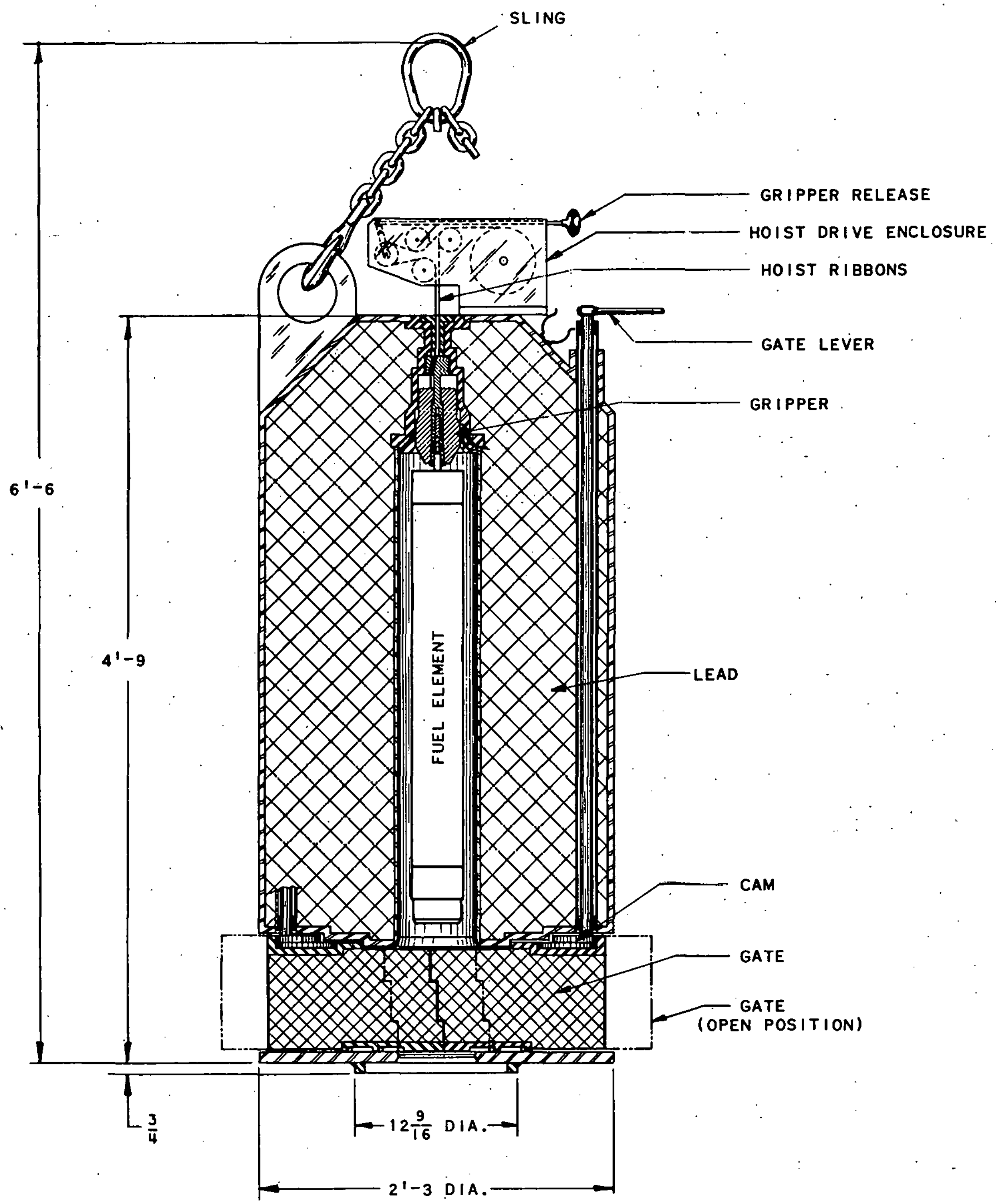
The relatively simple design of this system precludes troublesome operation and maintenance requirements are negligible.

H. FUEL HANDLING

Fuel transfer between the reactor and the fuel storage wells is accomplished by the use of a fuel transfer coffin, Figure 14. The reactor shield blocks are moved and the reactor vessel is filled with water for shielding. Then the control rod mechanisms and extension shafts are disconnected to clear the vessel head. Reactor fuel elements are raised individually into the coffin by the winch with gripper through an opening in the vessel head. The coffin, with contained element, is then placed over the fuel storage wells where the element is lowered into storage position.

Practice runs with a weighted dummy element revealed that it is possible to disengage an element in the coffin if the gripper head is not held taut by the cables. Also, if an element should become wedged in the coffin, there are no means provided for dislodging it safely.

It is recommended that the gripper mechanism be modified to provide a more positive means of attachment and that a method be devised for handling an uncoupled or wedged element in the coffin.



FUEL ELEMENT COFFIN

FIGURE 14

III INSTRUMENTATION AND CONTROL EVALUATION

The reactor plant instrumentation and control systems performed in accordance with design specifications. The automatic center rod control system was demonstrated to be effective over the full range of reactor power operation. Several minor modifications were made on the instrumentation, of which the primary change was the addition of captive key bypass switches in the reactor scram circuits to facilitate testing and maintenance during sustained operation. The principle modification recommended for improved performance involves rearranging the feedwater piping to permit the feedwater flow transducer to monitor water added to the reactor through the control rod seals.

A. NUCLEAR INSTRUMENTATION

The reactor nuclear instrumentation consists of eight separate channels. Figure 15 is a block diagram of the channel arrangements. The core neutron instrumentation, Channels I through VI, covers the range from source level to full power operation. The radiation detectors for these channels are positioned in vertical instrument wells in the gravel shield outside the reactor vessel.

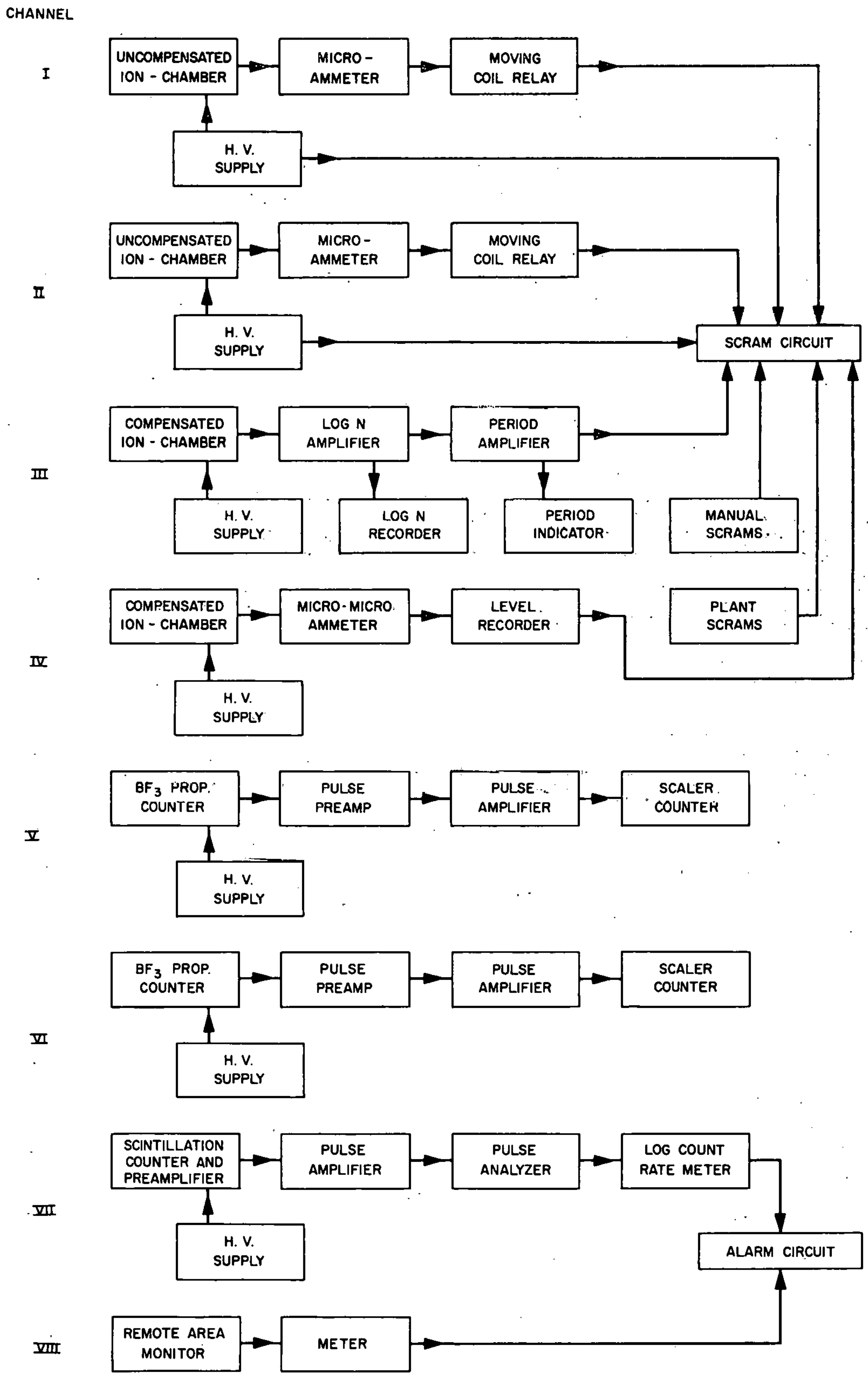
The gamma activity of the air ejector vent gases is monitored by a scintillation counter, Channel VII, to detect a fuel element rupture. In addition, the plant system, Channel VIII, is monitored by remote area detectors in five locations.

1. Discussion

Nuclear instrument changes made during the recent year include the alteration and relocation of Channel VII scintillation detector, the relocation of one of the remote area detectors for Channel VIII, the addition of a scram contact to the Channel IV linear level recorder, the addition of bypass switches for all reactor scrams, and the installation of an electronic rod drop timer.

The scintillation detector was physically moved from the operating floor to the fan floor to eliminate background interference and the detector head cover was changed from stainless steel to aluminum to minimize noise due to scattering in the stainless steel.

As the result of improper valving in the purification system during reactor shutdowns, the reactor core was partially uncovered which created a radiation hazard on the operating floor. The incident details are reported in Malfunction Report No. 7. To prevent a recurrence of



NUCLEAR INSTRUMENT BLOCK DIAGRAM

FIGURE 15

this incident, an alarm system was installed, which is actuated by a low reactor water level, and the constant air monitor on a remote area detector repositioned above the reactor vessel.

The purification vault remote area detector was replaced by a new detector located above the reactor, since the purification detector was affected by the temperature and humidity in the vault and frequently malfunctioned.

A scram contact was installed on the Channel IV linear level recorder to be used as an alternate for Channels I or II when maintenance or routine testing are performed. The Channel IV scram is normally bypassed.

Captive key actuated bypass switches were installed for all the reactor scrams to allow for maintenance and testing. Each scram circuit is tested weekly during sustained operation as well as prior to startup.

An electronic rod drop timer was permanently installed in the control rod circuit for continuous monitoring of rod drop performance. Rod drop times are individually recorded for cold and hot conditions during reactor startup and during shutdown when possible.

Calorimetric calibration curves for the nuclear instrumentation were taken. In practice, these curves are used only for approximations of operating power because of the inaccuracies caused by drift in the instruments. The operation crew keeps an hourly record of the reactor power as calculated from the reactor heat balance.

2. Conclusions

The nuclear instrumentation, generally, has been satisfactory, when the frequency of startups and shutdowns during the year's operation is taken into consideration. Preventive maintenance and calibrations were performed on a six month basis with several minor maintenance tasks accomplished as needed.

B. AUTOMATIC REACTOR CONTROL

There are two separate automatic control systems for the SL-1 reactor; the automatic center rod control, which regulates reactor power to maintain operating pressure in compensation for load changes, and the automatic bypass steam control system which balances total steam flow to maintain a constant operating pressure of 300 psig with steady reactor power output. The driving signal for both control systems is a differential pressure recorder. The two systems can not be operated simultaneously because of control feedback interference.

The automatic center rod control operation is favored over the bypass steam control, because of its ability to regulate reactor power directly.

The bypass control system was originally designed to operate in automatic with two parallel valves rated at 3,000 lbs/hr, or 10,000 lbs/hr. Because the capacity of the small valve exceeded 6,000 lbs/hr in actual operation, the automatic control driver for the larger valve was eliminated.

A test was performed to determine the recovery characteristics of the center rod controller. At 2 MW(t) reactor power, a 60 KW turbine load change caused an average reactor pressure deviation of 5.5 psi and the automatic center rod control regulated power to return the reactor to the nominal 300 psig in an average of one minute and forty seconds.

The performance of the center rod reactor control system was satisfactory and maintenance requirements were limited. Preventive maintenance was performed on a six month basis with minor system adjustments performed intermittently.

C. PROCESS INSTRUMENTATION

The reactor plant process instrumentation includes conventional pressure transducers, recorders, thermocouples, process alarms, and controls. The feedwater system and main condenser air system may be operated in automatic control.

The feedwater flow system may be selected for single or three element control. In single element control the driving signal is reactor water level. Three element control is based on reactor water level, total mainsteam flow, and feedwater flow.

The condenser air system control prevents the condenser from freezing when outside air temperatures are low. The average condenser inlet air temperature is used to drive a set of recirculating dampers to maintain temperature on the condenser inlet above 40 ^{o}F. In addition, the average condenser outlet air temperature is monitored to regulate the condenser fan speed through a hydraulic coupling between the fan and the drive motor.

1. Discussion

The main condenser fan speed control system was modified to automatically reduce fan speed following a scram condition. The fan motor circuit is opened on a scram signal and prior to the modification the reactor operator manually unloaded the drive motor to reduce the initial power surge when restarting the motor.

The purification pump control circuit was modified to incorporate an automatic cut-off when reactor water level is below -6 inches or above +36 inches. This limit switch insures that water cannot be removed or added unknowingly through the purification system when the reactor is shutdown.

The turbine vacuum recorder transducer was relocated to a position adjacent to the turbine exhaust trunk. In the former arrangement, condensed steam created a water log in the piping between the transducer and the exhaust trunk causing an erroneous vacuum reading.

The process instrumentation operated in accordance with design specifications except for the feedwater control system. The three element feedwater control was subject to drift and over compensation.

The feedwater flow signal did not account for water added to the reactor by rod seal-coolant water through the control rod seals. Consequently, when the feedwater and mainsteam flow were matched, the reactor water level increased due to rod seal-coolant flow. The feedwater controller closed the feedwater valve intermittently, compensating for the high water level resulting in feedwater flow oscillations. The system was eventually balanced by adding an error signal to the feedwater signal so that the combined rod seal-coolant flow and feedwater flow matched total mainsteam flow.

