IDO — 19300 [redacted]

CEND-128

26

SL-1 REACTOR ACCIDENT

On January 3, 1961

INTERIM REPORT

May 15, 1961

Contract AT(10-1)-967

U. S. ATOMIC ENERGY COMMISSION

NUCLEAR DIVISION

COMBUSTION ENGINEERING, INC.

[redacted]

[redacted]

IDO-19300
CEND-128

SL-1 REACTOR ACCIDENT

ON JANUARY 3, 1961

Interim Report

Public Reading Room
U. S. Department of Energy
Idaho Operations Office

May 15, 1961

Contract Number AT(10-1)-967

U. S. Atomic Energy Commission

Classification:

Official Use Only

John B. Anderson 5/15/61
Authorized Classifier Date

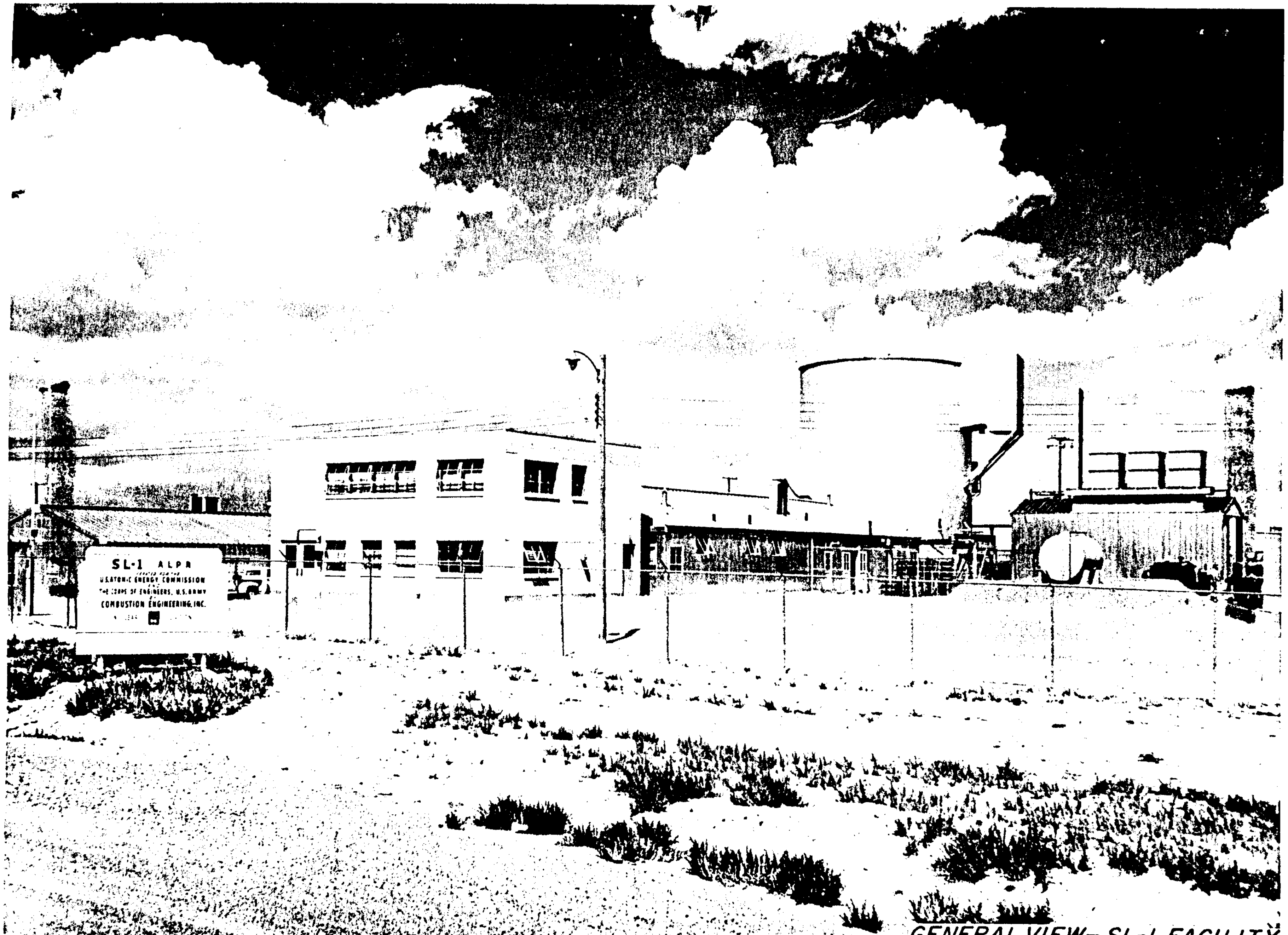
Approved

John B. Anderson

COMBUSTION ENGINEERING, INC.
Nuclear Division

Windsor Connecticut

[redacted]



GENERAL VIEW– SL-1 FACILITY

INTRODUCTION

The SL-1 power plant (originally designated ALPR), prototype for a remote arctic installation, was designed, constructed and initially operated by Argonne National Laboratory. It is located at the National Reactor Testing Station near Idaho Falls, Idaho. Combustion Engineering was selected as operating contractor for this plant on the basis of their response to an Atomic Energy Commission invitation issued in June, 1958, and assumed operating responsibility on February 5, 1959.

After nearly two years of operation a nuclear excursion occurred on the night of January 3, 1961, when a military crew of three men were assembling the reactor control rod drive mechanisms. The resulting blast killed the three crew members, produced extensive damage inside the reactor vessel and secondary damage to the reactor room by ejected missiles. High radiation levels from the reactor and ejected materials have restricted recovery operations. These high radiation levels are all within the reactor building and the SL-1 Facility area.

This interim report contains a chronology of the accident including a reconstruction of the condition of the reactor before and after the excursion based on presently available evidence. All of the evidence obtained to date is included in some detail. Based on this evidence, an evaluation as thorough as now possible of the nature, initiating mechanism and extent of the excursion is presented. In addition, the first part of the report contains extensive information covering contractual relationships, procedures, operational experience and analyses with emphasis on areas related to the excursion, or for which information has been requested.

CONTENTS

CHARTS AND TABLES

FIGURES

SUMMARY

The SL-1 is a small (200 kwe) nuclear power plant designed by Argonne National Laboratory to generate electric power and space heat for remote arctic installations. The plant, shown in the frontispiece, is located at the National Reactor Testing Station, Idaho. A cutaway perspective of the reactor building and the adjoining control room is shown in Figure 1. In general, the lower portion of the cylindrical building contains the natural circulation, direct cycle boiling water reactor (Figure 5) surrounded by gravel shielding; the turbine generator and plant equipment is in the reactor room at the middle level and the air-cooled condenser with its circulation fan is mounted above at the third level. An additional air-cooled condenser and related systems, installed by Combustion Engineering in 1960, is located in a separate building shown to the right of the reactor building in the frontispiece photograph.

The operation of SL-1 by Combustion Engineering started on February 5, 1959, after initial operation by Argonne National Laboratory. Under the contract with the Atomic Energy Commission, the reactor was operated by Combustion Engineering for the purposes of obtaining SL-1 sustained operation experience, training military crews, obtaining data and experience in support of improved designs in the Army Boiling Water Reactor Program and for testing components for such improved designs.

Military personnel were assigned to the SL-1 as operating crew and for training. Such personnel performed operational and maintenance functions under the over-all management and technical direction of Combustion Engineering, Inc. The plant was operated by military crews on a continuous rotating shift basis. This was in accordance with instructions from the Atomic Energy Commission. The Combustion Engineering operating budget did not include approval for a staff sufficient to provide supervision on all shifts.

The administrative arrangement, under which Combustion Engineering worked with the military for training crews, was limited to operator training with the SL-1 plant under the general direction of Combustion Engineering, except in the case of health physics where both classroom

and operational training were provided. Although Combustion Engineering was not responsible for the establishment of the military training program and was never requested to conduct a formal review of the program, it did consider the training to be adequate. The military training program was formal in content and presentation and was based on up-to-date information.

The SL-1 core and structure was fabricated of X-8001 aluminum alloy. The core consists of 40 flat plate fuel assemblies (Figs. 6 and 8), containing 14 kg of U-235. Control of the reactor was accomplished by five cruciform aluminum (X-8001) clad, cadmium control rods. In addition burnable poison was added in the form of aluminum-boron strips (some half and some full length) which were spot welded to the sides of the fuel assemblies. These aluminum-boron strips bowed between spot welds as burn-up progressed until in August, 1960 it was difficult to remove fuel assemblies for inspection (Figs. 15 and 16). During this inspection, several of these strips from a fuel assembly removed from the center of the core appeared to be almost completely disintegrated (Figs. 17 and 18). The fuel plates at this time showed no signs of radiation damage or corrosion.

During the analytical evaluation of the SL-1 reactor in 1959, the beginning of life reactivity margins and the lifetime behavior were estimated. The calculations showed a 2 to 3% reactivity bias when compared with the observed criticality. A large part of the bias appears to be attributable to the treatment of the self-shielding of the boron burnable poison, which is also the source of the major uncertainty in the lifetime calculations. The complexity of the boron distribution in the core and in the core geometry itself necessitated a simplified treatment of boron. Based on simplified lifetime calculations, a predicted rod bank curve was obtained. The difference between this curve and the observed rod bank curve has been used to estimate the amount of boron lost by corrosion, or mechanically from the core. In view of the uncertainties attached to the boron self-shielding, estimates of the loss of boron by comparison of observed with predicted rod bank positions cannot be considered reliable.

A reactivity history of the core has been compiled based on the observed critical rod bank positions taken from log entries for both controlled test conditions (Fig. 30) and on a routine basis (Fig. 33). Comparison of the data from the two sources indicates that they are mutually consistent. The slight outward motion of the rods early in core life is apparent, and may be explained by the lack of a consistent zeroing procedure prior to 100 MWD of operation and/or a decrease in boron self-shielding with a concurrent build-up of samarium. The steady inward motion of the rods during the period from 300 to 700 MWD of operation is also apparent both from the test data and the log data.