The single element control system was operated only for a brief period. Optimum performance was never attained primarily due to feedwater transducer troubles.

THIS PAGE

WAS INTENTIONALLY

LEFT BLANK

IV MECHANICAL SYSTEMS EVALUATION

A. MAIN STEAM SYSTEM

Steam at 300 psig, 421 ^{o}F is generated in the reactor and flows directly to the turbine generator, air ejectors, turbine gland seal system, and to the space heat exchanger. Steam bypassed but not used by the space heat system is discharged to the main air cooled condenser through a back pressure regulating valve, which is adjusted to maintain 40 psig below the bypass valves.

The system contains standard process instrumentation including pressure gauges, pressure and flow transducers, and safety valves. In addition, process scram circuits are provided which are actuated by high, or low, pressure and safety valve flow.

The system is of welded construction except for a 4 foot bolted flange spool piece and bolted valve bonnets.

Performance of the main steam system has been generally satisfactory. Operational records show that malfunctions, as enumerated in Section I have accounted for 573 hours of unscheduled downtime. A significant amount of the maintenance downtime can be attributed to the large number of heat cycles this system has been subjected to during the year. Specific modifications completed have improved performance of the system.

1. Discussion

The main steam system, piping and valves, is of all welded construction except for the valve bonnets and stem packing. The hot and cold cycling of the 304 stainless steel globe valves, ranging in size from 1/4 inch to 2 inches, has caused some downtime and excessive water loss due to leaks.

Manufacturer's recommendations for elimination of the leaks through the valve bonnets include replacement of the 304 type bonnet bolts with high tensile steel to prevent bolt stretch and/or installation of flexitalic gaskets. All bonnet bolts have been replaced and a program is in effect to maintain the valve bolts tightened following a heat cycle. This practice has reduced the valve maintenance downtime to a minor problem. Since the installation of conventional flexitalic gaskets would require removal of the valves for machining, this course of action is not presently planned at SL-1. Based upon SL-1 experience, conventional steam service valves are satisfactory; however, where significant thermal cycling service is anticipated, the use of high strength bonnet bolts and/or use of flexitalic gaskets should be considered. The accumulation of hydrogen in a boiling water reactor from dissociation of water required venting of the vessel during startup until steam flow is established and the air ejectors can be placed into service. The original method of venting the SL-1 reactor consisted of lifting the main steam safety valve for short periods at twenty minute intervals during startup. This method was considered unsatisfactory because of the excessive wear to the safety valve and lack of regulation of the venting time.

At SL-1, a reactor vent to the atmosphere was installed in the main steam line upstream of the main steam stop valve. The installation which includes a glove valve, followed by a quick acting lever type valve has operated satisfactorily. It is concluded from SL-1 operation that specific provision for venting of H_2 and O_2 during startup should be provided.

SL-1 safety valves are the enclosed type to prevent the escape of radioactive steam during valve operation. Rubber "O" rings are installed in these valves to provide a seal between the hand lifted shaft and the bonnet. Since the 60 lb. relief valve installed on the fan floor operates in a high ambient temperature, 130 °F, the rubber "O" rings have melted and siezed the shaft, thus resulting in a bent stem and valve malfunction. Replacement of valve internals resulted. A higher temperature service "O" ring was used.

Prior to installation of the hydrogen vent piping, the main steam safety valve was hand-lifted to vent the system to atmosphere. Frequent opening and closing of the valve caused wear and bending of the stem. Replacement of valve internals was performed. Repeated hand operation of safety valves is not recommended.

Testing of main steam safety valves, following their repair, was accomplished with reactor pressure after reinstallation of the valves. This scheme involves resetting the high pressure scram set point temporarily. The test was performed satisfactorily and the valves were resealed.

As in conventional and reactor steam plants, the periodic testing of safety valves is required. The method used, whether insite or factory test, must be considered by the reactor designer and operator.

The SL-1 automatic control system uses a pressure change signal to adjust the center control rod position. A throttle valve is set to bypass 1500 lbs/hr of steam to the space heat exchanger. When reactor pressure changes as a result of system load changes, a pressure sensing device signals a control system that automatically drives the center control rod in or out to compensate for the increase or decrease in reactor pressure.

The automatic control system has operated successfully in handling specified plant load changes. The automatic bypassing main steam to the condenser, as the system was initially designed, is not needed since the space heat system is capable of absorbing all the steam fluctuations necessary for routine operation for startup and shutdown. The only steam flow routinely required through the throttling valves is a constant 1500 lbs/hr to supply the space heat system.

2. Conclusions

Based upon SL-1 operation, it is concluded that control rod adjustment from a pressure signal is highly satisfactory. Further, the full flow bypass directly to the condenser is used only as a backup system for testing.

B. TURBINE GENERATOR UNIT

1. Description

A standard Worthington Turbine-drive unit, Serial No. 21211, consisting of one Curtis stage and eight Rateau stages is installed to drive a 300 KW EMMCO generator through an E-4 gear. The unit is designed to operate with inlet steam at 275 psig and 415 °F and exhaust to 5 in. Hg. abs. The casing material is cast steel at the inlet end, and cast iron at the exhaust end. Carbon packing rings serve as the shaft seals. Speed control is maintained by a direct acting Woodward UG-8 governor.

2. Discussion

During this report period the turbine generator unit was operated as an integral component in a direct cycle boiling water reactor plant. Operation of the unit was intermittent, with 189 startups recorded. The turbine operated 32% of the time, of which approximately 12% was at full power. Quality of the steam supplied to the turbine during full power operation was as follows:

Pressure - 300 psia
Temperature - 420 °F
Dissolved oxygen in condensed steam - 2-4 ppm
Condensed steam - pH 5 - 6.5
Resistivity of condensed steam - $>1.5 \times 10^6$
Moisture - $>0.3\%$
Non-condensable gases found in condensed steam - 56.6 cc/liter
Fission product activity - 3.16×10^3 d/m/ml (in condensed steam)

Following 3500 hours of turbine operation, the top half of the casing was removed for inspection of turbine internals. The condition of bearings, carbon packing, lubrication system, and reduction gears was found to be satisfactory. The major irregularity was corrosion of the cast steel casing and diaphragms.

a. Turbine Inspection and Maintenance

After lifting the turbine top casing the following work was accomplished:

A general inspection was performed
All turbine bearings were removed, inspected, and wear recorded
All carbon packing, end, and interstage was removed and replaced
Gear case was lifted, bearings removed for inspection, and wear recorded

All gears were inspected and backlash recorded
Coupling was removed, cleaned, and alignment checked
Total rotor float was recorded
Clearance of rotor off nozzle was recorded
Thrust bearing was removed, inspected, and clearance recorded
Lube oil sump was drained and cleaned
Rotor was lifted and deposits removed by bathing in alkaline rust remover
Rotor was dynamically balanced
Unit was assembled and tested

b. Turbine Corrosion

Pits were found generally on cast steel surfaces; however, pitting was concentrated on the high pressure end of the turbine casing as shown in Figure 16. Measurement of the most serious pitting indicated a maximum depth of 60 mils.

Slight oxygen corrocion resembling etchings was observed on the stainless steel surfaces. Figure 17 shows this condition on the exhaust side surface of the second impulse wheel blading.

Minor erosion was observed in the form of 1/64 in. nicks on the exhaust edge of the second Curtis wheel, see Figure 17. The entire rotor and casing was conspicuously colored, ranging from a bright red at the high pressure end to a black metallic sheen at the low pressure end.

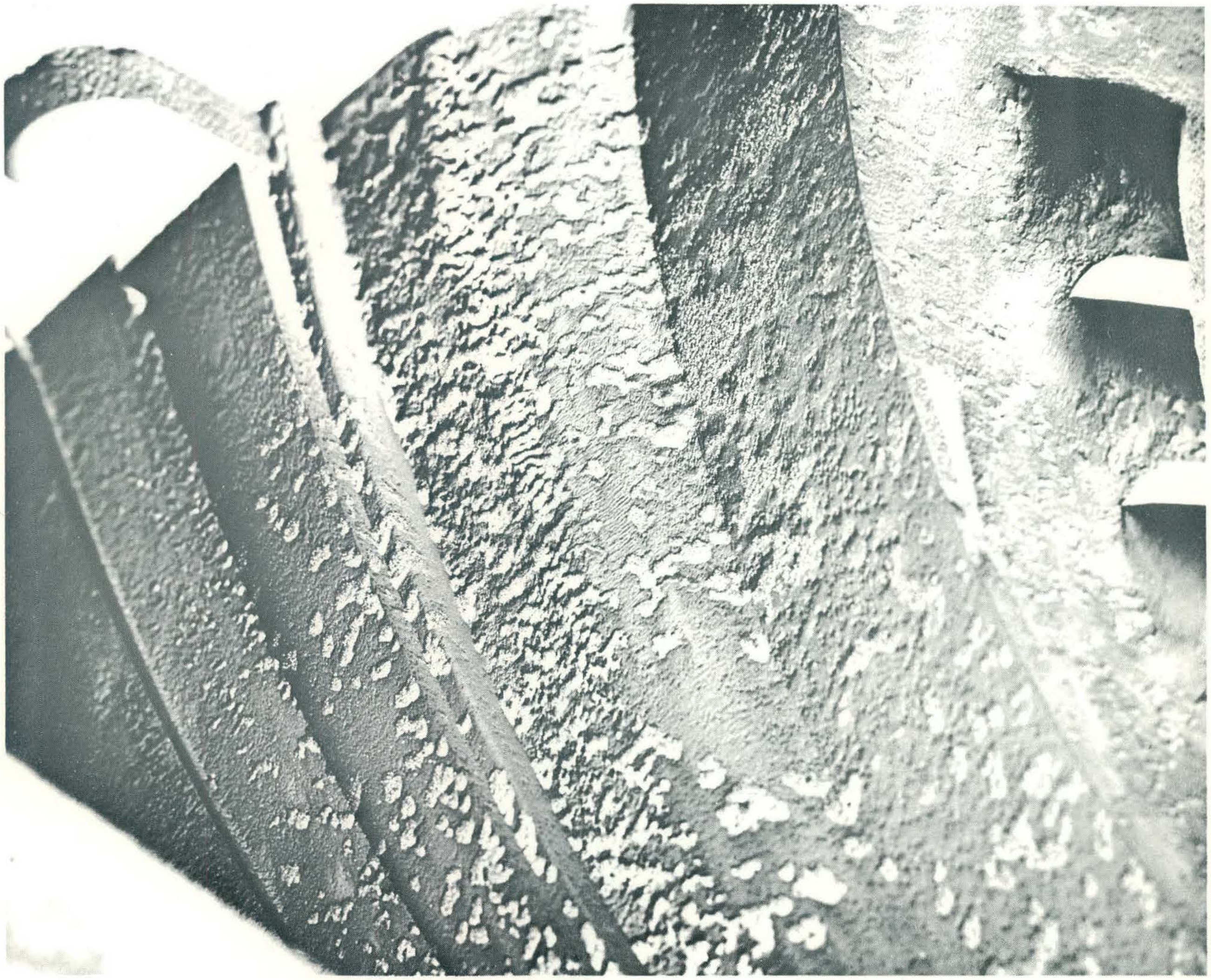
As shown on Figure 18, minor corrosion deposits were found to be built up on all shaft wheels in a radial pattern. These deposits ranged from approximately 1/32 in. on the impulse wheels to immeasurable marks on the last Rateau wheel. Corrosion deposits were also observed to have built up on the blade shrouding.

X-ray diffraction analysis of the rust deposit confirmed the material to be alpha hemitite or Fe_2O_3.

c. Activity During Direct Maintenance

The top half of the turbine casing was lifted 600 hours after reactor shutdown, and activity measurements were made to establish the nature, level, and precautions for further work. No activity was more than 20 mr/hr.

The activity of loose rust varied from 3 to 30 x 10^3 d/m/100 cm^2. Only routine health precautions were needed.

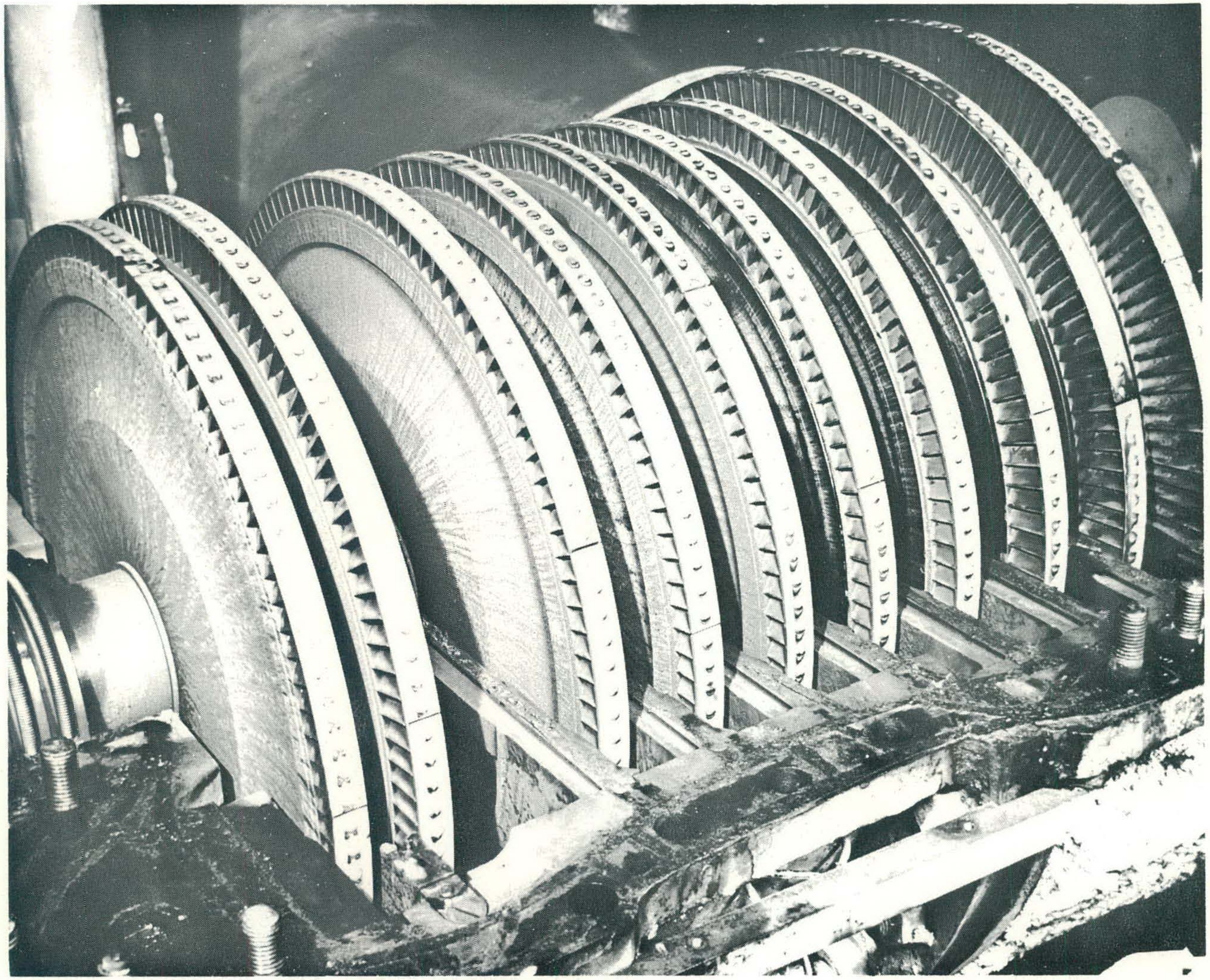


BOTTOM CASING, CURTIS STAGE

FIGURE 16

EXHAUST EDGE OF 2ND. CURTIS WHEEL BLADING

FIGURE 17



OVERALL VIEW OF ROTOR

FIGURE 18

d. Turbine Governor

Oscillations in the main steam flow were traced to faulty turbine governor operation. In two cases, turbine governor maintenance as recommended by the manufacturer was overdue, but because of sustained power operation, the maintenance had been postponed. The required maintenance was performed and operation resumed.

The manufacturer recommends changing the governor oil after 500 hours of active operation, or 1,000 hours of sustained operation. Governor preventive maintenance has been changed from every 1,000 hours to every 500 hours. If continuous operation is to be scheduled, a governor with less maintenance requirements must be specified.

2. Conclusions

Operation of the turbine and generator has been satisfactory. No excessive vibrations, temperature, or noise were observed. The excessive corrosion of the SL-1 turbine cast steel internals could be predicted based upon the abnormal stop and start operation to which the unit was subjected. A turbine governor requiring less frequent maintenance is necessary for continuous SL-1 turbine operation.

C. TURBINE LUBE OIL SYSTEM

The turbine oil system primarily consists of the following:

Main oil pump driven from the turbine shaft
Standby oil pump driven by an electric motor
Main air-to-oil heat exchanger (cooler)
Auxiliary water-to-oil heat exchanger (cooler)
Automatic oil temperature control valve at the turbine
Manual three-way valve in the oil line at the auxiliary water-to-oil cooler

The standby oil pump operates to lubricate the turbine and dissipate any heat during turbine startup and shutdown. Oil is taken from the oil reservoir tank by the oil pump and passes through the oil strainer to the oil cooler. From the oil cooler the oil is supplied to the turbine bearings at a pressure maintained by a relief valve in the line. Excess oil from the relief valve drains back to the oil tank. Oil is also supplied to the turbine gear reduction unit and governor drive shaft gears. All oil passes back to the oil tank to complete the cycle. The system has operated satisfactorily.

D. FEED WATER SYSTEM

Condensate from the main condenser and flash tank flow by gravity to the hot well, water from the hot well is pumped by either of two feedwater pumps installed in parallel to the reactor via an automatically operated feedwater regulating valve and a regenerative heat exchanger. A control for automatic change-over of feed pumps is provided in the event of a pump failure.

The feedwater system has satisfied design process requirements. Problem areas are discussed below.

1. Discussion

The original design called for the gland seals on the feed pumps to be supplied with water from the hot well. Because of oil leakage from one feed pump motor into the seal water, the seal discharge was repiped to a waste sump rather than to the hot well. Although this modification eliminated the problem of oil in the primary system, it increased plant water consumption approximately 25 gal/day. A solution to both problems is to provide a motor that would not leak and retain the original piping for seal water to the hot well.

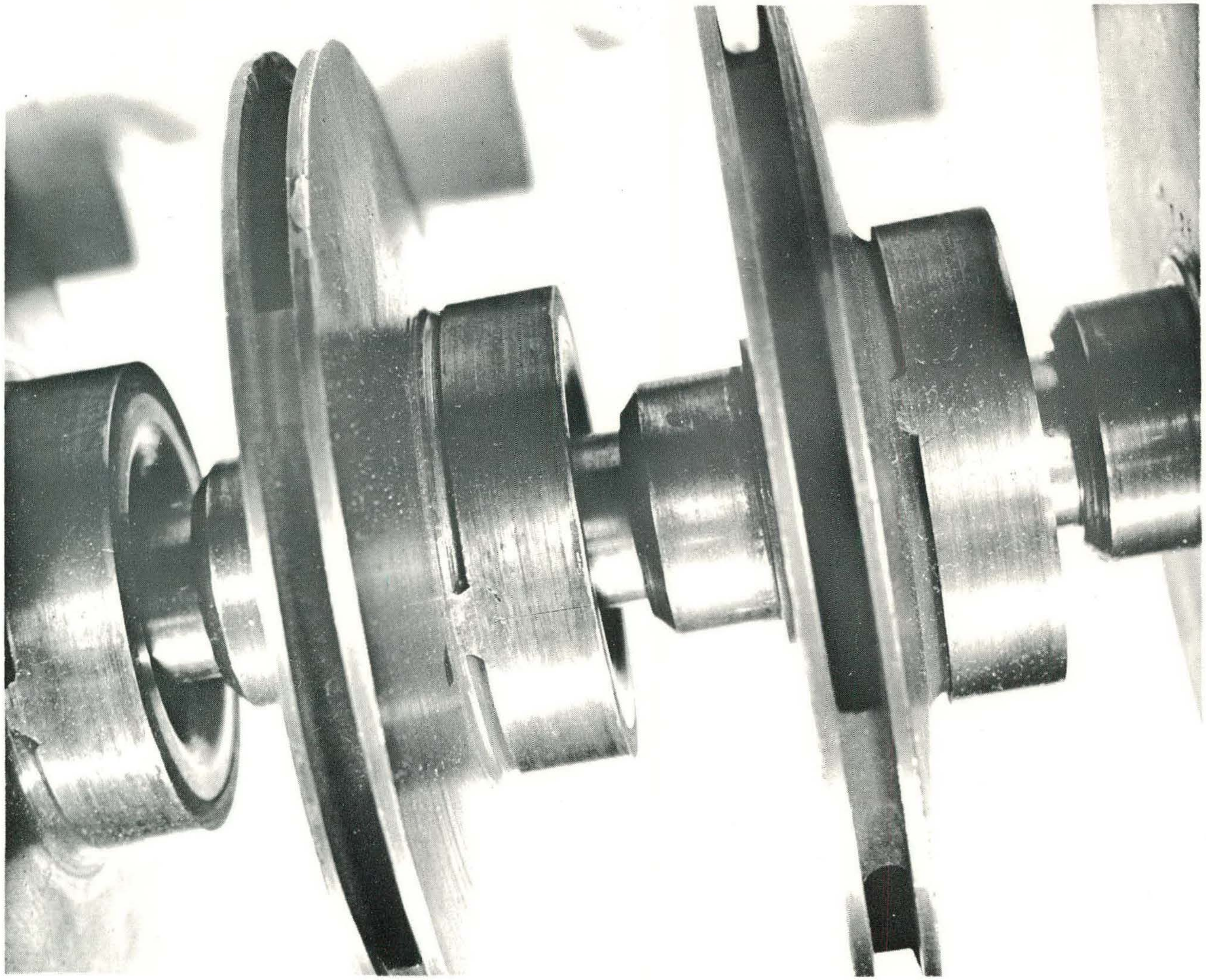
Experience indicates that the pumps are operating too close to their required net positive suction head. On numerous occasions of low pump output, high hot well temperatures or vacuums and worn suction packing glands have existed, the pumps have lost suction, become vapor bound, and seized. Because of the two-pump installation and automatic switchover circuit, plant downtime has not occurred. It is concluded that feed pumps with a greater margin between operating and shutoff head should have been selected.

Following one year's operation both feed pumps were disassembled for inspection. Normal wear of parts was found to exist in both pumps. Two abnormal conditions were also found to exist. First, examination showed the impeller wearing rings to be cracked across the tack welds as shown in Figure 19. Second, excessive impeller key wear resulted from the 700 starts and stops of the pumps since pump design allows the impellers to move axially for hydraulic balance.

SL-1 feedwater pumps are "off-the-shelf" standard units that have been subjected to an extreme operational period, the conditions found upon inspection of the pumps bear out severity of the operational cycle and necessitate the replacement of many parts.

2. Conclusions

Based on the operation of SL-1, consideration should be given to the following conclusions regarding feedwater pump installation and operation.



CRACKED IMPELLER WEARING RING

FIGURE 19

Drip-proof motors for vertical pumps are required to assure no contamination of gland seal coolant. Recirculation of feedwater gland seal water will reduce plant water consumption substantially. Pumps having locked impellers should be provided if frequent cycling operation of pumps is intended. Selection of a pump design which does not require tack welding of the wearing ring will simplify maintenance.

E. MAIN CONDENSING AND AIR EJECTORS SYSTEM

The main components in this system consist of the main air-cooled condenser, the main air ejector and condenser, the hot well and associated piping and valves. Steam exhausted from the turbine and discharged directly from the reactor and not used by the space heat exchanger passes to the main condenser where it condenses. The condensed water drains by gravity to the hot well. Noncondensable gases in the exhaust steam are removed from the main condenser by the main air ejector. The mixture of steam and noncondensables discharged by the air ejector pass to the air ejector condenser where the steam is condensed and returned to the system while the noncondensables are vented to atmosphere.

1. Discussion

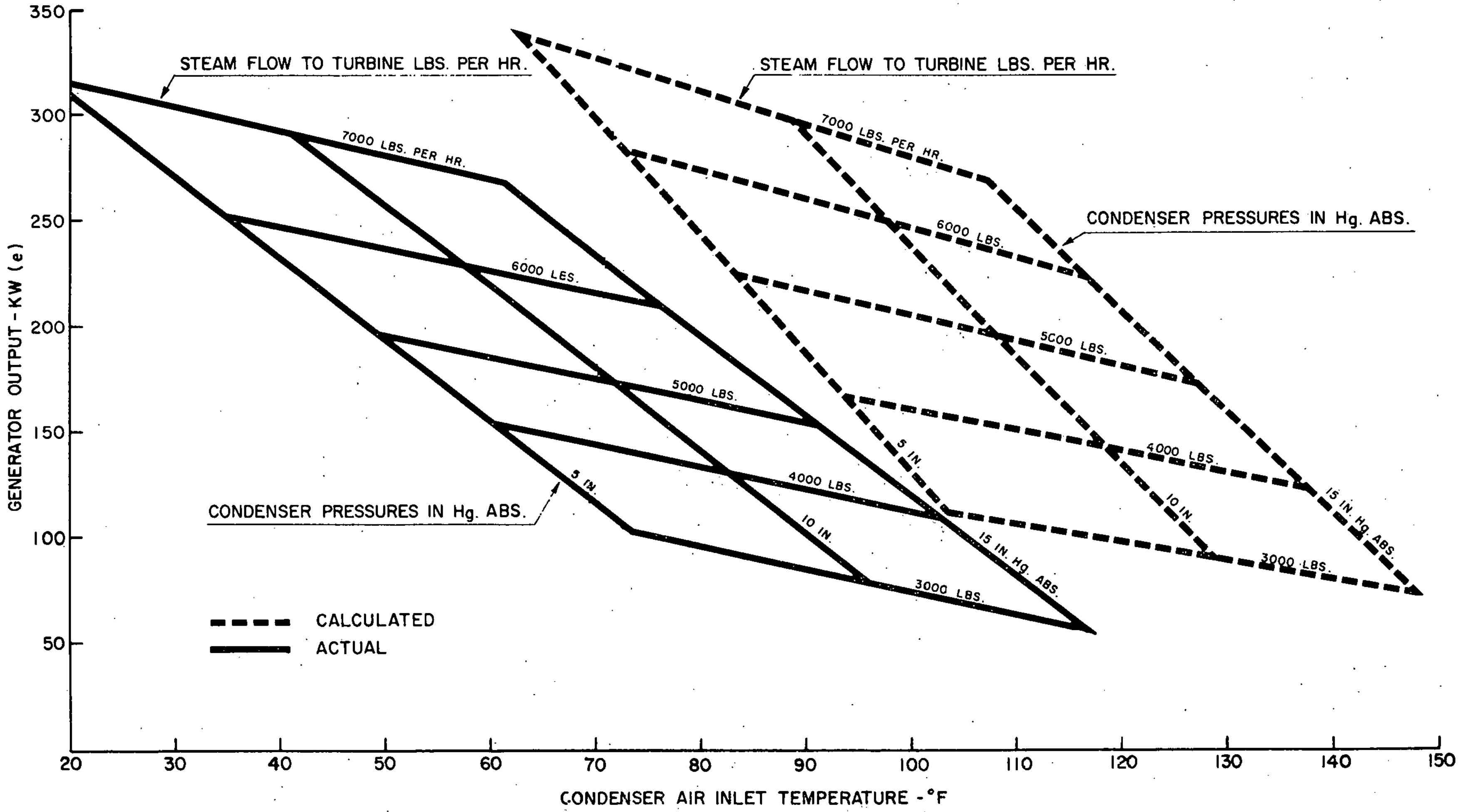
a. Condenser Performance

The main condensing system as installed at the Stationary Low Power Reactor No. 1 has operated satisfactorily. However, an evaluation of its performance indicates a 15 per cent underdesign. Testing has been done on the Stationary Low Power Reactor No. 1 condenser. This evaluation was previously reported. Performance data taken during one year of varied operation are shown in Figure 20. The dashed line was plotted from condenser design data. The solid line was plotted from actual plant performance information. If both figures are read at 7,000 lbs/hr steam flow and 10 in.Hg.abs. back pressure, a turbine generator output of slightly less than 300 KW is determined. From this point an ambient temperature of 89 ^{O}F is read from the calculated curve, and 42 ^{O}F from the actual curve.

This comparison confirms the 15 per cent underdesign of the condenser and identifies the condition that prevents the Stationary Low Power Reactor No. 1 plant from producing continuous full power during the summer months. Information from this evaluation has been published and is available for the design of future air-cooled condensers.

b. Main Condenser Fan Motor

Prior to the complete evaluation of the condenser, the only plant operational limit was a 10 in.Hg. condenser vacuum. The electric motor that drives the fan supplying circulating air to the main condenser is located on the discharge side of the condenser. Ambient temperatures in this area exceed 150 ^{O}F during hot weather and high power operation.