Using the observed cold rod bank position during life and the results of rod calibrations taken at several times during the life of the core, the shutdown margin as a function of core life was estimated. Using the most conservative (lowest rod worth) of the rod calibrations, it appears that the shutdown margin decreased from 3.4% $\frac{\Delta K}{K}$ at beginning of life to a minimum of 2.1% $\frac{\Delta K}{K}$ at 710 MWD (Fig. 34). Insertion of cadmium strips in two of the Tee rod slots raised the margin to 2.9%. This is the best estimate available for the shutdown margin at the time of the incident.

On the basis of this shutdown margin and calibration curves for rod No. 9, the indicated rod No. 9 positions (at 83°F with all other rods inserted) for critical, prompt critical and 1.8% $\Delta K/K$ supercritical are 17.3; 19.5; and 24.3 inches respectively. A disassembled control rod must be lifted a greater distance to reach these positions since when resting on the core, it is almost 4 inches below indicated zero.

The SL-1 control rod drive mechanism (Fig. 36) is a rack and pinion type with a controlled leakage, pressure breakdown seal. A geared motor drives the pinion shaft through a magnetic clutch for rod motion. On scram, the magnetic clutch is de-energized and the rods move into the core by gravity.

Procedures were prepared by Argonne National Laboratory and Combustion Engineering covering the operation of the SL-1 reactor including the various routine tests to be performed on the rod drive and scram system to insure their proper operation before and during regular reactor startup. These procedures were written to give the maximum assurance for safe reactor operation.

The control rod drive manual was written by the Training Branch Nuclear Power Field Office in lieu of a manufacturer's technical manual which was not available. The manual describes the assembly and disassembly procedures for the control rod and control rod drive assembly components. Using this procedure, the disassembly and assembly of control rod and control rod drive components had been performed many times by the military crews prior to the January 3, 1961 incident. The military were well trained in this procedure, and the men in charge on January 3 had carried it out on the

SL-1 reactor a number of times before.

The Operating Logs, covering the SL-1 operating history, list instances where control rods were not operating satisfactorily during rod withdrawals, rod exercises, rod drop tests or during rod scrams (complete listing in Appendix A). The percentage of individual rod sticking incidents during a scram, or rod drop tests (greater than 10") was approximately 2.5% during the first twenty-two months of operation (up until November 18, 1960). During the month prior to the last shutdown (November 18 to December 23, 1960) the percentage of similar incidents increased to 13%. The stickings always occurred in a very erratic and random fashion.

Considering the marginal performance of some of the components, it is believed that the reason for rod sticking was due largely to the control rod mechanism. It is also believed that control rod misalignment and inward distortion of the shroud (which may be present) contributed to the over-all frictional resistance of the system but was not in itself a prime cause. If inward distortion of the shrouds due to distortion of the aluminum-boron strips was the prime cause for sticking, the central control rod (No. 9) would have been more affected than any other rod because it is completely surrounded by fuel assemblies containing two aluminum-boron strips each. No. 9 rod has the best over-all operational record and had been successfully scrammed 130 times during the six months prior to the last shutdown period, with only one instance of sticking where it hesitated momentarily at the start of a scram.

Recognizing these problems, Combustion Engineering's design effort for a PL type replacement core and rod drive mechanisms for SL-1 was directed toward a mechanism which takes full advantage of the SL-1 operating experience. The major improvements in the PL rod drive mechanism design include the elimination of the need to raise a control rod during a coupling operation; an improved scram shock absorption system; the use of improved pinion bearings and a face type seal, or an increased clearance pressure breakdown seal. The PL type lead mechanism, as shown in Figures 39 and 40, has been fabricated and is about to be subjected to a rigorous test program in the laboratories at Windsor, Connecticut.

An approach to the limit of the stable operation range and incipient instability of the reactor occurred in November, 1960 during a program to increase the operating power level to 4.7 MW in order to test the recently installed PL type condenser. The existence of such a stability limit for natural circulation boiling reactors is well known and it was possible to improve the stability characteristics of the SL-1 to give stable operation at 4.7 MW by a revision of the control rod programming. Prior to the revision in rod programming, the reactor experienced an over-power scram during a test investigating the instability. This occurred during operation after the insertion of the cadmium shims to improve the shutdown margin These shims being at the perimeter of the core increased the maximum-to-average power ratio, thus producing a somewhat more unstable situation.

Water quality control for the SL-1 has been satisfactory during operatio Operation below the specification limits of pH of 6 to 7.5 and resistivity greater than 500,000 ohms has occurred in isolated instances but steps were immediately taken to rectify the situation.

Fission product release has been constant for some time. To the date of the incident, however, no major clad ruptures had been noted. Activity is primarily from inert gases released through the air ejector, and iodine isotopes carried over in the steam. It has been determined that fission product release is delayed by some unknown mechanism and is not due to recoil from surface uranium contamination. Non-volatile radioactive particl are well contained in the reactor vessel due to the high water-to-steam decontamination factor of 10,000. Reactor water purification is controlled by the ion exchange system.

There is no evidence that any of the operational problems or changes that occurred in the SL-1 reactor during its operation, as discussed above, made a direct contribution to the accident that occurred on the night of January 3, 1961. This accident, that produced the effects of an explosion, was a nuclear excursion. The explosion fatally injured the military crew of three men who were engaged in reassembling the control rod drive mechanism. Severe damage to the reactor resulted while damage outside the reactor appears to be limited to the effects of missiles on the building interior (Figs. 51; 52; 58 and 61 are typical). The major missiles were several shield plugs ejected from control rod ports in the reactor head (Fig. 56). Intense radiation levels hampered the operation to remove the bodies and determine the condition of the reactor. The removal of the bodies was completed January 8, 1961, and by April, 1961 Combustion Engineering had photographed both the outside of the reactor head and the internals of the reactor and probed to the bottom of the reactor vessel to find no indication of the presence of water.

The assembly of the SL-1 control rod drives requires limited lifting (4 to 6 in.) of the control rod to install a nut and washer. The evidence indicates that the crew was at this stage of the assembly operation when the incident occurred. Presumably, the central control rod (No. 9) was lifted too high for some unexplained reason. Study of the interior of the reactor after the accident indicates that the four outer control rods are apparently still in place and the central rod, No. 9, and some of the structure guiding the rod in the core is lying on top of the core (Fig. 59).

An attempt has been made to correlate the observed mechanical and nuclear evidence with the probable characteristics of a power excursion as inferred from BORAX and SPERT experience. This experience makes possible an estimate of the excess reactivity required to produce the mechanical effects, which is thought to be reasonably good. The estimate indicates a value of about 1.8% excess reactivity above delayed critical. Tests on a mock-up of the SL-1 control rod assembly indicate that this amount of reactivity could be added manually at a sufficiently rapid rate to produce an excursion, although the addition of this much reactivity apparently would have required almost full withdrawal of the center control rod.

The theoretical estimates of the nuclear energy release associated with such an excursion are more uncertain, and cover a range from 80 to 270 MW sec. It is probable that the actual energy release was closer to the lower limit of this range. The nuclear evidence appears to bracket this range of estimates, with the determination from fission product analysis being somewhat below the lower limit, and the indication from external activation and external radiation monitors lying above the upper limit. Interpretation of the nuclear evidence involves large uncertainties.

There does not appear to be any need to postulate energy releases other than nuclear to account for the observed effects.

I. DESCRIPTION OF SL-1 REACTOR AND PLANT

A. GENERAL ARRANGEMENT

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

Figure 1 is a cutaway perspective of the 38 ft. 7 in. diameter by 48 ft. high reactor building and adjoining control room. In general, the lower portion of the cylindrical building contains the reactor vessel surrounded by gravel shielding; the turbine-generator and other plant equipment is located on the reactor room floor at the middle level (see photos, Figs. 2, 3 and 4) and the air-cooled condenser with its circulation fan is mounted above at the third level. The control room in the adjoining building is connected to the reactor operating floor by a stairway. An additional air-cooled condenser, provided by Combustion Engineering, is located in a separate building as shown to the right in the frontispiece photograph of the SL-1 facility. Design details of this plant are given in four reports:

ANL 5744	-	Hazard Summary Report on the Argonne Low Power Reactor (ALPR)	October, 1957
ANL 6084	-	Initial Testing and Operation of the Argonne Low Power Reactor (ALPR)	December, 1959
IDO-19003	-	SL-1 Reactor Evaluation Final Report	July 15, 1959
IDO-19016	-	SL-1 Plant Expansion Hazards Evaluation	June, 1960

A summary of some SL-1 characteristics follows:

Reactor heat output	3 MW(t)
Steam Production	9020 lb/hr.
Steam pressure	300 psig
Steam temperature	421°F (saturated)
Turbine generator output	300 KW(e)
Space heating load	400 KW(t)
Core design lifetime	3 years
Core fuel loading, U^{235}	14 Kg
Burnable poison, B^{10}	22.6 gm

B. REACTOR CORE

A general elevation view of the core is shown in Figure 5. A photograph of the reactor core looking down into the vessel from above is shown in Figure 6.

The SL-1 core was fabricated from an aluminum-nickel alloy (Alcoa X-8001). The core structure is made up of two main components, the core shroud and the core support grid. The entire core weight is borne by the stainless steel support grid, which is bolted to the core support pads attached to the thermal shield, as shown in Figure 7. Figure 5 shows the sheet aluminum shrouding riveted to the core stanchions to form both control rod scabbards and envelopes to contain fuel assemblies. The control rod scabbards extend about 26 inches above the core (Figs. 5, 6 and 7) to form a shroud around the rods when they are raised. There are five cruciform control rod scabbards and four Tee scabbards.