COMPARISON OF CALCULATED AND ACTUAL CONDENSER PERFORMANCE

FIGURE 20

The insulation in the condenser fan motor is rated for a maximum temperature of 194 °F. During a power run in a hot month, the motor failed due to insulation breakdown. Following the motor repair a thermocouple was installed through the motor housing and monitored to limit the fan motor temperature to 175 °F.

In future installations, the ambient temperature in the fan motor installation area should be considered relative to the insulation rating of the motor. No modification is intended at SL-1; however, power will be adjusted to maintain fan motor temperature below 175 °F.

c. Air Ejector Vent Flow Measurement

To determine the total amount of activity released to the NRTS, the flow through the air ejector vent must be measured. The original design called for an air rotometer installed approximately 10 feet upstream of the air ejector condenser in the vent line. Because of condensate accumulation in this line during large load changes, the installed measuring instrument could not be used and was removed to prevent its damage.

Other flow instruments have been tested in this vent line with unsatisfactory results. Portable instruments are used to obtain flow during plant operation.

To monitor effluent gases, a permanent flow meter capable of measuring air with a high moisture content should be obtained.

F. CONDENSATE COOLING SYSTEM

The condensate circulating system consists of a canned rotor pump circulating feedwater from the hot well through the shield cooling heat exchanger, the air ejectors precooler and after condenser, and back to the hot well via the flash tank. It also supplies sealing water to the main feed pump gland.

All components of this system operated satisfactorily, although hot well water at above design temperature, which was caused by main condenser performance, limited heat transfer in the systems exchangers and reduced the efficiency of the shield cooling and air ejector systems.

Although the system has operated continually without incident, any failure to a component in the system would have caused a complete plant shutdown. A standby condensate circulating pump should insure sustained operation. Inspection of the installed condensate pumps at 1500 hour intervals, as recommended by the manufacturer, has in all cases revealed wear that necessitated the installation of replacement parts. Therefore, if a single canned pump is to be used, operation must be scheduled around the 1500 hour inspection points or a pump with less maintenance requirements provided.

G. PURIFICATION SYSTEM

The SL-1 reactor water purification system consists of a canned rotor pump, heat exchanger, filter, automatic temperature limiting valve, and two resin columns, one cation, and one mixed bed. Water to five GPM is drawn from the reactor, passes through a regenerative heat exchanger where its temperature is lowered to 175 ^{o}F, then is discharged by a canned rotor pump through a temperature limiting valve that automatically closes at 175 ^{o}F to either or both cation and mixed bed resin columns. From the columns, water passes through a filter and returns to the reactor via the feedwater line. Resistivity or p^{H} of reactor water can be controlled by the available modes of operation of this system. The system contains a means to allow changing of resin, normal instrumentation, and piping for venting or priming.

1. Discussion

The current limits of quality of reactor water for reactor operations are p^{H} 6.5 to 7.5 and a resistivity $>$ than 500,000 Ω. During the past year, water quality has been maintained within these limits while various resins of different types and manufacturers have been operationally evaluated for SL-1 use. NR-14 for the mixed bed and any nuclear grade strong acid type cation resin, NR-1 or XE 77, have performed satisfactorily.

Purification water temperature entering the columns is limited to 175 ^{o}F. The purification water temperature is determined by its flow through a regenerative heat exchanger where feedwater is used as the coolant. With a feedwater flow of 9000 lbs/hr, 135 ^{o}F, this heat exchanger was designed to cool five GPM of purification water from 420 ^{o}F to 170 ^{o}F.

In actual operation, 135 ^{o}F feedwater corresponding to a 5.16 in. Hg. absolute pressure in the main condenser is not available due to condenser under design and feedwater temperature is normally between 150 ^{o}F to 160 ^{o}F. With this feedwater temperature, it is necessary to reduce purification water flow to below two GPM to maintain its temperature below 175 ^{o}F.

2. Conclusions

In order to maintain reactor water quality during normal operation and during a planned power extrapolation experiment, five GPM purification water flow is necessary. The original SL-1 design is incapable of this flow.

In order to attain five GPM purification system flow at any power level, an additional heat exchanger has been installed.

Although the design layout is overly complex and inaccessible for ease of operation, the system as revised is capable of accomplishing its designed purpose and is satisfactory for SL-1 operation.

H. PLANT MAKE-UP WATER

Well water which is stored in a 50,000 gal. tank is pumped through a diatomaceous earth oil filter. The demineralizer contains 1.5 cubic feet of low temperature mixed bed resin, where its resistivity is increased to greater than 5000,000 Ω and into a 1,000 gal. plant make-up water storage tank. Makeup water is also circulated through the shield cooling system and simulated space heat system and the hotwell, or vacuum drag to compensate for system losses. The raw water can be discharged to the turbine lube oil cooler, purification shutdown cooler, steam sample coolers, or the raw water demineralizer. The makeup water system operates satisfactorily.

SL-1 operating experience has established a makeup water rate of 50 gallons per day while at steady operation. Water consumption is shown in Figure 21.

Although the SL-1 plant water consumption is acceptable, improvements can be made.

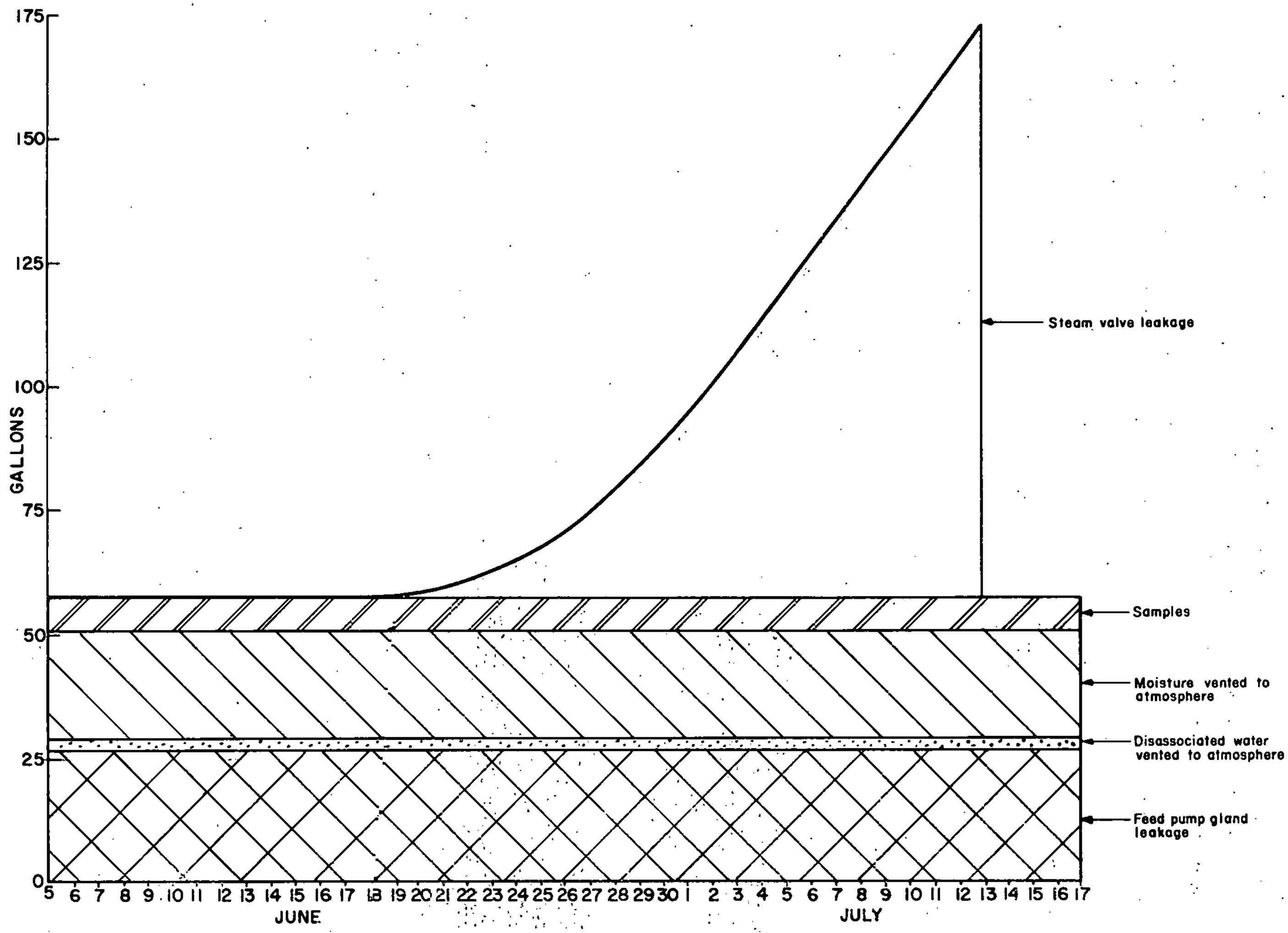
The makeup resin column has a capacity for demineralizing approximately 1,000 gallons of raw water with normal consumption; it is therefore necessary to replace resin every three weeks since no provision for resin regeneration at SL-1 exists. The addition of a zeolite column to increase ion exchange capacity is planned.

I. SIMULATED HEAT SYSTEM

To simulate the use of hot water for space heating at a remote location, main steam at 40 psig is regulated to a simulated heat exchanger where it is used to heat a circulating mixture of ethylene glycol and water. The heated mixture is then pumped through two fin fan exchangers, where the heat is dumped to the atmosphere. With a circulating flow of 135 GPM, the system is capable of dissipating 400 KW_{th}/hr.

This system has functioned as designed. All mechanical components operated satisfactorily.

The simulated space heat system is capable of continually removing its designed 400 KW/hr. A recommended modification would be the installation of a power totalizer. This installation would relieve the operator of making an hourly calculation and provide continuous monitoring and recording of system output at a small expense.



WATER CONSUMPTION DURING 1000 HOUR TEST
FIGURE 21

J. FUEL ELEMENT STORAGE SYSTEM COOLING

The fuel element storage cooling system is used to remove heat from the fuel storage wells that has been produced by spent fuel elements stored there. A small secondary source of heat will be from the shielding gravel around the wells. The heat is dissipated by vapor condensation through a one-pipe system to a condenser on the fan floor. One fuel well has been used to store two hot fuel elements.

Remote water level indication for these wells is needed. The present procedure calls for lifting each well top plug and measuring level with a sounding line. With the containment of a full core charge and with limited personnel, the installation of a permanent float-type level indicating device would provide more positive, continuous assurance of level.

K. MISCELLANEOUS DRAINS AND VAPOR LEAK-OFF SYSTEM

The miscellaneous drains and vapor leak-off systems are two separate plant water and vapor collection systems which return condensed water to the reactor via the flash tank. Each system consists of a tank, pump, and float switch.

One system designated the Low Pressure Condensate Return System operates at condenser vacuum since it is vented directly to the condenser. It collects condensate and vapor from the turbine exhaust piping, the turbine seal glands and turbine drains. It returns the water to the flash tank by means of a pump and the vapor vents to the turbine exhaust line.

The other system, designated the High Pressure Condensate Return System, operated at approximately three inches Hg. below atmospheric pressure. It collects condensate from the pressure vessel leak-off groove between the gaskets, the air ejector after-condenser and vapor precooler, and also receives the reactor control rod drive water seal effluent. The water collected is returned to the flash tank by use of a steam trap. For capacities beyond the steam trap, a pump is automatically started. The vapor is vented to the vapor precooler. This system has operated satisfactorily since installation.

L. REACTOR BUILDING, HEATING, AND VENTILATION

For the purpose of this description, the system will be considered in two parts; first, that used when the reactor plant is secured, and second, when the reactor is operating. When secured, the only heat supplied to the reactor building operating and fan floor is from four space heat exchangers that are supplied hot water from a separate furnace located in the support facilities building. Ventilation is by a 1.5 hp blower taking suction from the main condenser fan intake duct and discharging to two locations on the operating floor.

While at power, heating and ventilation is the same as when shutdown with the additional provision that the ventilation blower suction is thermostatically controlled to draw from the main condenser fan suction or discharge, depending on operating floor temperature.

1. Discussion

Performance of the reactor building heating and ventilation system is inadequate. In below freezing weather and during plant shutdown, the four space heaters are inadequate, which results in the freezing of equipment and temperatures that prohibit normal accessibility by personnel. In weather above 60 °F during plant operation, ventilation is inadequate, which causes thermal overload trips of electrical equipment, and inconveniences operating personnel.

Building insulation and structure should be improved. Additional space heaters should be installed on the operating and fan floor for shutdown heating.

The reactor building ventilation system consists of a 1.5 hp blower which takes suction from either the main condenser fan suction or discharge, depending on the position of the blower intake louvers, and discharges to two locations on the operating floor. The intake louvers are controlled by a thermostat located on the operating floor. Initially it was discovered that the blower would not discharge its rated capacity when the main condenser fan was operating above half speed. By adjustment of the V-pulley on the motor belt drive, maximum blower RPM was reached and an air flow to the operating floor reached design regardless of main condenser fan speed.

The discharged air was found to be heating rather than ventilating the operating floor. Although the thermostatically controlled blowers from the discharge of the main condenser fan were shut, the differential pressure across the louvers was enough to cause sufficient leakage of hot air. To prevent this, the intake from the main condenser fan discharge was sealed shut.

The ventilation system was found inadequate in size to maintain the operating floor at a reasonable temperature during the summer months. It was necessary to operate with the cargo hatch and emergency exit door open. The ambient temperatures on the operating floor still ranged above 100 °F. Portable fans were positioned to cool various electrical components to allow their continuous operation.

The ventilation system meets design specifications with respect to flow, although the system is not capable of properly ventilating the operating floor when outside temperatures exceed 60 °F. The ability of the system to ventilate with warm air in the winter is unnecessary. Additional ventilating capacity should be installed for summer months.

V ELECTRICAL SYSTEMS EVALUATION

A. SIMULATED ELECTRICAL LOAD SYSTEM

The simulated electrical load consists of the following:

1. Three 60 KW three phase units, each of which is controlled by a 225 ampere circuit breaker located on section three of the main distribution panel, and permanently connected to the utility bus.

2. Four 10 KW three phase units, each of which is controlled by a 40 ampere circuit breaker located on the lower portion of section four of the main distribution panel, and permanently connected to the utility bus. These circuit breakers are provided with overcurrent trips only.

3. Seven 10 KW three phase units, each of which is controlled by a 40 ampere circuit breaker located in the center portion of section four of the main distribution panel. These circuit breakers are provided with overcurrent trips only.

This group of load units may be fed from the utility bus by throwing the operating handle of the load transfer switch to its upper position. The load units may be fed from the equipment bus by throwing the switch to its lower position. In the center position of this switch no connection is made to either bus. The performance of this system has been in accordance with design.

B. MAIN SWITCH GEAR

This system includes normal breakers, switches, and instrumentation to connect the diesel generator, turbine generator, and Idaho power to either or both electrical buses, namely, the utility and equipment bus, and to control their operation. It also includes the switches, breakers, and controls to distribute electrical power to the various distribution panels through the facility.

Commercial quality power has, therefore, been provided by both buses. Although this system has operated satisfactorily, erratic tripping of the motor control center breaker has been caused by high ambient temperatures in the motor control cabinet.

In the original design, provision was made for two electrical buses. The utility bus connected all power plant loads to the various electrical circuits through normal switching. The purpose of the equipment bus was to supply non-power plant equipment such as radar. The equipment bus received its power from the utility bus through a bus tie breaker. Provision was made between the buses for the installation of voltage compensating transformers; however, these were not installed.

To assist in plant startup, an additional electrical tachometer indicator was installed at the turbine control station. Tachometers are now installed in the reactor control room and the turbine control station.

C. ELECTRICAL AUXILIARIES

The electrical auxiliaries include all motors and their controllers in the SL-1 plant. Also included is the plant lighting system.

All electrical auxiliaries have performed satisfactorily and in accordance with design.

Based upon SL-1 experience, consideration should be given to the location of motors and controllers with regard to their expected ambient temperature. The SL-1 main condenser fan motor experienced insulation breakdown and subsequent failure while operating in a temperature exceeding its insulation rating. The controller for the plant ventilation fan required relocating due to high ambient temperatures causing continued thermal overload trips.

The installation of electrical motors containing a lubrication system that leaks should be eliminated, or installed in such a manner as to prevent the contamination of primary water.

D. EMERGENCY POWER SUPPLY

The emergency power supply is installed to supply emergency power for reactor control upon loss of normal control power. It consists of a three-unit motor-generator set and its associated control panel. The motor-generator set consists, in turn, of a synchronous alternator and a direct current motor-generator. Under normal conditions the synchronous motor drives the alternator to supply certain instruments, the rod drive motors, and the instrument well exhaust fan, and the direct current machine, which acts as a generator and maintains the charge on the station battery. Upon failure of normal alternating current supply voltage, the set is driven by the direct current machine, acting as a motor, which derives energy from the station battery. Once the unit is started, the necessary switching and contact actions automatically follow a failure of normal power

The emergency power supply has operated satisfactorily since installation. It has proved capable of supplying emergency power for reactor control upon loss of normal power for more than five hours.

E. DIESEL GENERATOR UNIT

One 60 KW diesel generator unit is installed at the SL-1 facility and wired through a circuit breaker and switch to the utility and equipment bus. The unit serves as an emergency power supply upon loss of Idaho Power.

The unit has operated satisfactorily since installation, however, it is not of adequate size to allow cold startup of the SL-1 plant.

The unit as installed requires three men to parallel its power with the SL-1 buses due to the physical location of its controls. In future installations, the unit's electrical control should be repositioned to the main electrical control board.

The generator's main circuit break is not of sufficient capacity to allow starting of the main condenser fan motor, due to the motor's required high starting amperage. Additional capacity should be installed in future installations to allow a complete cold startup.

THIS PAGE

WAS INTENTIONALLY

LEFT BLANK

VI FACILITY SYSTEMS EVALUATION

A. DEEP WELL, PUMP AND STORAGE SYSTEM

The water supply for SL-1 is from a 50,000 gallon water storage tank. An operating level in the tank is maintained between 27,000 and 35,000 gallons by a pressure operated valve that automatically starts and stops a deep well pump. This well and pump also serve a 75,000 gallon storage tank for the Army Hot Cell Area. Low tank level alarms are connected in the guard shack and NRTS Central Fire Headquarters.

Difficulty initially was experienced with the flow control system since no level indicating device was installed on the storage tank. A line indicator was subsequently installed and the system functions satisfactorily.

The pump installed at SL-1 requires its shaft bearing to be lubricated with oil. This oil drips into the water supplied and has necessitated the installation of a diatomaceous earth oil filter. Future installation should eliminate the use of oil lubricated water pump shaft bearing because sources of oil contamination in the primary system should be avoided.

B. DOMESTIC AND FIRE WATER SYSTEM

A single domestic water pump taking suction from the 50,000 gallon water storage tank maintains 40 psig water pressure in the domestic system and on the SL-1 fire hydrants. With loss of 40 psig in this system a 150 psig fire pump automatically starts. Water supplied for domestic use is chlorinated in a holdup tank. Excess domestic water is recirculated to the storage tank to prevent freezing.

A 40 psig regulating valve to protect low pressure domestic piping from fire pump pressure was originally installed upstream of the chlorination tank. This caused the downstream tank to burst when subjected to the 150 psig discharge pressure of the fire pump. The regulating valve was subsequently relocated to the upstream side of the tank and the tank was repaired. This system has since performed satisfactorily.

C. PLANT WATER SUPPLY SYSTEM

One pump taking suction from the 50,000 gallon storage tank discharges water through a diatomaceous earth oil filter to the operating floor. Controls for this pump are on the operating floor. The water passes through a mixed bed resin volume to the 1,000 gallon make-up storage tank as needed or is used for lube oil cooling, steam sample coolers, purification shutdown coolers, and makeup to the shield cooling and simulated heat load surge tanks. Excess or used water drains by gravity to a desert ditch.

This system has operated satisfactorily except for the drainage system which freezes during the winter months.

Drain piping normally cannot be used all winter due to freezing. Although temporary modifications using electric heating tape and insulation alleviate the situation, piping should be modified to be sufficiently underground to prevent freezing. The drainage ditch should be revised to include an underground dry well.

D. CONTAMINATED LIQUID WASTE DISPOSAL SYSTEM

A liquid waste collection system provides:

1. Drainage from the purification system, mainsteam relief valves, etc., that are radioactive, by gravity to a contaminated waste storage tank located in the gravel in the reactor building.

2. Collection of contaminated water from the chemistry laboratory and low level lay down area in a 500 gallon retention tank outside the building.

3. Drainage of reactor floor, feed pump gland leakage, and purification vault, to two waste storage sumps located in the reactor building gravel.

4. Drainage from the decontamination building to a retention tank outside the building.

No difficulty has been experienced with this system.

Removal of contamination waste from the several retention tanks is accomplished by a portable pump rig. The rig is moved and attached to each of the four tanks for waste removal.

As operating time is accumulated on the SL-1 plant, the concentration of radioactive liquids is expected to increase. The present portable rig does not adequately provide for sampling, contamination control, and dilution. A more permanent piping system from each tank to a manifold, sampling point, and dilution line, is being planned.

E. PLANT FACILITIES

This section includes the support facilities and reactor building general construction, heating, and ventilation for the support facilities, and yard piping and drains.

The reactor building heating and ventilation system has been described earlier. A contributing factor to the inadequacy is the construction of the reactor building itself. The cargo doors located on the operating floor are not tight fitting and result in a heating problem in the winter months. An inadequate ventilation system necessitates operating with the operating doors open in the summer to protect equipment from high temperature. This introduces a dust and dirt problem on the operating floor.

The support facilities building heating system is by hot air. Only one thermostat is installed, consequently, temperature control is poor.

All plant drains and vents to atmosphere are problems in the winter due to freezing. Extensive use of temporary heating elements is required to maintain plant operation.

To summarize the SL-1 building construction, heating, and ventilation systems, and outside drain and vents are not satisfactory for the -30 °F to +100 °F temperatures encountered at NRTS. In future installations, more consideration should be given to these problems and the experience gained at SL-1 applied.

THIS PAGE

WAS INTENTIONALLY

LEFT BLANK

VII HEALTH AND SAFETY EVALUATION

The radiation safety evaluation of one year of operation of the SL-1 plant is significant since the facility has served as an operation prototype for PL plants. The SL-1 plant has operated with full access to all plant machinery. The plant has been operated by military crews and with a source of low level fission products in the system. Experience in removal of irradiated fuel elements, replacement of spent demineralizer resins, removing and cleaning control rod drive mechanisms, and replacing the reactor vessel head gaskets, has been obtained. Data on gaseous, liquid, and solid contaminated wastes have been accumulated. Direct maintenance of the turbine, feed pumps, valves and other components has been demonstrated.

During this period, no over exposures to personnel have resulted.

A. SHIELDING EFFECTIVENESS

Leakage through the 16 foot radius gravel shield and shield tank is well within tolerance. Radiation through the concrete top shield to the operating floor is negligible. Leakage under the reactor makes the area under the shield tank inaccessible during reactor operation.

B. PLANT RADIATION LEVELS

The operating floor was designed as an accessible area during reactor operation. A continuous radiation survey is conducted to assure safe levels of radiation. Table VI presents radiation data for personnel and equipment locations on the plant operating floor.

The highest activity level in the simulated load heat exchanger traps was reduced by venting N^{16} to the stack. The activity over the ion exchange columns at the end of resin life were reduced by location of lead shielding over the purification vault cover.

No serious radiation sources exist in the SL-1 plant during full power operation. Further, full access to the operating floor during operation is routine practice.

TABLE VI

SL-1 RADIATION LEVELS DURING FULL POWER OPERATION

Location	mr/hour			
	Dec. '58	May '59	July '59	Nov. '59
Lower Landing of Stairway to Operating Floor	0.1	0.6	0.7	0.7
Between Lower and Middle Landing	0.1	0.1	0.1	0.1
Doorway, Operating Floor Level	0.8	0.9	0.9	1.0
General Operating Floor	-	3.0	3.0	4.0
Middle, Turbine Control Panel	5.0	6.0	6.0	7.0
Purification Panel Near Valve No. 8	-	17.0	24.0	31.0
Floor Level Over Purification Vault	-	65.0	125.0	150.0
Floor Over Purification Vault After Addition of Lead	-	-	-	20.0
Operating Floor Between Purification Panel and Shield Block	-	10.0	12.0	15.0
Feedwater Pump	1.0	1.0	1.0	1.4
Feedwater Filter	-	12.0	-	-
Main Steam Line Near Reactor	30.0	20.0	30.0	30.0
Turbine casing	4.0	6.0	6.0	11.0
Main Steam Line Near Turbine	5.0	8.0	8.0	8.0
Condenser	-	-	-	45.0
Hotwell	-	3.0	3.0	8.0
Simulated Heat Load Heat Exchanger Trap	300 max.	28.0	30.0	40.0

C. AIR ACTIVITY LEVELS

An air monitoring system is installed in all plant areas which may be sources of radioactivity. Experience at the SL-1 plant indicates an activity level close to background during normal reactor operation. Steam leaks from valves and instrument lines have periodically resulted in some air contamination. The most severe occurrence was attributed to a malfunction of the main steam isolation valve in which a concentration of 0.2 $\mu c/cc$ was measured. For comparison the maximum permissible level for the measured isotopes is 2×10^{-4} $\mu c/cc$. Standard health physics procedures required respiratory protection for personnel, thus no inhalation of activity occurred. The maximum whole body exposure was less than 35 mrem.

SL-1 experience has indicated no significant air activity problem.

D. PERSONNEL EXPOSURES

Radiation exposure records of SL-1 operating and maintenance personnel were maintained for the first year of plant operation. These data are summarized in Figure 22.

The maximum accumulated whole body exposure for the period is 215 mrem as compared to an annual maximum permissible dose of five rem. As part of the military personnel monitoring program, two crew members were returned to Walter Reed Hospital for a second year for whole body counts. Tests revealed no significant increase in body burden.

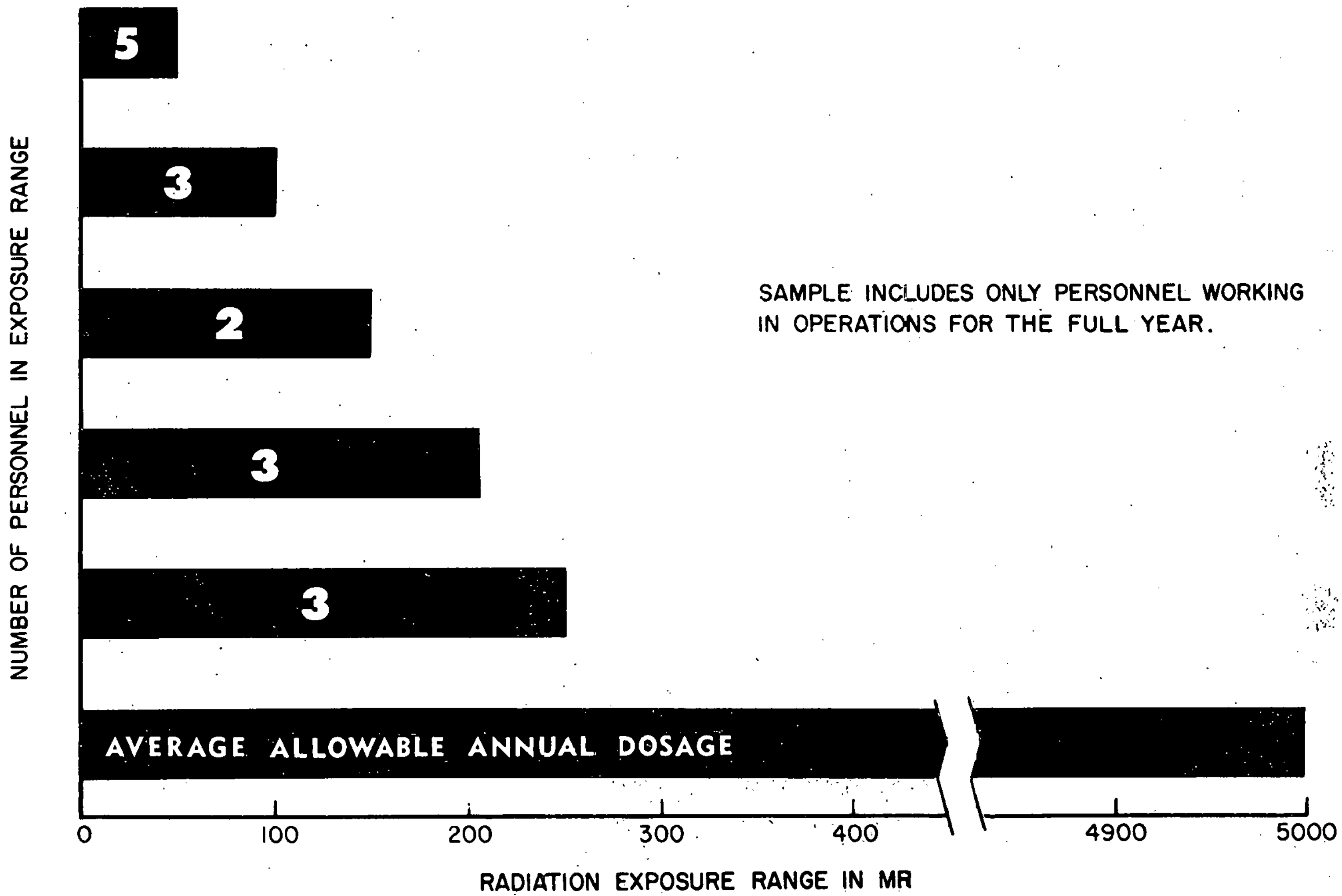
E. SPECIAL RADIATION HAZARDS

1. Resin Removal

Resin removal is accomplished at SL-1 by attaching a portable pump, pipe, and disposable resin flask. The flask is placed in a water filled retention tank for shielding. The pump and transfer lines are located on the operating floor. Although the radiation field is from 1 to 40 R/hr from the portable rig, personnel exposures have been kept well below tolerance. An improved apparatus is being fabricated which will reduce the radiation field.