The structure provides a total of sixteen (16) envelopes to contain fuel assemblies. The four corner envelopes hold three fuel assemblies each and the remaining twelve (12) hold four fuel assemblies each. The maximum core capacity is thus fifty-nine (59) fuel assemblies and one source assembly. The SL-1 core loading consisted of only forty (40) fuel assemblies arranged to approximate a right circular cylinder and twenty (20) dummy assemblies, one of which contained an Sb-Be neutron source. This arrangement of the core is shown in Figure 8, which is a drawing of the core configuration as it existed just prior to the incident. The active core is 25.8 inches high with an equivalent diameter of 31.4 inches and an over-all water to metal ratio of 2 to 1.

The fuel assemblies, as shown in Figure 9, consist of nine 0.120 inch thick fuel plates assembled to two side plates by spot welding to form a box 3-7/8 inches square. A fuel plate consists of a 0.050 inch thick by 3.5 inches wide and 25.8 inches long center portion of aluminum-nickel-uranium alloy in a picture frame of X-8001 aluminum alloy, and side cladding of .035 inch thick X-8001 aluminum per side.

Each fuel assembly has a full length burnable poison strip of aluminum-nickel containing boron which is spot welded to one side plate. The strip is 25.8 inches long by 3.875 inches wide by a nominal 0.026 inch thick and contains 0.5 gram of B^{10}. In addition, the sixteen (16) center fuel assemblies have a half-length strip welded to the lower half of the opposite side plate. This strip is nominally 0.021 inch thick and contains 0.2 gram of B^{10}.

The fuel assembly spacing is maintained by Inconel springs which are fastened on each of the four sides at the top of the assembly. Fuel handling is accomplished by a gripper mechanism which attaches to a stainless steel gripper tip threaded and pinned into the upper end of the fuel assembly.

A holddown device rests on top of each group of four assemblies. It consists of a 7-7/8 inch square box, 3 inches high, fabricated of X-8001 aluminum alloy (Fig. 5). A 1/2-inch thick cross welded within the box has a gripper tip mounted in its center which is identical to the gripper tips on the fuel assemblies. Each core holddown device is intended to prevent hydraulic lifting of the fuel elements. Calculations, considering a fuel assembly as a free body, indicate that insufficient hydraulic forces occur during normal reactor operation to lift the fuel assemblies, therefore, the holddown boxes are not necessary.

Holddowns were not installed when the reactor core was initially assembled. During the period April 3 to 23, 1959, when the vessel head was removed to replace head gaskets, ten holddowns were installed. Figure 6 is a photographic view of the reactor core taken after installation of seven of the ten holddowns.

The forty (40) fuel assembly core utilizes five cruciform control rods composed of cadmium sheets with X-8001 aluminum alloy cladding. Figure 10 shows the control rod. The cadmium portion of the cruciform is 14 inches 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 cladding is dimpled and spot welded. The centrally located rod (No. 9) has a 17 inch bottom extension made of solid X-8001 aluminum alloy plate and the remaining rods have 5 inch extensions. Stainless steel ball-joint end fittings riveted to the upper end of the rods are used to connect the control rods to the drive mechanisms.

C. CONTROL ROD DRIVE MECHANISM

The five cruciform control rods are actuated by rack and pinion drive mechanisms. Figure 5 shows the control rod drive mechanisms mounted on the vessel head nozzles. The mechanism installed on No. 4 nozzle is a test mechanism which does not affect reactor control. The rack teeth of No. 9 mechanism in the cutaway view of Figure 5 are shown rotated 180° from true orientation for illustrative purposes.

A detailed explanation of the control rod drive mechanisms and their drive packages is given in Section IIC3. (also see Fig. 36).

D. VESSEL AND HEAD

The SL-1 reactor pressure vessel is carbon steel (Type SA-212) clad with stainless steel (Type 304). It was designed for 400 psig pressure with a metal temperature of 500°F. The vessel consists of an ellipsoidal dished bottom head, a cylindrical center section with a top flange, and a flat upper head. Figure 5 shows a view of the reactor pressure vessel. The internal diameter is 52-1/8 inches and the inside length is 14 ft. 6 inches.

The stainless steel clad upper head has nine flanged, 6-inch diameter nozzles for control rod drives, one 4-inch diameter liquid level control opening, and one 2-1/2 inch diameter liquid level control opening. The over-all height of this head assembly is 24 inches.

A cylinder of 1/4-inch steel is welded to the top of the head surrounding the nozzles to form a container for shielding material consisting of iron punchings, boron-steel and gravel (Fig. 5). A cover with appropriate holes for the nozzle flanges is tack welded to the cylinder. The upper head closure is a bolted connection sealed with two spiral wound metallic gaskets having a leak-off groove between them. The over-all vessel and head height from the inside bottom of the vessel to the top of the head nozzles is 16 ft. 6 inches.

The vessel has three internal base pads spaced 120° apart to support the thermal shield. There are five nozzle penetrations in the upper section of the vessel for the following piping: 1) a 4-inch diameter steam outlet; 2) a 1-1/2 inch diameter upper spray ring; 3) a 1-1/2 inch diameter lower spray ring; 4) a 1-inch diameter steam separator return line; and 5) a 1-inch diameter purification system outlet pipe.

The vessel is thermally insulated with a 3-inch banded layer of magnesia which is protected by a 1/4-inch steel cover. The pressure vessel is installed inside a steel cylinder consisting of two half shells bolted together along vertical seams. It is supported by its upper flange resting on the top edge of the support cylinder. The cylinder, in turn, is supported by the reactor building steel structure.

E. REACTOR AND POWER PLANT SYSTEMS

The SL-1 facility, being a natural circulation, boiling water reactor plant, has a simple compact arrangement. Steam produced in the reactor vessel passes through the turbine to an air-cooled condenser, where it is condensed and returned to the reactor with the main feed pump. A brief description of the major systems follows:

1. Main Steam System

Steam from the reactor flows through a pressure control valve to either the turbine or the turbine pressure regulator. The turbine pressure regulator by-passes a set amount of steam and is capable of passing all of the turbine steam load in the event of a turbine trip. A second back pressure regulating valve serves to maintain a pressure of 40 psig in the space heat exchanger and downstream from the turbine pressure regulator.

2. Condensate System

Exhaust steam from the turbine is condensed in an air-cooled finned tube-type condenser operating at 5 in. of mercury absolute pressure. Condenser cooling air is supplied at the proper temperature and flow rate to provide constant turbine back pressure. This is accomplished by controlled mixing of the re-circulating and incoming air streams.

3. Feedwater System

Condensed steam from the condenser, air ejectors, and space heating system is collected in the hotwell tank. This water is returned to the reactor by one of two feedwater pumps. Condensate is also used as cooling water in the primary shield cooling heat exchangers and the air ejector after-condensers. A separate condensate circulating pump is used to supply these systems with water. Primary shield cooling is also provided by a natural circulation loop to an air-cooled finned tube-type heat exchanger.

Water level in the hotwell is maintained to provide adequate submergence of the feedwater pumps. Water can be added manually for make-up reasons from the demineralized water storage system. The returning feedwater serves as the coolant for the purification water cooler. In this cooler the 135°F feedwater is heated to 175°F by the heat supplied from the reactor water. The feedwater is passed through a filter and then enters the reactor through a spray ring located at the level of the top of the reactor core.

4. Primary Water Purification System

Reactor water is continuously re-circulated through a purification system at the rate of 3 to 5 gpm. This system removes suspended and dissolved impurities in order to control the build-up of radioactivity by deposition in the plant systems and turbine.

Water from the reactor is taken out near the top of the core and returned through the feedwater line. The water, coming from the reactor, first passes through a 5-gallon purge water holdup tank to reduce the N^{16} activity. Then the water is cooled by regenerative heat exchange with the feedwater. After cooling, the water is pumped through a filter, a mixed bed demineralizer and returned to the feedwater line. Part of the flow by-passes the mixed bed demineralizer and flows to a cation demineralizer to maintain pH between 6.5 and 7.

5. Poison Injection System

A back-up shutdown system is included in the design of the SL-1 plant, which provides for the addition of boric acid to the reactor water. At the discretion of the operating personnel, a concentrated boric acid solution may be pumped into the reactor through the lower feedwater spray ring. The manually operated pump has a capacity of at least 25 gal/hr. when the reactor is at operating pressure. With the vessel at atmospheric pressure, the solution can be introduced through the upper feedwater spray ring by gravity feed through a by-pass hose.

6. Plant Expansion Facility

The purpose of the plant expansion facility (designed and installed by Combustion Engineering) is to provide additional heat dump capacity for higher power operation of the original SL-1 plant. This system will handle an additional 13,000 lbs/hr. steam flow, thus providing capacity for reactor operation at powers up to 8 MW. It consists mainly of a PL-2 type air-cooled condenser, hotwell, air ejectors, return booster pump, and required instrumentation and controls.

A 3" take-off from the main steam line leads to the expansion facility where the steam flow is controlled by a motor operated throttling valve. The valve and transition piece downstream reduce the reactor steam pressure to approximately atmospheric pressure as it is fed into the condenser inlet header. A PL-2 type air-cooled condenser is used to condense and sub-cool the steam. The condensate drains to a hotwell, and with the feedwater booster pump is returned to the suction of the SL-1 feedwater pumps. Twin air ejectors are provided to remove non-condensable gases from the condenser and the hotwell. Steam from these units is condensed in an air ejector condenser cooled by the feedwater from the booster pumps.

7. Nuclear Instrumentation

The nuclear instrumentation system is composed of startup instrumentation, containing source range and intermediate range equipment, and power range equipment utilized during power operation to monitor reactor neutron flux level and provide over power protection.

The present SL-1 installation uses two boron trifluoride counters with scaler readout for the source range channels. These startup channels provide indication only, with no automatic reactor protection.

Two compensated ion chamber channels are utilized during startup in the intermediate range. One channel provides linear readout (indicating and recording) with no automatic reactor protection. The other channel provides log readout (indicating and recording) with automatic period protection for the reactor. These two channels will operate over the intermediate flux range and the power range.

Signals for reactor over-power protection are generated from two uncompensated ion chamber channels. Meter relay trip circuits and indicated neutron flux readout is available at the control panel for these power range channels.

8. Process Instrumentation

Process instrumentation signals are used for indicating or recording the plant parameters. Feedwater flow, reactor steam pressure, main steam flow, by-pass steam flow, condenser vacuum, condenser air in and out temperature, feedwater temperature and reactor water level are recorded on the main process panel. Feedwater pressure, main steam pressure, hotwell level, main steam pressure, $P-P_o$, system temperatures (48 points) and conductivity are indicated.

9. Control Systems

The SL-1 control systems include detectors, controllers and actuators which use vacuum tubes and slide wires.

a. Reactor Control

The steam void coefficient of reactivity acts on the reactor to move the reactor power in a direction opposite to that required to follow a load change. The SL-1 reactor control system accommodates load changes by adjusting the reactor thermal output to the level of the load or by by-passing steam to keep the output of the reactor constant during load changes. A pressure signal (P) from the main steam line is fed to the pressure deviation recorder where it is compared with the pressure reference setting (P_o). From the pressure deviation recorder a signal proportional to $P-P_o$ is retransmitted by means of a slide wire to a position controller. For the first control mode the controller drives the center control rod to increase or decrease reactor power as required. For the second control mode the controller operates a valve in the steam by-pass line around the turbine.

b. Reactor Water Level Control

There are two displacement float liquid level sensor transmitters in the reactor. The signal from the first is used for recording liquid level, controlling the feedwater regulating valve and for high and low level alarms. The signal from the second is used to give a high and low liquid level scram.

Since the steam flow from the reactor will vary with time, it is necessary to control the flow of feedwater to the reactor to maintain the reactor water level within the desired limits. Either of two methods are available for controlling the feedwater regulating valve in the feedwater line. For the first method, three-point control, signals from the steam flow, reactor water level and the feedwater flow are combined and fed into the controller which actuates the flow regulating valve. In this controller the reactor water level signal is over-riding. For the second method, single point control, the reactor water level signal alone is fed to the controller.

II. OPERATION OF THE SL-1 POWER PLANT FACILITY BY COMBUSTION ENGINEERING, INC.

A. CONTRACTUAL AGREEMENT

Contract No. AT(10-1)-967, as amended, between Combustion Engineering, Inc. and the Atomic Energy Commission is for the term between December 14, 1958 and September 30, 1962. It is a cost-plus-a-fixed-fee contract for operation of the reactor and for the performance of research and development work at Combustion Engineering's plant in Windsor, Connecticut. The objectives of the contract are:

(a) to gain, through SL-1 plant operations:

  (i) data and experience at design and off-design conditions in support of the Army Boiling Water Reactor Program.

  (ii) knowledge of the costs of operating the SL-1 on both a commercial and a Government-accounting basis.

  (iii) familiarity with the problem areas encountered through sustained operation.

(b) to train and assist others in training crews to operate the SL-1 and other reactor installations.

The contract is administered by the Idaho Operations Office, AEC-with the day-to-day administration (through the time of the incident) being carried on by the Military Reactors Division of that Office. Military personnel from three services (Army, Navy, and Air Force) were assigned to the SL-1 as operating crew and for training. Such personnel performed operational and maintenance functions under the over-all management and technical direction of Combustion Engineering, Inc.

B. OPERATIONS

1. Transfer from Argonne National Laboratory

"The Argonne Low Power Reactor (ALPR) was designed as a prototype of a low power, boiling-water-reactor plant to be used in geographically remote locations.[1 & 2] Upon completion of construction of the ALPR at the"National Reactor Testing Stations, near Arco, Idaho, "Zero power" testing of the plant in general and the reactor itself was carried out by engineers and scientists of the Argonne National Laboratory with a very considerable participation by personnel from a military cadre assigned to the facility.[2] A program of reactor physics experiments at very low power was begun on August 11, 1958, on which date the first critical loading was attained; this program was completed approximately two months later. Shortly after this, on October 24, 1958, the reactor was brought to its operating conditions of pressure (300 psig) and temperature ($\sim 420^{\circ}F$) by nuclear heating for the first time, and it was operated at essentially normal maximum load demand. When various test programs and a 500-hour sustained power run had been completed, the operation of the ALPR was phased out to Combustion Engineering, Inc. and the facility has been operated by that company since February 5, 1959. The plant was relabeled SL-1 in accordance with its place in the Army Reactor Program."[3]

During the month of January, 1959, preceding Combustion's assumption of operating responsibility, the Combustion Engineering staff worked side by side with both the ANL personnel and the military personnel in performing plant operations and maintenance. Two members of Combustion's staff were already qualified and experienced reactor operators and another was qualified and experienced on operation of power plant systems. The ANL staff checked them out as operators on SL-1. All members of the Combustion staff attended the military classroom and operating instruction sessions to familiarize themselves with both plant details and with content of the military instruction program. There was full mutual understanding and cooperation during this period between all three parties and the Commission.

At the time of the actual transfer to Combustion Engineering, Inc. a minimum number of documents were handed to Combustion Engineering, Inc. These were: a) "Hazards Summary Report on the Argonne Low Power Reactor (ALPR)" - ANL 5744; b) a set of plant drawings (not up-to-date); c) a brief Standard Operating Procedure for the SL-1 Reactor. Later during the calendar year 1959 additional reports were issued: a) "Initial Testing and Operation of ALPR" - ANL 6084; b) Final Specifications for Government Purchased Equipment for ALPR; c) "Recommendations for Improved Operation of ALPR."

The operating organization under Combustion Engineering at the time of transfer (February 5, 1959) was identical to that used by ANL.

	ANL	Combustion Engineering
Operations Supervisor	Wallin	Crudele
Reactor Engineering	Thie	Canfield
Power Plant Engineer	Cerchione	Rausch
Health Physicist	Stoddard	Vallario

Additional positions in the Combustion Engineering organization included: SL-1 Project Manager, W. B. Allred; Test Supervisor, L. E. Anderson; and Chemist, Glynn. (See Chart A)

The military cadre organization in existence during ANL's operation of the reactor was continued "as is" under Combustion Engineering.

2. Operating Organization and Procedures

In reviewing the operating organization and procedures - written and verbal - it must be borne in mind that one is dealing with a "field test" operation. According to, and in compliance with the contract, the SL-1 plant was used as a developmental test and training facility. Through plant operation, performance data at design and off-design conditions were obtained and used as a basis for design of advanced plants.[4]

The operating organization under Combustion Engineering at the time of assuming responsibility for operating SL-1 and the subsequent modifications are indicated in the accompanying Charts A, B and C. The military organization has remained the same since its original inception. Under this concept operating orders and all other necessary information was given to the plant superintendent - a military man - who in turn passed them down to the shift supervisor (military).

Authority and responsibility for operating action was delegated, verbally or in writing, to Combustion Engineering by the Commission. In turn, Combustion, verbally or in writing (such as; Operating Procedures and Standing Orders) passed the responsibility for implementing action on to the military personnel.

The operating crews for SL-1 were composed entirely of military personnel (SL-1 Cadre). The Cadre was responsible, under the direction of the contractor (CE), for operating the plant. The Plant Superintendent (Cadre) was responsible to the Operations Supervisor for the safe and efficient operation of the plant as well as the performance of maintenance on the reactor, plant, and associated equipment. In addition, he evaluated