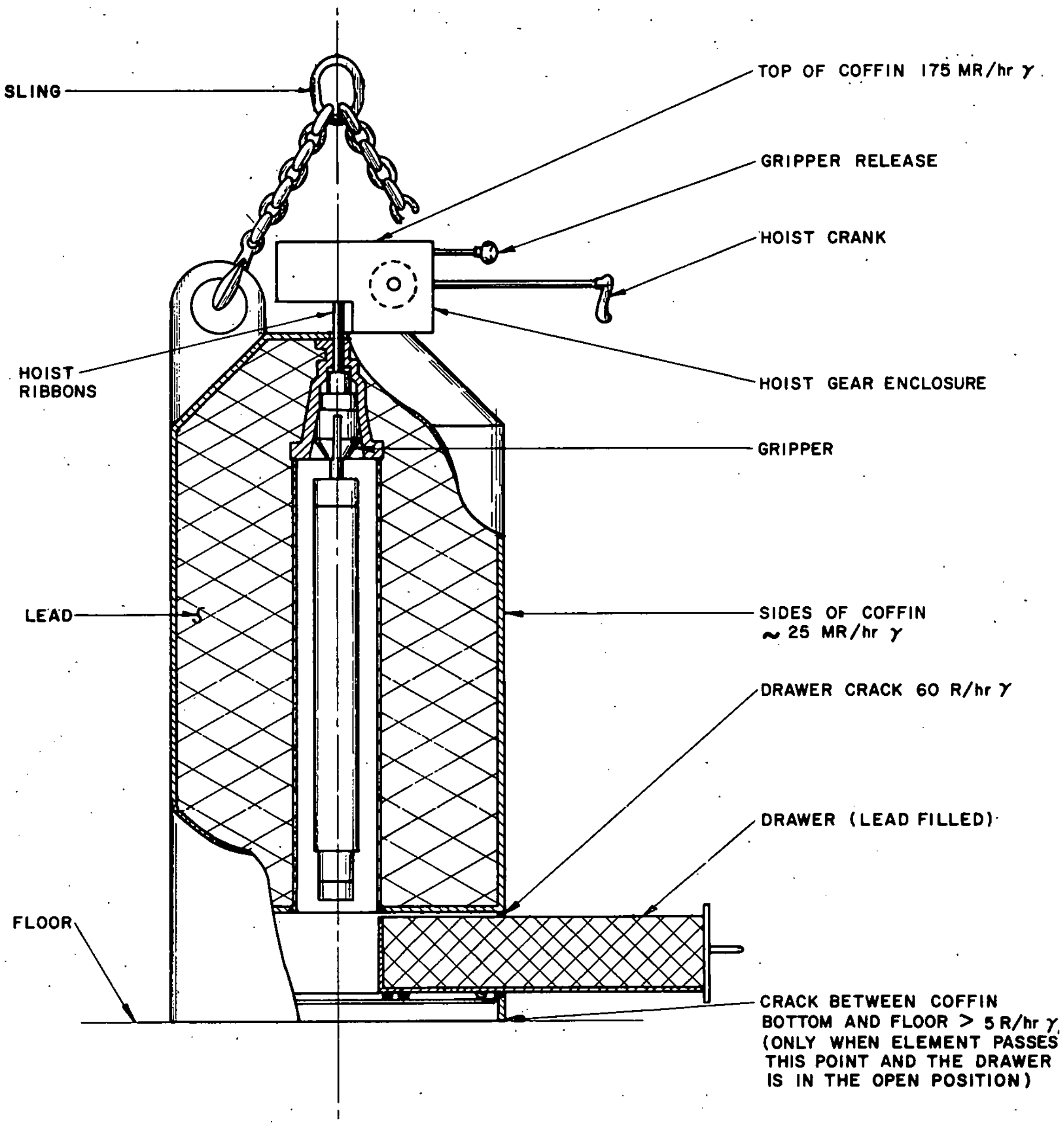
2. Fuel Element Transfer

The removal of a fuel element from the SL-1 core is performed by pulling the element through an opening in the vessel head and into a cask. The element is raised or lowered by a winch mounted on top of the cask, see Figure 23.



SL-1 PERSONNEL EXPOSURE SUMMARY FOR 1959

FIGURE 22



FUEL ELEMENT TRANSFER COFFIN DOSE RATE EVALUATION

FIGURE 23

A transfer demonstration of a fuel element following 230 MWD service and a three-day decay period was accomplished and radiation fields recorded.

The maximum level recorded was leakage between the operating floor and the bottom of the cask totaling $<$ 160 R/hr gamma. Leakage around the coffin door was 60 R/hr, and through the cask lead walls 25 mr/hr. Special precautions limited personnel exposures to 35 mrem during the transfer.

Since the fuel element was exposed to only 15 per cent of life, it is concluded that the transfer cask will not provide adequate shielding for end of life fuel elements. A redesign transfer cask will be obtained.

F. WASTE DISPOSAL

1. Air Ejector Effluent Release

A continuous monitoring effort was conducted to measure activity release and identification from the plant air ejectors. The conclusions from this study are as follows:

a. 0.4 curie/MWD off-gas release to the environment has been determined by computing air ejector flow (3.14) with total activity at T = 0 for 5,000 lb/hr main steam or 1.5 megawatts.

b. There has been no significant increase of effluent released to the environment during this report period.

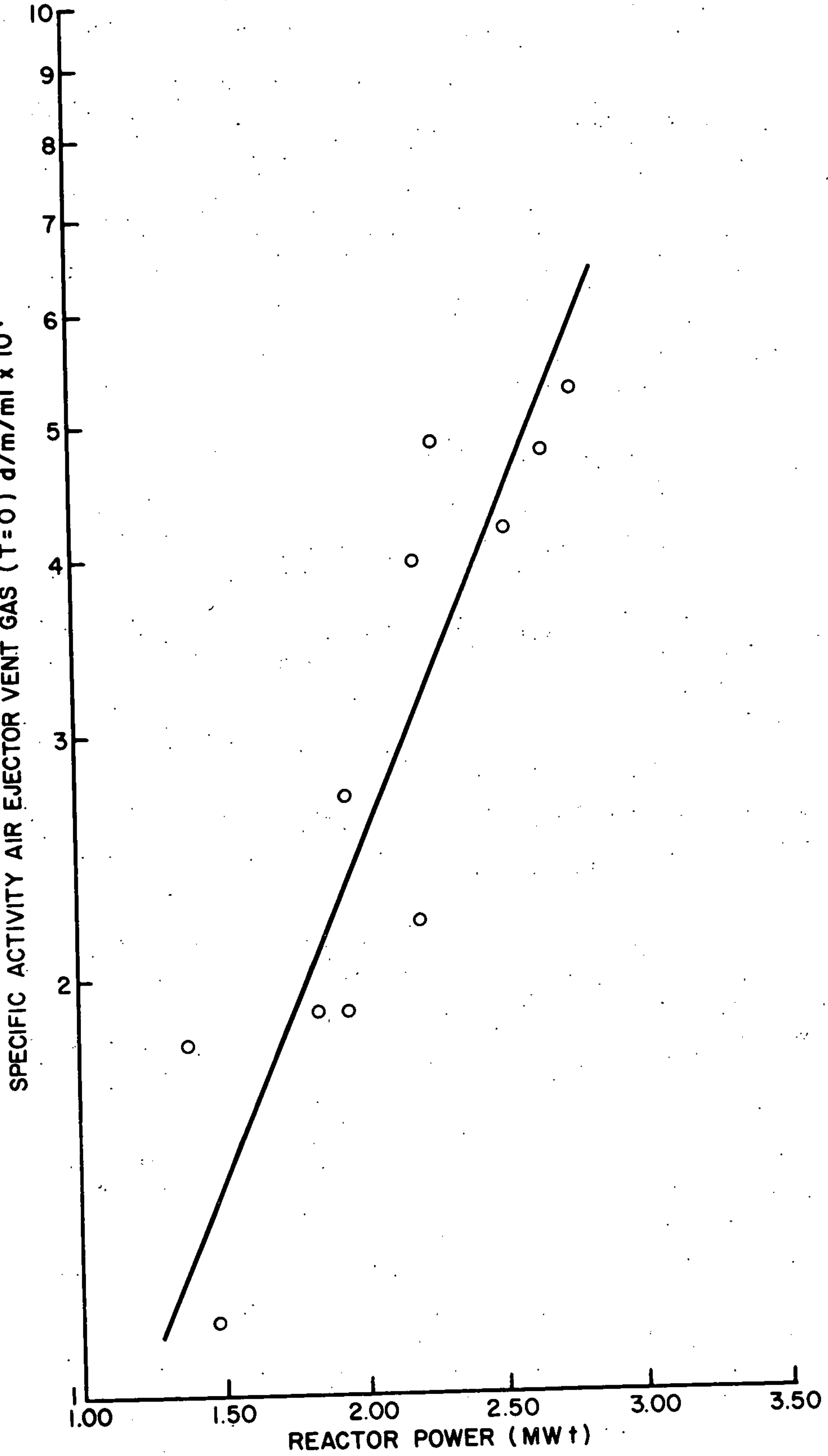
c. Off-gas specific activities have been identified as Xe^{138}, Xe^{135M}, Xe^{133}, Kr^{88}, and Kr^{85}. The composition of total activity may be noted in Table VII.

d. An MPC_a for off-gas has been computed to be 5×10^{-5} $\mu c/cc$. This calculation was based on the relative percentages of specific activity as noted in paragraph "c" of this section.

e. An average of results of samples taken at specific power outputs have been plotted as illustrated in Figure 24, to determine specific activity as a function of reactor power.

f. A total release to the environment for the period January 1, 1959, through December 31, 1959, has been computed to be 166.31 curies total activity for a total of 303.13 MWD.

g. Calculations using Sutton's Diffusion Methods were made to determine a maximum point of concentration downwind assuming an average wind velocity of 10 mph and a strong lapse rate condition. The maximum point of concentration downwind was estimated to be 70 meters. At this point a concentration of 1.5×10^{-8} $\mu c/cc$ was computed as compared to the MPC_a of 5×10^{-5} $\mu c/cc$. At no time during this annual report period has off-gas release precluded safe operation of the SL-1 plant.



SPECIFIC ACTIVITY SL-1 AIR EJECTOR VENT GAS

FIGURE 24

TABLE VII

SL-1 STACK EFFLUENT RELEASE TO ENVIRONMENT-1959

Month	MWD	Form	Total Activity Curies/Month	Identification
January	8.14	Gaseous	3.25	Xe^{138}, Xe^{135}, Xe^{135M} Xe^{133}, Kr^{88}, Kr^{85M}
February	None			
March	None			
April	11.30	Gaseous	4.25	Xe^{138}, Xe^{135}, Xe^{135M} Xe^{133}, Kr^{88}, Kr^{85M}
May	20.82	Gaseous	17.41	Xe^{138}, Xe^{135}, Xe^{135M} Xe^{133}, Kr^{88}, Kr^{85M}
June	59.74	Gaseous	23.89	Xe^{138}, Xe^{135}, Xe^{135M} Xe^{133}, Kr^{88}, Kr^{85M}
July	39.30	Gaseous	15.72	Xe^{138}, Xe^{135}, Xe^{135M} Xe^{133}, Kr^{88}, Kr^{85M}
August	0.33	Gaseous	0.26	Xe^{138}, Xe^{135}, Xe^{135M} Xe^{133}, Kr^{88}, Kr^{85M}
September	23.25	Gaseous	28.16	Xe^{138}, Xe^{135}, Xe^{135M} Xe^{133}, Kr^{88}, Kr^{85M}
October	55.16	Gaseous	22.06	Xe^{138}, Xe^{135}, Xe^{135M} Xe^{133}, Kr^{88}, Kr^{85M}
November	44.01	Gaseous	26.40	Xe^{138}, Xe^{135}, Xe^{135M} Xe^{133}, Kr^{88}, Kr^{85M}
December	41.08	Gaseous	24.64	Xe^{138}, Xe^{135}, Xe^{135M} Xe^{133}, Kr^{88}, Kr^{85M}
Total	303.13		166.31	

2. Liquid Waste

All liquid waste resulting from SL-1 operations was collected in retention tanks for sampling, measurement, and subsequent discharge to a leaching bed. The low level contaminated waste resulted from gland seal leakage, equipment and plant decontamination activities during maintenance work, spillage, water used to transfer spent ion exchange resins, and laboratory drains.

Table VIII presents a liquid waste chronology for the year 1959. The greater than average discharge in February, 1959, resulted from flushing of the entire system to remove oil caused by leakage from the deep well pump. This pump is not a part of the reactor system. The principal activity in the liquid waste for the annual period as reported was Na^{24}. The low activity level is attributed to several days storage of this waste before sampling and discharge to the leaching bed.

Liquid wastes were stored for several days in retention tanks within the plant gravel shield prior to discharge, thus permitting the decay of short-lived products.

Based upon SL-1 experience liquid waste produced was held to an average of 40 gals/day under all conditions. For periods of normal operation, the average was 30 gals/day.

3. Solid Waste Disposal

Solid wastes resulting from SL-1 operation is in the form of contaminated paper, rags, filters, and spent ion exchange resins. A total of 104 cubic feet of solid waste containing 27 μc (principally Na^{24}) and weighing 950 lbs was shipped to the NRTS burial ground.

G. FACILITY AND EQUIPMENT MODIFICATIONS AND ADDITIONS

To improve the health and safety conditions and techniques at SL-1, several modifications and additions to facilities and equipment were made.

1. Buffer Area

A room adjacent to the reactor building was modified to confine contamination to the support facility. The room includes a shower, change booth, equipment locker, survey equipment rack, hand and foot counter, and waste and clothing containers.