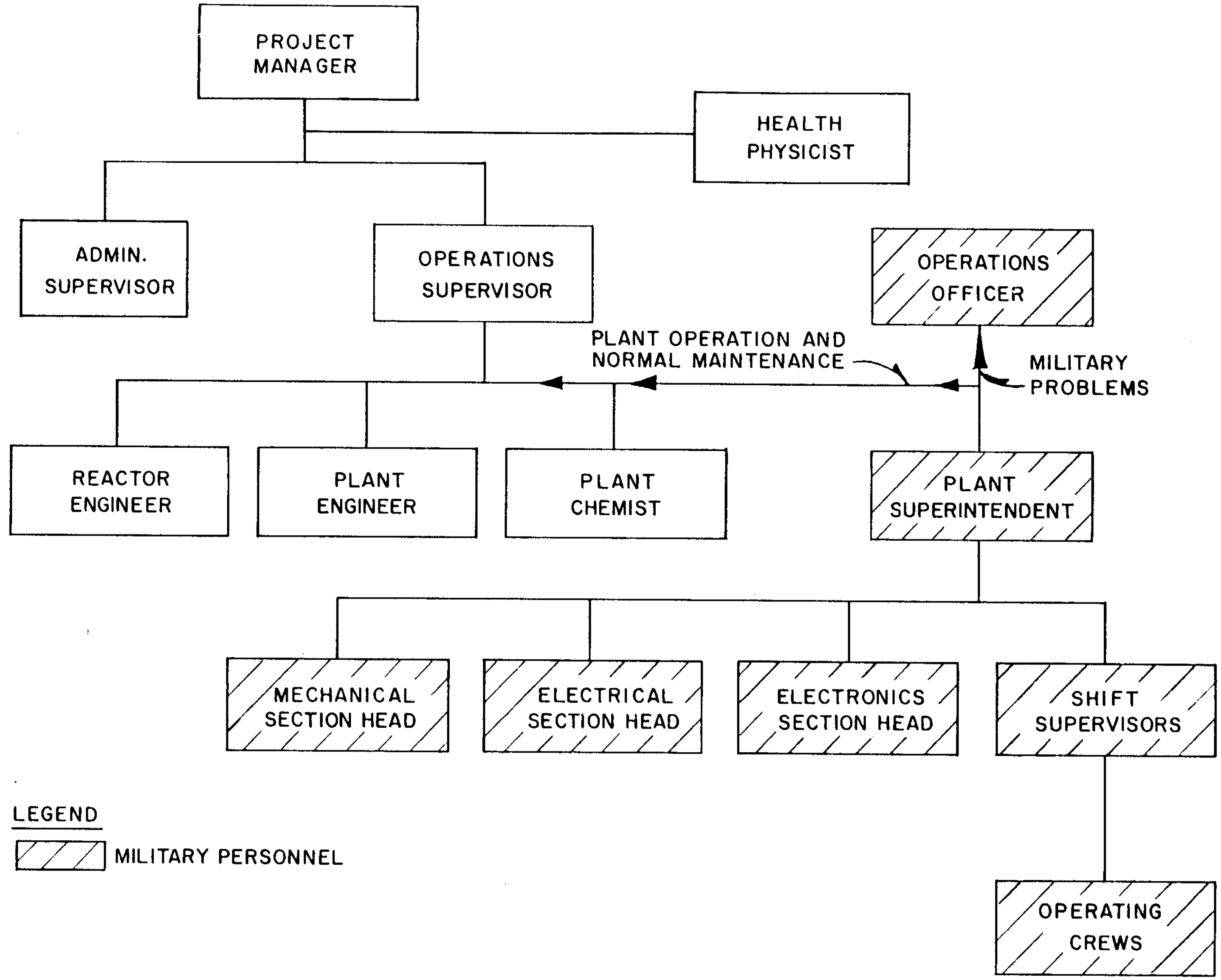


CHART A SL-1 ORGANIZATION
FEB. 1959

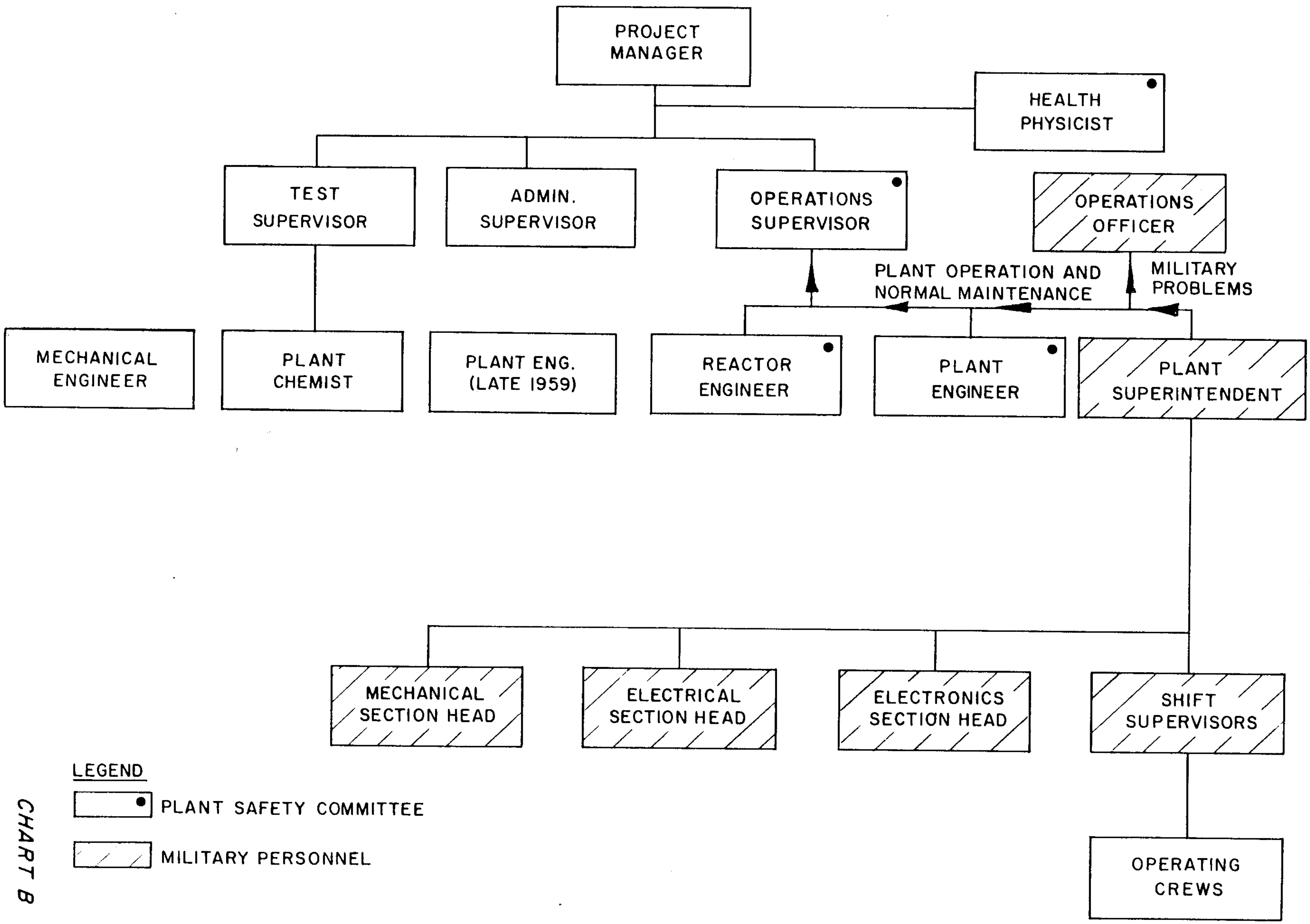


CHART B SL-1 SITE ORGANIZATION ABOUT APRIL 1959

CHART B

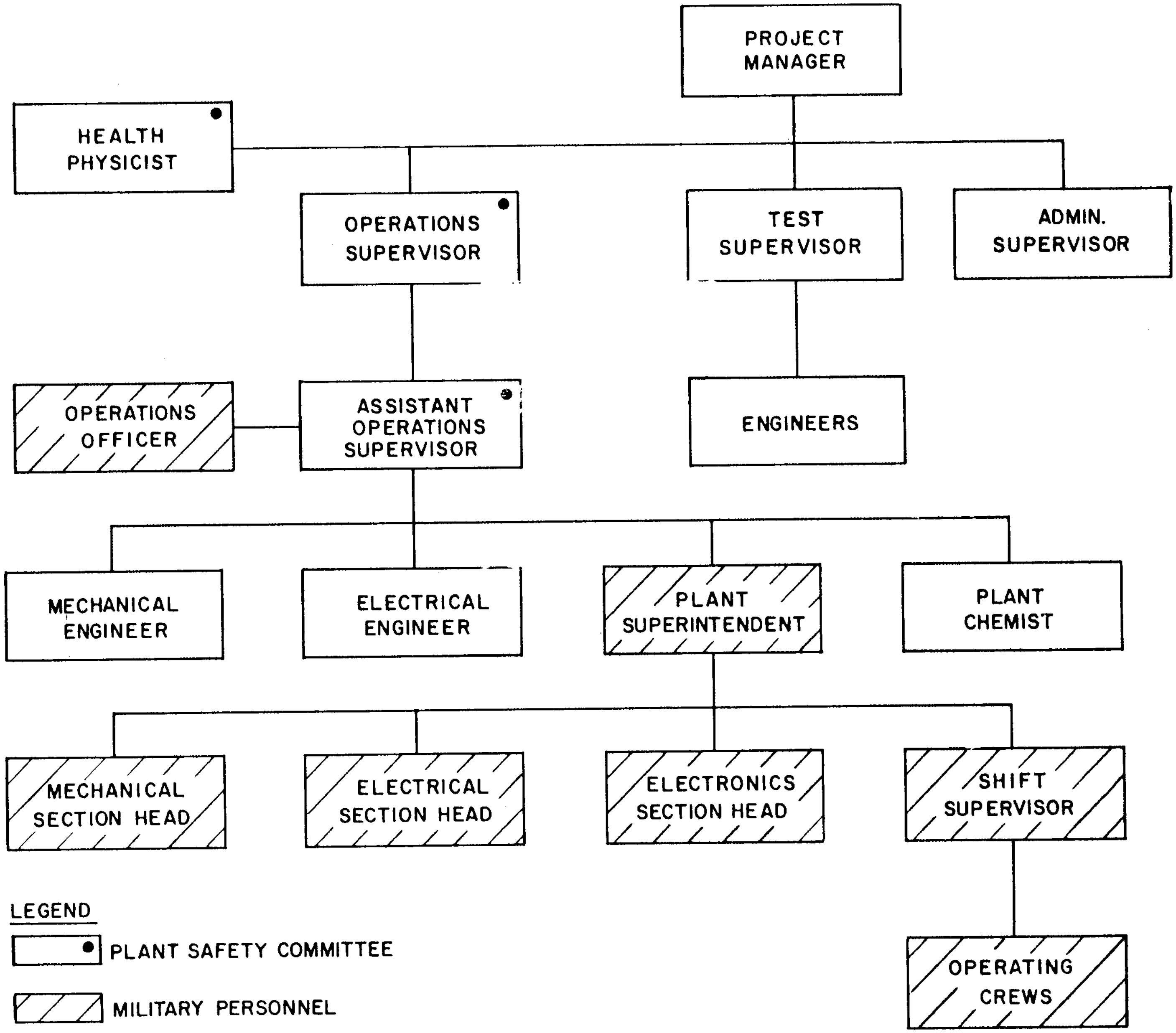


CHART C SL-1 SITE ORGANIZATION
FALL 1960

CHART C

the qualifications of operating crews and assisted in setting schedules.

The Operating personnel operated the plant on a continuous rotating shift basis. Originally crews were organized into three-man shifts during plant operation, with those personnel not on shift work performing maintenance duties during the day shift. Each three-man crew was composed of a Chief Operator and two Operators. The Chief Operator supervised reactor and plant operations, plant maintenance and Health Physics activities, and was responsible for the safe and efficient operation during his shift.

In order to satisfy military criteria to determine the minimum operating crew necessary, the shift crews were reduced to two-men in September, 1959. Each two-man shift was composed of a Chief Operator and and an Operator. This reduction was approved by Combustion Engineering, the military and the Commission and proved satisfactory. During the performance of the test programs approved by the Commission, Combustion Engineering personnel supervised and operated the reactor.

Under the provisions of the contract, Combustion Engineering was assigned responsibility to manage and operate the reactor on the basis of around-the-clock capability. In the ABWR program proposal submitted to the Commission(5) in September, 1960, the addition of two staff personnel was proposed to provide Combustion Engineering supervision on all shifts. The Commission requested that Combustion Engineering delete our request for supervision on all shifts and therefore no staff additions are included in the final approved program.(6) A letter was sent to the Director of Military Reactors on November 29, 1960, reflecting the fact that at the instructions of the Commission, the military supervised all shift operation (routine) and maintenance and requesting the Commission's specific confirmation of this arrangement. A written response had not been received at the time of the incident.

The operating procedure manual, prepared by ANL, provided less than minimal information. Combustion Engineering was requested by the Commission to prepare a "complete operating manual" for full scale operation of the plant. The Commission furthermore furnished "an outline for the chapters on individual systems," and each completed segment of the manual was forwarded to them for approval.(7) The reactor was operated on a limited basis during this period to obtain data and information for the writing of the manual. Each such operation required specific approval from the Director, Division of Military Reactors. The manual prepared by Combustion Engineering was approved in March, 1959. This manual of operating procedures was subject to continuous revision based on operating experience.(8 & 9) Such revisions were made periodically and in September, 1960, Volume II, Operating Procedures, of a new, completely revised SL-1 Operating Manual was submitted in draft form to the Commission for review and comments. Volume I covering Reactor and System Descriptions was to follow early in 1961. Subsequently Combustion Engineering was to re-submit the manual for final approval.

To conduct plant maintenance Combustion Engineering and the military initiated preparation of maintenance procedures. Combustion Engineering reviewed, expanded, and issued these as part of the operating manual. The Commission did not require a maintenance manual. Initially all maintenance was accomplished during the day shift. In the summer of 1960 the practice of performing maintenance on each shift was initiated to utilize personnel and time more effectively. The plant superintendent, under whose direct supervision the maintenance had been performed on the day shift, assigned the specific maintenance operations to the Chief Operator of each shift. These assignments were reviewed and approved by Combustion Engineering personnel.

The SL-1 Safety Committee served as an advisory committee to the Project Manager. Its function was to review procedures and plant changes and make recommendations as required to the Project Manager. The Committee met as required, and such meetings were called by the Committee Chairman. The Committee functioned as a working group with members conducting investigations in their own specialties. All work was not necessarily done in formal committee meetings. The military was represented in an ex-officio manner by the Military Operations Officer. Specific instructions to the Cadre operating crews on safety or other matters were transmitted through normal channels, i.e., the SL-1 Operations Supervisor. Minutes of meetings were kept.

The Combustion Engineering, Nuclear Division, Safety Committee at Windsor, Connecticut, was brought into specific SL-1 problems at the request of the SL-1 Project Manager, or of the Nuclear Division management.

Examples of referrals to the Division Safety Committee are as follows:

(1) Review of SL-1 Operating procedures and plant design - March, 1959

(2) Review of SL-1 Malfunction Report #7 (loss of water) - December, 1959

(3) Review of SL-1 plant expansion hazards evaluation - June, 1960

(4) Review of loss of boron from the SL-1 core - November, 1960

3. Training

The administrative arrangement under which Combustion Engineering worked with the military for training crews was limited to operator training within the plant under the general direction of Combustion Engineering. The single exception to this was the health physics training in which Combustion Engineering provided both the classroom and the operational training. In addition, Combustion Engineering had reviewed the military's training program for content and had concurred in the adequacy of the training material for operation of SL-1. This step was undertaken since Combustion Engineering later qualified personnel to operate the reactor and was therefore required to be familiar with training material.

Training literature was in use at the time of SL-1 transfer to Combustion Engineering. This literature had been prepared by the military under the guidance of a training officer to conform with their procedural standards. Additional or modified material to be incorporated into the training literature was regularly being up-dated and revised during the two years that the SL-1 was being operated under Combustion Engineering supervision. It should be noted that the responsibility for preparing this literature was entirely a military function, however, the military did request informally that Combustion Engineering review the literature and provide technical comments. Combustion Engineering did review the material as requested and did provide technical comments and design information. A major training item was the SL-1 Operating Manual.

Although Combustion Engineering was not responsible for the establishment of the military training program and was never requested to conduct a formal review of the program, it did consider the training to be adequate. The military training program was formal in content and presentation and was based on up-to-date information.

The Military Training Program ran as follows:

For Operator:

(a) Arrived at SL-1 following basic academic training at Fort Belvoir, Virginia, which is conducted by the Military.

(b) Sixteen week training by SL-1 military Cadre including operational experience on reactor.

(c) Written examination given by military instructors.

(d) Oral examination given by military training review board.

(e) Oral examination given by Combustion Engineering Assistant Operation, Supervisor.

Chief Operator:

(a) At least six months operation of SL-1 plant as qualified operator.

(b) Written examination given by military instructors.

(c) Oral examination given by military; attended by all section heads, the training group, and plant superintendent.

(d) Oral examination given by Combustion Engineering board:

CEI Health Physicist
CEI Operations Supervisor
CEI Assistant Operation Supervisor
CEI Physicist
Military CO or Operations Officer

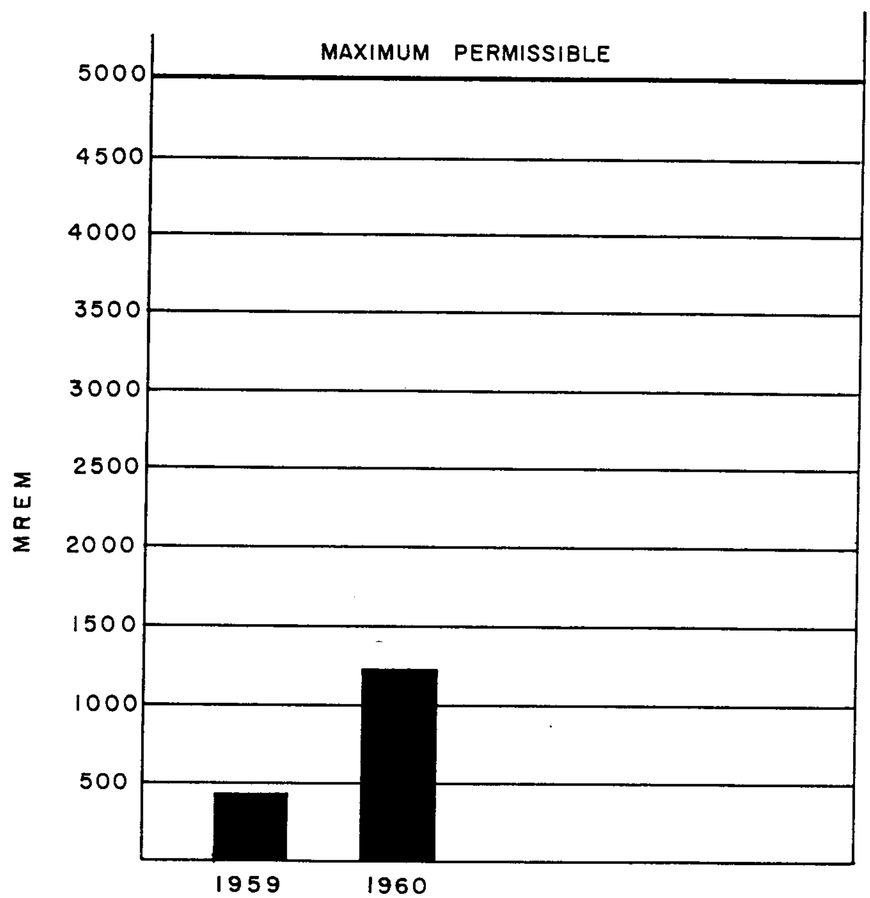
During the Combustion Engineering examinations for operators, approximately 40% of the candidates did not pass initially. In all cases the man, training group and his supervisor were informed of his failing points. Following a specified time period (during which the man was retrained and re-recommended by the military) he was re-examined by Combustion Engineering. All men were finally passed and qualified by Combustion Engineering. One man took three examination. Of the operators possessing the necessary time qualification, only 13 were selected for Chief Operator training, and all 13 qualified.

The Cadre selected their instructors from Chief Operators who had an outstanding aptitude for training. The men were selected by the SL-1 Cadre Chief based on his evaluation of their general knowledge of the plant as well as other factors. Combustion Engineering did not participate in this selection.

4. Radiation Exposure History

Combustion Engineering published "SL-1 (ALPR) Health Physics and Safety Procedures" in February, 1959. This manual was subsequently revised in December, 1960. The basic guideline for these manuals was the "Code of Federal Regulation, Title 10, Chapter I, Part 20, Standards for Protection Against Radiation," and the National Bureau of Standards Handbook 69."

The effectiveness of the health physics procedures is demonstrated by the fact that personnel exposures were maintained at a consistent, low level over the two-year period, January 1959 through December, 1960 (Chart D). The highest whole body accumlated exposure in any one quarter was 390 mrem. In 1959, the highest whole body accumulated exposure was 410 mrem. The highest whole body accumulated exposure for 1960 was less than 1200 mrem. Thirty-one of the forty-one personnel at the SL-1 during the calendar year 1960 received less than 1000 mrem. In the two-year period there was never an over-exposure and the highest radiation exposure was at least a factor of four below the maximum permissible.