2. Constant Air Monitor

The constant air monitor collection system was modified to permit simultaneous or individual air monitoring of the operating floor and/or fan floor.

TABLE VIII

SL-1 RADIOACTIVE LIQUID WASTE SUMMARY - 1959*

Month	Volume (gallons)	Total Activity μc	Identification (Specific Activity)
January	-	-	-
February	5,000	150	Na^{24} (99%), I^{131}, I^{133}
March	-	-	-
April	-	-	-
May	800	120 **(Sr^{90} $1.3 \times 10^{-6} \mu c/cc$)	Na^{24}, Cd^{115}, I^{131}, I^{133} Xe^{133}, Sr^{90}
June	-	-	-
July	800	18.4	Na^{24} (99%)
August	1,900	17.07	Na^{24}, Cd^{115}, I^{131}
September	800	60 **(Sr^{90} $1.2 \times 10^{-6} \mu c/cc$)	Na^{24}, Mn^{56}, Sr^{90}
October	700	0.1	Na^{24} (99%)
November	1,250	62.2	Na^{24} (99%)
December	1,623	16.7 **(Sr^{90} $6.9 \times 10^{-7} \mu c/cc$)	Na^{24}, I^{131}, Sr^{90}
TOTALS	12,873	244.47	

* Prior to dilution and discharge to leaching bed

** Denotes Sr^{90} $\mu c/cc$ concentration of the total activity as listed for the month

3. Eberline Portal Radiation Monitor

An Eberline Portal Radiation Monitor was installed in the guard house for monitoring of personnel before exiting from the SL-1 site.

4. Vacuum Flask

A two-liter vacuum flask equipped with a 1B85 Victoreen counting tube was modified to improve spot sampling of the air ejector off-gases for total activity. A standardized US6 vacuum gauge was adapted to the chamber to measure vacuum. The chamber was redesigned for ease of decontamination. A millipore filter trap was added to catch particulates. The sampler is shown in Figure 25.

5. Hand and Foot Counter

Lead shadow shielding was installed in the hand and foot counter to reduce background activity seen by this equipment in the buffer area.

6. Emergency Shower

An emergency shower and eye wash were installed in the chemistry laboratory.

7. Stretcher Kit

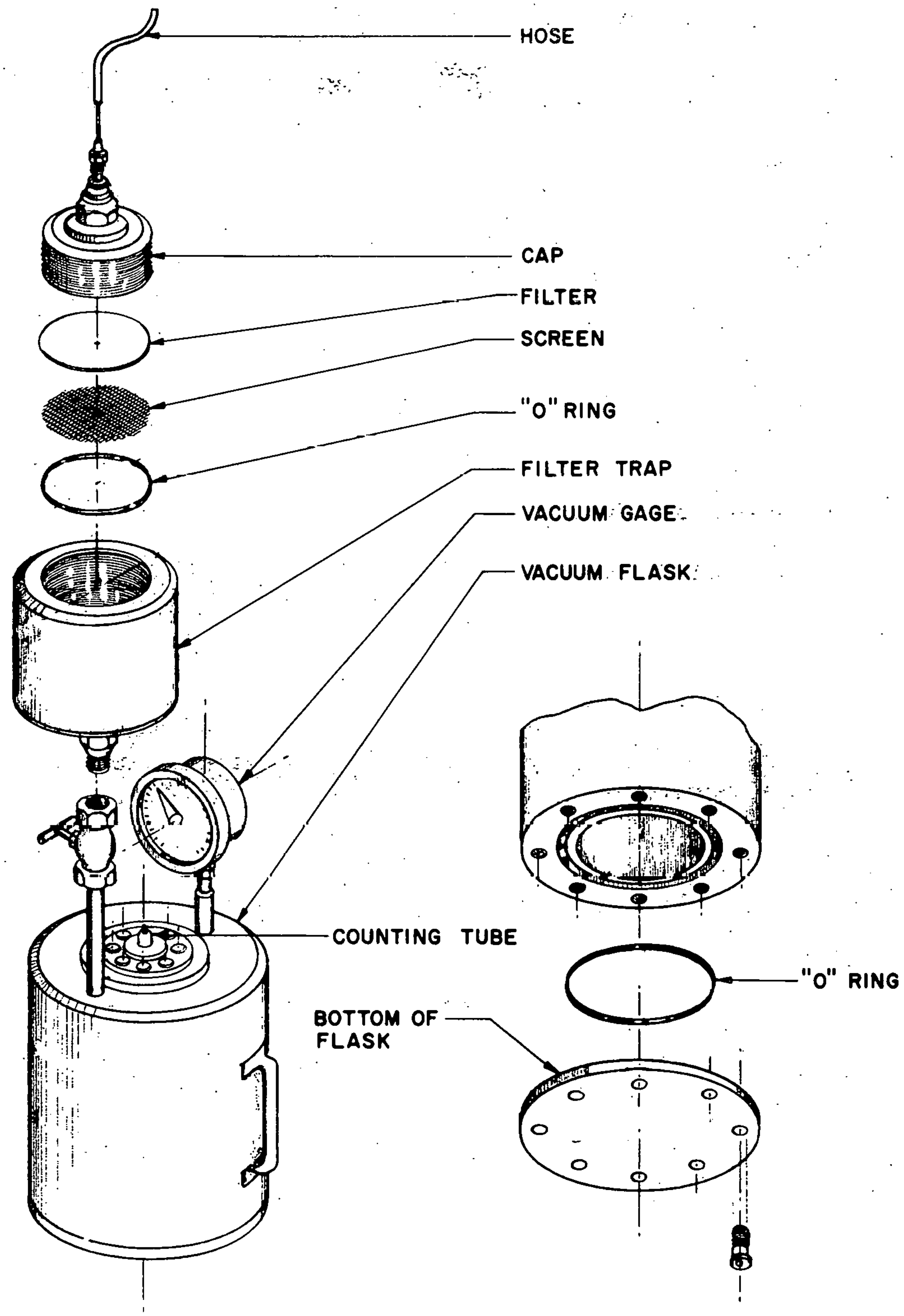
Stretcher kits were installed.

8. Safety Shoe Bins

Safety shoe bins, laboratory and coat racks were received and installed.

H. SAFETY

Based on a total of 120,797 man hours on contract AT(10-1)-967 including Windsor assigned personnel, SL-1 Cadre, and Combustion Engineering, Inc. assigned personnel for the period January 4 through December 31, 1959, there has been no lost time injury. Several minor injuries have occurred and were attributable to improper handling of small tools. In general, the safety performance of SL-1 personnel has been excellent.



GAS COLLECTION CHAMBER

FIGURE 25

VIII SL-1 OPERATIONAL COSTING PROGRAM

In accordance with contractual requirements, an SL-1 operational cost collection program was inaugurated on an informal basis as of May 1, 1959. The procedure used was based on the USAEC Manual, Accounting for Power Reactor Operations, and on an evaluation of the reporting systems presently employed at the Shippingport and APPR-1 Sites.

The scope of the SL-1 operational program, coupled with operating limitations, preclude effective performance analysis of actual program costs. The SL-1 test and training requirements; the under-design of the air condenser; operations at ambient temperatures higher than design; and the designed capacity of the turbine generator for sea level affect a true power plant costing method.

These problems, coupled with variations in manpower requirements between test and purely operational functions, must be considered before a true economic evaluation can be made based on actual cost data. The intent of this section, therefore, will be to evaluate individual cost centers, emphasizing military labor, and normal materials and service charges relative to expected performance, rather than using total actual Combustion Engineering, Inc. and military costs as a basis.

A. OPERATIONS COSTS

The established operational accounts include:

Operations supervision and engineering
Reactor operations
Plant operations
Health physics and safety
Station supplies and expense

The first four accounts are basically labor accounts. An analysis of the effectiveness of the labor collection system may best be made by a review of Table IX.

TABLE IX

MWD - MILITARY OPERATIONS LABOR COMPARISON

Month	MWD	Military Man Months	MWD Per Man Month	Notes
May	20.82	6.2	3.4	Startup after gasket replacement; 3 man crew
June	59.74	7.6	7.9	1,000 hour sustained power; 3 man crew
July	39.30	7.5	5.2	1,000 hour sustained power; 3 man crew
August	0.33	1.1	0.3	Evaluation after 1,000 hour test
September	23.25	4.2	5.5	Instrumented element test; 3 & 2 man crews
October	55.16	5.4	10.2	Sustained power & test; 2 man crew
November	44.03	5.1	8.6	Operational training; 2 man crew
December	41.10	3.7	11.1	Sustained power & annual maintenance; 2 man crew
January	0.00	0.0	0.0	Annual maintenance

In developing Table IX a man month/hour equivalent of 240 hours was used; a calculation was then made indicating the MWD's produced per man month of labor. The results generally show that labor charges were correctly assigned during the period. Further, the high MWD/man month ratio during October and December reflect the use of two, rather than three, man crews. However, a decrease in the man month effectiveness may be noted during both September and November. The megawatt days produced per operational hour during these months were lower due to test and training requirements. As discussed in previous sections, tests were often conducted at zero or varying power levels and training requirements dictated cycling the reactor. But, whatever the power level, a full operational crew is still required. This precludes effective operational cost reporting.

B. FUEL COSTS

Three elements comprise fuel costs:

Fuel assembly consumption
Reprocessing costs
Use charge (commercial basis only)

A conservative estimate of six megawatt years was used in establishing fuel assembly consumption and reprocessing costs. Using actual core costs, including use charges on a commercial basis, and considering the expected residual fuel value, costs per MWD(t) of $54.00 and $56.52 on government and commercial bases respectively, were established. The reprocessing costs of $19,000 were also converted to $8.15 per MWD consumed. The use charge on a commercial basis represents a fixed charge based on actual fuel costs of four per cent per annum or $918.49 per month.

These costs follow actual megawatt day production and are realistic. However, seasonal temperature changes affected the plant efficiency. Differences resulting from this condition would not be found at any Arctic Site.

Summarily, fuel costs of $17,634 and $25,694 on government and commercial bases respectively were accumulated between May and January, representing 283.73 megawatt days. The operational period prior to May 1, 1959 represents 16.6 per cent of estimated design life.

C. MAINTENANCE COSTS

Maintenance accounts established to permit direct collection of SL-1 maintenance costs include:

Maintenance supervision and engineering
Mechanical maintenance
Electrical maintenance
Instrument maintenance
Facility maintenance
Reactor core servicing

A purchase, rather than inventory, costing method was used to facilitate cost collection. This was deemed satisfactory because of the immediate availability of the Central Facilities Watehouse. This allows maintenance of less than 30-day stock levels at the SL-1 site. The system effectiveness may be determined from Table X.

TABLE X

MAINTENANCE MATERIAL, SERVICE AND LABOR ANALYSES

Month	Maintenance Materials & Service Costs	Military Maintenance Man Hours	Cost of Materials & Service Per Maintenance Man Hour
May	$ 758	634	$1.20
June	$ 829	328	$2.53
July	$1372	594	$2.31
August	$2201	1530	$1.44
September	$1148	925	$1.24
October	$1297	679	$1.91
November	$ 652	379	$1.72
December	$1263	824	$1.53
January	$2871	4529	$0.63
TOTALS	$12391	10422	-
Average	$ 1377	1158	$1.19

As may be noted in the cost factors for May, June, and July, there is a discrepancy between recorded costs and the labor hours charged. Materials costing during that period was between one and two months late. Some May material charges were not recorded until June and July. This was corrected by use of the transfer cost method during subsequent months. The low cost per man hour during January was attributable to the inclusion of Military and Maritime trainee charges during the annual maintenance and plant clean-up period. The cost per man hour excluding trainees was $1.39 and would revise the average cost per Military hour from $1.19 to $1.55.

Based on nine months experience, the average monthly SL-1 maintenance materials and service charge was $1,377.00. This is deemed higher than would be normal for an operating plant of the SL-1 type. This was the initial operational period and therefore some of the maintenance costs could be considered as part of the normal debugging process.

D. DEPRECIATION

In accordance with power plant costing procedures, a monthly depreciation rate was established for the SL-1 plant and equipment. When the initial cost procedure was established, only one total plant cost figure of $1,988,783.00 was available. Based on a depreciation rate of five per cent per annum, monthly charges of $8,286.00 and $8,556.00 were established on government and commercial bases.

Initially, these charges were automatically assigned to "Costs Related to Power" each month. This is in accordance with normal power plant costing procedures. However, increased emphasis has been placed on the establishment of the SL-1 as a test and training facility, thereby channeling more plant hours to experimental development and training rather than normal power operations. To permit more realistic costing, a method of allocating the fixed depreciation charge between power and non-power costs was established. This permits depreciation allocation based on the percentage of operational and maintenance hours relative to the total hours available during a given month. This agrees with the depreciation method currently used in the APPR-1 report.

E. CONCLUSION

A review of the above sections will readily indicate that any economic evaluation of the SL-1 plant based on actual costs would not be realistic. However, the cost data collected can be more meaningful if a selective analytical approach is used limiting labor costs to those incurred by the military personnel assigned to plant operations and maintenance. The above has been accomplished subsesuent to the annual operating report period.

It is appropriate to note that, using the equivalent of thirteen operations personnel, the overall monthly mil rate would average between 90-110 mils. For the above reason, it is believed that the collection of operational costs should be continued.

THIS PAGE

WAS INTENTIONALLY

LEFT BLANK

IX APPENDIX

THIS PAGE

WAS INTENTIONALLY

LEFT BLANK

APPENDIX A

COMPOSITE LIST OF TESTS PERFORMED

A. PHYSICS

1. Core Operational Data (PH-5)

The object of the test was to obtain experimental information on the SL-1 forty element core. Data obtained can be used to check reactor code calculations. The clean reactor information obtained included:

a. Cold comparison by periods
b. Rod bank position during heatup
c. Hot rod comparison by periods
d. Rod position at zero and 2000 lbs/hr steam for:

i. Five rod bank
ii. Four rod bank with Rod No. 9 in
iii. Four rod bank with Rod No. 9 out

e. Rod bank position at various power levels

2. Xenon Buildup and Decay (PH-6)

The purpose of this test was to determine in terms of rod bank position:

a. The clean reactor xenon buildup for full power operation
b. Equilibrium xenon at full power
c. Xenon decay after shutdown

3. Rod Bank Position as a Function of Temperature (PH-7)

The rod bank position as a function of temperature during cooldown obtained from this test will provide data to estimate the reactivity effect due to changes in water temperature.