SL-1 OPERATIONS HIGHEST ACCUMULATED PERSONNEL EXPOSURE - CHART D

C. REACTOR OPERATING EXPERIENCE AND EVALUATION

1. Metallurgical History of the Core

a. Fuel Elements

The SL-1 fuel plates were fabricated[10] by standard roll bonding techniques modified to include silicon bonding. This process was developed by ANL to improve bonding between the clad and fuel bearing alloy. The fuel alloy, containing approximately 17.5 w/o U, 2 w/o Ni and 81.5 w/o Al, was induction melted at 100 micron pressure and cast into 2 in. x 4 in. x 18 in. billets which were cross-rolled to 11 in. to 12 in. wide and long-rolled to 0.200 in. in thickness.

The fuel fillers were placed into X-8001 aluminum picture frames and clad with X-8001 cover plates by means of the silicon bonding process. This process consisted of spraying the top and bottom cover plates with glycerine and then with silicon powder. The entire fuel plate assembly was then hot pressed at 1500 psi at a temperature of 1100°F. Fuel plates were hot rolled in air at 980°F to three 25 per cent reductions which were followed by one cold pass. Elements were then annealed at 1020°F for one hour and inspected for blisters. Acceptable plates were then radiographed, sheared to size, pickled and ultrasonically inspected for non-bonds.

Assembly of nine individual fuel plates into a subassembly was accomplished by forming right angle bends along each fuel plate edge and spot welding these edge flanges to the X-8001 aluminum side plates. Radiographs were retaken after the flanges were formed and before spot welding. Extruded end fittings were heliarc, hand welded at the top and bottom of each of the fuel assemblies.

Information on radiation damage studies for U-Al fuels has been generally confined to MTR fuels under MTR conditions. An analysis of MTR data[11] indicates that over 1,000 fuel elements were irradiated to average U-235 burnups of up to 30 per cent (approximately 0.7 atom per cent burnup) and that elements with higher U loadings were irradiated to 85 per cent burnup of the U-235 atoms (approximately 2 atom per cent burnup). No failures were observed which could be attributed to radiation damage.

Extrapolation of the above data to SL-1 conditions was questionable because the difference in fuel temperatures of 250° to $300^{\circ}F$ represented a change in the ratio of operating temperature to melting temperature from 0.4 under MTR conditions to 0.6 under anticipated SL-1 conditions. There was a potential transition from non-damaging to damaging effects in extrapolating MTR data to SL-1 conditions because the critical ratio appears to be about 0.5.(12) An analysis of the radiation damage resistance of SL-1 fuel elements under operation at 3 MW has been presented in Reference 17.

An irradiation test of a prototype SL-1 fuel plate in the ANL-2 loop in MTR(13) was terminated after 227 loop days (156 MTR full power days) because of fission product activity in the loop. Although the maximum burnup of 1.3 atom per cent at the high flux end of the plate was roughly equivalent to that anticipated for SL-1, the peak heat flux, coolant velocity, and ratio of coolant volume to surface area were all greater than those encountered in SL-1. Post-irradiation examination of the fuel plate revealed two cladding failures in the form of erupted corrosion pits located four to five inches from the high flux end of the plate. Swelling was observed for a length of 2-1/4 inches from the high flux end of the fuel. A clad failure was not found in this region. It was postulated by ANL that the erupted corrosion pits were a result of loop operation at a high pH for a period of ten days prior to the detection of activity in the loop.

Although swelling was attributed to the combination of high burnup and high temperature (estimated at $1000^{\circ}F$), the latter was based upon assumptions of corrosion film thickness, rate of build-up of the film, and the presence of a thick film during the lifetime of the experiment. Consideration of all these factors led the authors to state,(13) "Considering the range of thickness values and the uncertainty of the conductivity of the scale, it is impossible to calculate fuel temperatures with any degree of accuracy." The temperature has not been calculated for the region 2-1/2 inches from the high flux end of the plate, beyond which swelling was not observed.

Recently, ANL completed(14) an irradiation of a second SL-1 prototype fuel element in the ANL-2 loop. There was no evidence of swelling after

0.92 atom per cent burnup, based on thickness measurements at the high burnup end. Examination of the element in the ANL hot cell has not been completed. The peak burnup in SL-1 at the time of the incident was 0.53 a/o, which is less than 60% of the burnup in this second test.

Because irradiation data were lacking for the SL-1 fuel under design conditions, several methods for estimating the probable swelling temperature were used. Based upon probable temperatures for fission gas mobility,[12] creep and recrystallization relationships,[15] Wyatt's swelling equation,[16] as well as the belief that the irradiated fuel alloy would exhibit properties somewhat analogous to a dispersion hardened aluminum system, a swelling temperature of 475°F was estimated[17] for the end of life condition in SL-1. At that time, the best estimate of the maximum SL-1 fuel temperature using corrosion film temperature drops based on Hanford and Chalk River[18] data, was 542°F.

Measurements of the maximum fuel temperature in the SL-1 core were made in August 1960 by means of an instrumented fuel assembly. Maximum temperatures of 442° and 457°F, for 3 MW and 4.5 MW, respectively, were measured. Because both temperatures were below the calculated swelling temperature, it was unlikely that swelling would have occurred in the SL-1 fuel by the end of life. Also tests run at Combustion Engineering on X-8001 aluminum tubes in autoclaves to determine the effect of corrosion film on temperature indicated temperature drops which closely approximated those calculated from boiling film temperature drops. No additional ΔT was measured which could be attributed to the build-up of a corrosion film.

The first instrumented (thermocouples) fuel assembly test[19] was discontinued on September 25, 1959 after it was observed that only bulk water temperatures were being measured. This instrumented fuel assembly, as well as fuel assembly #38, were placed in the SL-1 storage well during the October 3, 1959 shutdown.

A second fuel assembly was instrumented with thermocouples.[20] This test assembly was in the reactor at the time of the incident. Spare fuel element No. 1 was taken from the storage facility in Idaho for this test. Twelve holes were drilled into the end fuel plate as shown in Figure 11. The holes were thoroughly cleaned with acetone, rinsed with distilled water and dried.

Magnesium-oxide insulated chromel-alumel thermocouples, twenty-five feet long, were used for the instrumentation. They were clad with 0.066 in. O. D. Type 347 stainless steel tubing with 0.008 in. wall thickness. The hot junctions were formed by fusing the chromel and alumel wires into a solid mass with the clad material, thus insuring good conduction. The thermocouples were tested for continuity between leads, resistance between leads and helium leak tightness. The hot junctions were radiographed and calibrated three separate times between the temperatures of 400° and 600°F at 50°F intervals.

Eight of the above thermocouples were inserted in the fuel plate. Four of them were brazed at the mouth of the hole to prevent water from entering the annulus. On the other four, the annulus was peened shut around the thermocouple sheath.

Four special thermocouples were prepared for the remaining four holes in the fuel plate. On these four, the cladding was stripped back from the thermocouple wires a distance equal to the depth of the hole. The wires were insulated through the hole with MgO insulating beads. A hot junction was formed on the end and the thermocouple inserted so that it was in contact with the bottom of the hole. The sheath was brazed at the mouth of the hole to prevent water from entering the annulus. One of this type (No. 6) was removed and the hole brazed shut when an unrepairable sheath crack occurred during fabrication. The location of each type of thermocouple is shown in Figure 11.

A half-length boron-aluminum strip was formed into a guide or cover plate for the thermocouples going up the side of the fuel element. The eleven leads were safety-wired into a stalk as they emerged from the cover plate to provide for rigidity and ease of routing in the reactor. A photograph of the test assembly is shown in Figure 12.

A flame was traversed along the full length of each thermocouple just prior to insertion in the reactor to show that only one hot junction existed in each case. Thermocouple No. 8 was found defective. During insertion in the reactor thermocouple No. 7 became erratic, consequently numbers 6, 7 and 8 were not usable.

During the week of August 16, 1960, fuel element No. 42 was removed from position No. 55 and the instrumented element No. 1 was inserted. Results of the temperature measurements are presented in Figures 13 and 14.

Two periodic visual examinations of a number of SL-1 fuel assemblies were made prior to the incident. The first fuel element inspection was performed on August 27 and 28, 1959. The following fuel elements were withdrawn[19] for visual inspection:

Fuel Element No.	Position No.
6	45
19	85
59	75
3	46
14	57

The general appearance of the fuel elements inspected was good and the corrosion rates were found to be low. Fingerprints were still visible.[19] If corrosion had been excessive, all evidence of fingerprints would have disappeared.

During the period August 13 to August 21, 1960, a second visual examination of fuel elements was made[21] by placing them so that they could be viewed through control rod ports. Because of the extent of the aluminum-boron strip deterioration which made it difficult to remove the fuel assemblies, visual inspection was terminated after three fuel assemblies had been viewed. The fuel plates in the assemblies observed (No. 19, No. 59, and No. 42) did not show any sign of warpage, swelling, distortion or corrosion. It was concluded [21] that corrosion or film build-up had not changed noticeably since the last inspection. These visual observations indicating no corrosion film build-up corroborate maximum fuel temperature measurements obtained in the second instrumented fuel assembly test.