4. * Core Lifetime Data (PH-9)

Periodic measurements of the critical rod bank position under various operating conditions will be made to confirm burnup calculations and core lifetime studies. The critical rod bank position as a function of equivalent full power hours of operation will be recorded for the following:

a. Cold clean reactor
b. Hot clean reactor
c. 100% power, no xenon

* These tests are continuing or being rerun to obtain additional data.

d. 100% power, equilibrium xenon
e. Zero power, equilibrium xenon
f. Zero power, maximum xenon
g. 100% power, maximum xenon

5. Feedwater Temperature Effects (PH-10)

The purpose of this test was to determine the effect of feedwater temperature on the core reactivity.

B. REACTOR ENGINEERING ANALYSIS

1. * Decay Heat Removal Test RA-1

The purpose of this test was to demonstrate the safety of the SL-1 reactor following a scram and complete loss of power. With complete loss of power, water cannot be added to the reactor by the feedpumps and the water level change due to decay heat becomes important.

2. Instrumental Subassembly Test (RA-3)

This test was conducted to determine the use in temperature as a function of heat flux, the maximum fuel plate temperature and a plot of the flux in the test element. The test was a failure as the thermocouples indicated reactor water temperature instead of fuel element sweat temperatures.

3. * Visual Fuel Element Inspection (RA-5)

Several of the fuel elements are visually inspected periodically for corrosion and visible radiation damage.

C. REACTOR ENGINEERING - MECHANICAL DESIGN

1. Temperature Measurements in the Reactor Vessel and Head Flanges (RM-1)

This test was conducted to determine the temperatures in the flanges and will provide a tool for estimating temperatures for future design.

2. Core Holddown Box Removal (RM-2)

This test was conducted to determine the difficulty in removing and replacing the core hold down boxes, working through the openings in the pressure vessel head.

D. PLANT MECHANICAL DESIGN

1. * Steam Quality Measurements (MS-7)

The object of this test was to measure the steam quality at varying reactor water levels and steam flows.

* These tests are continuing or being rerun to obtain additional data.

2. Plant Water Consumption (MS-8)

The purpose of this test was to determine the plant water consumption by measuring the makeup water supplied to the demineralized water storage tank during normal plant operation.

3. Main Condenser Performance (MS-9)

This test was conducted to measure the performance of the main condenser at varying loads and at various positions of louvers, air ejector suction valves, and fan speeds.

4. Plant Ventilation Test (MS-10)

Actual operation of the plant ventilation system; a varying main condenser fan speed was compared with design specifications.

5. Shield Cooling System Performance (MS-11)

The purpose of this test was to evaluate the shield cooling system performance.

6. Feedwater Flow Measurements (MS-14)

The object of this test was to determine the flow and pressure drop characteristics of the feedwater system.

E. PLANT ELECTRICAL DESIGN

1. Calorimetric Calibration of Nuclear Instruments (ES-1)

The four nuclear power level channels were calibrated against a reactor thermal heat balance.

2. Emergency Power Supply Characteristics (ES-2)

The object of this test was to determine the capacity of the emergency supply batteries. The results of this test will furnish design information for predicting the length of time the emergency batteries will supply power to various loads.

3. Instrument Well Temperature Measurements (ES-6)

This test was conducted to determine the use in the instrument wells caused by a loss of the instrument well ventilating fan.

F. SHIELDING AND HEALTH PHYSICS

1. Gamma Dose Measurement on the SL-1 Hold-up Tank (HS-9)

The dose rate on the surface of the holdup tank due to N^{16} decay will be measured. This data will be used to check the calculations presently being used to determine the N^{16} concentration in the reactor water.

2. Gravel Activation (HS-10)

The purpose of this test is to determine the degree of activation of the reactor gravel shielding.

3. Gamma Survey in Gravel (HS-11)

The gamma dose rate as a function of distance through the gravel was measured to compare with calculated values.

G. CHEMISTRY AND METALLURGY

1. * Hydrogen Buildup During Startup (CM-2)

The composition of the gases in the reactor and the rate of hydrogen build-up is of interest in order to determine the venting procedure to be followed during startup.

2. * Measurement of Water Decomposition (CM-3)

The purpose of this test is to determine the rate of water decomposition and the oxygen formation rate as a function of reactor power and reactor water quality. The oxygen content is of concern because of its effect on corrosion.

3. * Determination of System Decontamination Factors (CM-4)

The excessive entrainment of long-lived radioactive substances and their subsequent retention on the power generating machinery could lead to significant problems in shielding and maintenance. The purpose of this test is to determine the decontamination factors between the reactor water and steam as a function of steam flow rates.

4. * Mixed Bed Ion Exchange Resin Evaluation (CM-9)

This test is being conducted to evaluate the chemical properties of ion exchange resins for use in the SL-1 plant.

* These tests are continuing or being rerun to obtain additional data.

APPENDIX B

MAJOR MODIFICATIONS OF THE SL-1 PLANT DURING THE YEAR

Control rod drive seal shafts plated

Core holddown units installed

Spool piece for Rod #9 removed

Interlock removed from electrical bus tie circuit breaker

Rod #9 microswitches modified

Gland air ejector piping modified

Diatomaceous earth oil filter installed in plant water supply piping

Piping from the main condenser was revised to provide equal removal of non-condensable gases from all sections

Boric acid piping system modified to allow hydro-testing of pump

Permanent reactor venting valve for startup installed

Valve tags installed throughout the plant

Log N recorder to allow calibration of entire range

P-Po modified indicator cam and limit switches installed

Electrical rod drop timer installed

Fuel transfer coffin and mechanism modified to allow ease of operation

Bolt and gasket material changed in all stainless steel steam valves

Additional tachometer installation installed for TG unit

Purification pump interlocks installed

Fuel element transfer procedure modified

Control rod cooling water filter installed

Liquid flow meter installed in the shield cooling system

Visual reactor water level indicator fabricated for hot calibration of permanent water level indicators

Repiped purification system to allow 5 gpm flow

Domestic water No. 40 regulating valve relocated

Plant water drain ditches modified

Gage glasses installed on HP and LP tanks

Vent valve material changed

Ventilation overload switch relocated

Auto rod upper limit changed

Power supply for control rod clutch and nuclear panel ventilation fans separated

Scram protection installed on nuclear Channel IV

Bypass switches on all scram circuits

APPENDIX C

SL-1 MAINTENANCE AND INSPECTION TASK LIST

Following one year of operation of the SL-1 facility, a thorough maintenance and inspection program was performed. The scope of the program was dictated by the need for a thorough evaluation of systems and equipment performance and wear. The maintenance inspection period totaled 21 days on an eight hour per day basis. A total of 2,370 manhours effort was required.

It is concluded that this scope and manpower effort cannot be considered a guide for PL plants since evaluation was the objective rather than repair to operational condition of the equipment. It is further concluded that maintenance requirements limit SL-1 plant operation only on the turbine governor and on the four canned rotor pumps where parallel equipment is not installed. Replacement of the governor with a conventional turbine governor with a constant recirculating oil purification should remove this limitation. The use of parallel backup pumps for the four canned rotor pumps will remove this maintenance limitation.

The work list of maintenance and inspection task is presented below:

ELECTRICAL

1. Rewire and conduit work for additional and relocation of receptacles in Chemistry Laboratory.
2. New power cables and receptacles for maintenance shop bench relocations and lathe.
3. Check out lathe electrically.
4. Remove connection box SB-30 and replace with larger one; necessitates rewire, cutting old two inch conduit, rethreading cut conduit, installing terminal strips.
5. Replace purification pump controller.
6. Repair and replace purification pump mercoids and rewire.
7. Maintenance and inspection of turbine generator.
8. Insulation checks of all electrical motors.
9. Repair one voltage meter on electrical panel.
10. One watt meter on Generator Control Center for face remarking.
11. Clean all motors and turbine generator and paint. Color code conduit.
12. Repair bus tie break circuit.
13. Label all electrical fuses.
14. Install additional 120 volt receptacle in electric shop.
15. Install new gasket in No. 2 feedwater pump.
16. Install microswitch on shield water pump.

MECHANICAL

1. Remove, inspect, clean all control rod drives.
2. Install hose racks in pump house.
3. Spring load hydrogen vent valve.
4. Repair leaks in simulated heat load system.
5. Install compound gages on all sections of the main condenser.
6. Clean up water in fuel element storage wells.
7. Recheck all valve tags.
8. Complete pipe marking and flow direction on all piping system.
9. Install relief valve on chlorination tank.
10. Install hooks for shield block lifting gear.
11. Clean boron system, inspect, replace solution.
12. Pump out waste storage sumps.
13. Install temperature bulb in turbine exhaust line.
14. Install additional pressure gage on turbine steam line.
15. Install compound gage in flash tank.
16. Flush out and inspect shield cooling system.
17. Plate conductivity cells and standardize.
18. Remove, hand test and inspect main steam safety valves.
19. Fabricate reactor visual water level indicator.
20. Lift turbine casing, inspect blades, nozzles, glands, etc.
21. Flush turbine lubricating oil system, replace oil, test.
22. Calibrate pH instruments.
23. Clean all air condensers.
24. Clean and inspect 15 tanks.
25. Patch asphalt cover on gravel shield.
26. Repack all valves.
27. Replace all chemical pump bearings.
28. Remove casing, clean, inspect main reduction gears.
29. Clean all ventilation ducts.
30. Clean and repack all gage glasses.
31. Inspect and clean 11 steam and water strainers.
32. Replace all ion exchange resins.
33. Clean and paint entire plant.
34. Tighten all nuts, bolts, etc.
35. Pull all heat exchanger heads for inspection.
36. Relag all broken lines.
37. Change all filters.
38. Inspect and replace diaphragms in all regulating valves.
39. Inspect all motor operated valves.
40. Inspect all centrifugal pumps for wear and replace parts as necessary.
41. Test all relief valves using compressed air.
42. Replace turbine throttle valve and stem.
43. Clean and inspect all hypochlorinators.
44. Bring up to date all maintenance records.
45. Replace SL-1 drawings with latest revised prints.
46. Install lathe.

INSTRUMENTATION

Calibrate indicator:

1. Condenser from hotwell.
2. Condenser from precooler.
3. Condenser to precooler.
4. Gland seal leak.
5. Retention tank temperature.
6. Simulated heat load temperature (TI 231).
7. Simulated heat load temperature (TI 230).
8. Turbine exhaust end bearing drain temperature.
9. Turbine gear drain temperature.
10. Turbine steam end bearing drain temperature.
11. Turbine oil tank temperature.
12. Turbine oil cooler outlet temperature.
13. Controller, air ejector after-condenser temperature.
14. Controller, main steam relief valve.
15. Controller, main steam safety valve.
16. Controller, purification system temperature.
17. Controller, condenser pressure (mercoid).
18. Controller, reactor steam pressure high (mercoid)
19. Controller, reactor steam pressure low (mercoid).
20. Main steam pressure.
21. Purification pump pressure.
22. Reactor steam pressure.
23. Retention tank pump pressure.
24. Simulated heat load shell pressure.
25. Steam to condenser air ejector.
26. Steam to gland air ejectors.
27. Turbine bearing oil pressure.
28. Turbine exhaust end gland pressure.
29. Turbine steam end gland pressure.
30. Bypass valve discharge.
31. Controller, shield cooler pump discharge pressure.
32. Controller, turbine oil pressure.
33. Resistance check, uncompensated chamber, channel 1.
34. Resistance check, uncompensated chamber, channel 2.
35. Resistance check, compensated chamber, channel 3.
36. Resistance check, compensated chamber, channel 4.
37. Resistance check, chamber BF_3, channel 5.
38. Resistance check, chamber BF_3, channel 6.
39. Health Physics condenser return tank vacuum.
40. Turbine exhaust vacuum.

Preventive maintenance:

41. Calibrate recorder, condenser vacuum.
42. Calibrate transducer, condenser vacuum.
43. Micro-ammeter, channel 1.
44. Micro-ammeter, channel 2.
45. And repair micro-micro-ammeter, channel 2.
46. Constant air monitor.

47. Drive unit bypass valve.
48. Drive unit feedwater valve.
49. Hand and foot counter.
50. Repair, calibrate channel 8, remote area monitor.
51. Indicator, resistivity.
52. Linear count rate meter.
53. Control rod drive units 1, 3, 5, 7, 9.
54. Damper controller.
55. Channel 1 power supply.
56. Channel 2 power supply.
57. Channel 4 power supply.
58. Channel 5 power supply.
59. Channel 6 power supply.
60. Channel 7 power supply.
61. Calibrate recorder, bypass.
62. Calibrate recorder, condenser air inlet temperature.
63. Calibrate recorder, condenser air outlet temperature.
64. Calibrate recorder, reactor water level.
65. Calibrate recorder, feedwater temperature.
66. Calibrate indicator, P-Po.
67. Calibrate recorder, main steam flow.
68. Calibrate recorder, reactor steam pressure.
69. Calibrate transmitter, bypass steam flow.
70. Calibrate transmitter, hotwell level.
71. Calibrate transmitter, P-Po.
72. Calibrate transmitter, main steam pressure.
73. Calibrate transmitter, reactor steam pressure.
74. Tube tester.
75. Calibrate reactor water level system, primary.

Other maintenance:

76. Install AC alarm system.
77. Install wiring change on instrument panel fan circuit.
78. Repair water level scram, low microswitch.
79. Modify scram actuator on reactor pressure scram mercoid.
80. Complete inspection and repair of secondary water level head.
81. Calibrate, hot, secondary level trol.
82. Preventive maintenance and calibrate of channels 5 and 6.
83. Clean external surfaces of transducers.

APPENDIX D

SUMMARY OF CAPITAL EQUIPMENT

Administrative Equipment

1	Thermofax Copy Machine
2	IBM electric typewriters
10	Executive desks
2	Steno desks
1	Remington-Rand Printing Calculator
1	Pacific Floor Machine, including scrubbing brush and polishing brush
2	Steel office tables
3	Bookcase sections, including bookcase base.
1	Multilith "80"
1	Marchant Calculator

Laboratory and Test Equipment

1	Anemometer
1	Electric drying oven
1	Analytical balance, including weights and table.
1	Clinical centrifuge
1	Neptune mobile electric air compressor
1	Offner, Type R, Dynograph Assembly
1	Multipoint Recorder

Health Physics and Safety Equipment

2	First aid stretchers and kits
2	Scot Air Pak, complete with Scot aramic face piece
1	Stack effluent monitoring equipment
1	Portal radiation monitor

THIS PAGE

WAS INTENTIONALLY

LEFT BLANK

APPENDIX E

SUMMARY OF OFFICIAL VISITS

A total of 179 official visits were made to the SL-1 site by representatives from many interested organizations. These visits are tabulated below:

Number of Visitors	Affilitation
55	U. S. Atomic Energy Commission, Washington, D. C.
42	U. S. Military Services
9	U. S. Congress
34	Foreign organizations
12	Medical groups
7	States Marine Lines
5	U. S. Public Health Service
15	Miscellaneous