Fuel assembly No. 42 removed from the core to permit insertion of the second instrumented fuel assembly, is now in the AREA hot cell facility. Prior to the incident, visual examination, which had been completed, indicated no abnormalities or unusual appearances. Localized thin brown areas located on the non-active side plates were observed and were believed to be iron oxide. Macrographs have been taken at a number of angles. Channel gap measurements were being completed at the time of the accident. Additional operations will include cutting of side plates to free active fuel plates for additional measurements, metallographic examination for signs of fission gas bubble agglomeration as well as measurements of corrosion film for correlation with data obtained from the instrumented fuel assembly. Burnup analyses will be made on sections adjacent to the metallographic samples.

b. Boron-Aluminum Poison Strips

Attempts to incorporate boron into the fuel bearing portion of the fuel plates were unsuccessful. As a substitute, strips of X-8001 aluminum containing either 0.5 w/o B^{10}, or 0.4 w/o B^{10} (for full and half strips, respectively) were spot welded to the sides of fuel assemblies (93% enriched B^{10} was used). Each of 16 central fuel assemblies had one full length strip and one half strip; the remaining assemblies had a full strip only. The aluminum-boron strips were fabricated[17] by extruding, at 850°F, a billet containing a powder mixture of X-8001 aluminum and B^{10} powder, clad in X-8001. The extruded sections were cold rolled to final thickness, annealed, cut and then spot welded to the side plates of the fuel assemblies. Although it was assumed that these plates would still be clad with X-8001 after extrusion, metallographic examination of sections of an unused plate, as well as sections of recovered aluminum-boron strips, do not have complete cladding. It is certain that the edges and ends were unclad. Spot welding would expose aluminum-boron meat even if cladding were present.

Bowing of the aluminum-boron strips was observed during the first fuel element inspection on August 27 and 28, 1959. Bowing between spot welds of about 0.080 in. was measured. Contact between the strip and the shroud during withdrawal was indicated by scratches along the bowed portion of the poison strips.

During the second visual inspection of fuel elements in the period August 13 to 21, 1960, the bowing on one of the peripheral fuel assemblies had increased to 0.170 in. as measured on one plate in one spot (see Fig. 15 and 16). This visual inspection was stopped after removal of assebmly No. 42 from position No. 55 indicated a complete loss of the aluminum-boron strips from this assembly and one from an adjacent assembly. Removal of element No. 42 was achieved with much difficulty and resulted in tearing loose the remaining solid portions of the aluminum-boron strips. Photographs of the strip remnants and fragments recovered from the bottom of the core are shown in Figures 17 and 18. It was originally concluded that these aluminum-boron strip remnants were both from fuel assembly No. 42, however, a subsequent closer examination showed the identification numbers of No. 35 and No. 42. Thus, the remnants were from the top portion of the strips that had been on fuel elements No. 8 and No. 42 respectively (see Fig. 21). The half strip, No. 80, from fuel element No. 42 appears to be completely gone. Since the greatest degree of deterioration has always been observed on the central fuel elements, the mechanism appears to be burnup dependent.

After the above observations had been made, a program for the evaluation of the aluminum-boron strip remnants was formulated. This program was being pursued at the time of the incident. A summary of important facets of this program is presented below:

(1) Thickness traverses are being made of poison strip remnants to obtain data on possible swelling due to helium generation.

(2) Macro-examination is being made of all surface areas; abnormalities will be noted and macrographs will be taken.

(3) Metallographic samples are being taken from high and low burnup regions in search of indications, such as gas generation around borides and local corrosion, which might demonstrate a burnup dependency.

(4) Corrosion tests are being made of sections taken from high and low burnup regions in an attempt to evaluate the effect of burnup on corrosion.

(5) B^{10} burnups are being established by isotopic analysis for areas adjacent to samples in (3) and (4) above. B^{10} analyses were also requested for particulate residue taken from the bottom of the reactor.

At this time dimensions have been obtained and are being analyzed. A series of photomicrographs taken at various locations in the recovered strip sections are being evaluated.

c. Cadmium Shims

The inferred loss of boron poison was to be compensated for by the addition of cadmium poison shims to two of the Tee rod positions. Six cadmium shims were installed, three of each, into Tee slots of rod positions No. 2 and No. 6 on November 15, 1960.

An as-built drawing of a cadmium shim assembly is shown in Figure 19. A modified picture frame technique was used in which three cadmium sheets, 0.02 in. x 4-13/16 in. x 29 in., were placed on top of each other to build the thickness up to 0.060 in. The 2S aluminum picture frame was composed of side strips which were 11/32 in. wide x 30-1/8 in. long x 1/16 in. thick. The bottom and top strips of the aluminum picture frame were 4-3/16 in. long x 1/8 in. wide x 1/16 in. thick and 4-3/16 in. long x 1 in. wide x 1/16 in. thick respectively. The 2S aluminum cover plates were 30-1/8 in. long x 5-1/2 in. wide x 1/16 in. thick. The cadmium was completely enclosed by means of a heliarc fusion weld along all top, bottom and side edges. Two, 0.040 in. diameter, vent holes were drilled on each face of the shims, through the cladding and into the cadmium. All connections were heliarc welded.

d. Control Rods

Cruciform shaped control rods containing 0.060 in. thick cadmium sheets clad with X-8001 aluminum were assembled [17] so that bonding would not exist between the cadmium and the aluminum. Prior to assembly, the X-8001 cladding plates were dimpled and then bent into 90° angle sheets. These dimples, in X-8001 on opposite sides of the cadmium sheets were spot welded together through 0.5 in. diameter holes in the cadmium sheet. The four 90° angle aluminum cladding sheets were welded at the edges to provide a continuous seal. Bottom and top X-8001 extensions were welded to the cruciform type control rods in a tongue and groove configuration in which reduced sections of the extensions were placed between control rod cladding and welded to the cladding[22]. Vent holes are provided at the top of the control rod blade. Venting is provided along the entire length of the active section of the control rod by the dimensional mismatch between the squared cadmium and curved aluminum at the inside corners of the cruciform.

e. Test Specimens in the Reactor

Corrosion tests of Ag-In-Cd for PL-2 control rods were performed in the SL-1 reactor in two series. The first series of six unclad specimens were fastened to two aluminum holders and positioned[20] in a dummy fuel element which was inserted into position No. 27 in July, 1960. Three of these specimens were in the reactor at the time of the incident. The samples and their holders are shown in Figure 20.

A second set of nine stainless steel clad, Ag-In-Cd corrosion coupons was positioned in groups of three in each of three stainless steel holders in November, 1960. These holders were placed on the gripper tips of each of three central fuel assemblies in SL-1. Two sets of these corrosion specimens were in the reactor at the time of the incident.

2. Reactivity History and Analysis of the Core

a. SL-1 Nuclear Evaluation

Shortly after Combustion Engineering assumed operational responsibility for the SL-1, a three-month analytical evaluation of the reactor was conducted. The purposes of this program were to obtain estimates of the reactivity margins of the core, and to estimate its capabilities in terms of the design requirements. A detailed description of the nuclear evaluation is contained in IDO 19003 - "SL-1 Reactor Evaluation" (July 15, 1959), which is summarized here.

Considering the arrangement of materials in the SL-1 core, it is evident that the many heterogeneities in the core introduced a considerable complexity into the analysis. The flux distribution and reactivity associated with a "control cell" depend markedly on whether or not control rods are inserted or whether followers are inserted. A control cell is defined as a region of the core bounded by the center lines of the control rod channel, and includes four fuel boxes, a variety of boron containing plates, shrouding, water gaps, and control rods, followers or the water in the rod scabbard. A control cell near the center of the core has four or eight aluminum-boron strips depending on whether it is in the upper or lower half of the core, respectively. The lower central region may have one rod follower inserted (the 17-inch central follower) and has a higher boron loading than the outer region of the core. The control rods extend into the reflector region, thus requiring consideration of a rodded and unrodded reflector. In summary, although the concept of a control cell is useful in the analysis, there is no cell region of the core which can be defined as "typical." Hence, considerable simplification and a number of approximations were required in the homogenization of the core materials which are described below.

The first step in the nuclear analysis was to calculate several experimentally observed critical configurations of the core. The critical rod bank positions under cold (83°F), hot, zero power (420°F) and operating (2.56 mw, no xenon) conditions were measured and the reactivity of the core under these conditions was calculated. The analysis was conducted using two neutron groups. Group averaged cross sections were obtained by means of the MUFT IV code for one fast neutron group. Thermal cross sections were averaged over a Maxwellian spectrum of thermal neutrons. Homogenization of the materials in a control cell for the thermal group was carried out by computing the thermal flux distribution in the cell with all components represented explicitly. The number densities were then flux weighted in such a manner that the total number of thermal absorptions in each material in the homogenized region was the same as the number of absorptions in that material with the explicit representation. The thermal flux distributions were computed by means of the SIMPL code in one dimensional geometry using a double P-1 approximation. This was done first for a single fuel plate, and the associated coolant channel. With a homogenized version of the fuel region the process was repeated for a control cell which included control rods, followers, or water, boron plates, structural material and water gaps-all treated explicitly. The homogenization was carried out for the large variety of core regions under cold and hot conditions, and with various vapor volume fractions.

The criticality of the core at the measured rod bank positions was computed using two-group diffusion theory in cylindrical geometry by means of the PDQ code in R-Z geometry. Fast and thermal core constants were obtained as described above. The control rods were represented as a homogeneous poison of such magnitude as to yield the same eigenvalue in a circular core cross section as in an explicit X-Y PDQ calculation. The regions of the reactor were treated as cylinders, or cylindrical annuli, in such a manner as to conserve core volume, and the rod poison extended over the observed rodded region of the core and the reflector. The eigenvalues resulting from the analysis were 1.034, 1.020 and 1.030 for the cold, hot and operating conditions respectively. These biases in the calculations can be attributed to the homogenization and the treatment of this high leakage core as a cylinder. Probably the major uncertainties of the homogenization procedure are first, the one-dimensional treatment, since the disposition of boron in the cell is not adequately represented in one dimension, and second, the treatment of fuel regions near the reflector as cells even though they are adjacent to a large water region.

Using these results as an indication of the errors inherent in the calculational scheme, subsequent results were corrected for these biases. Table I shows eigenvalues and reactivity differences for the core under various operating conditions. These were obtained from one-dimensional calculations with the radial buckling based on the leakage determined from the R-Z PDQ calculations. The results were corrected for the bias inferred from the R-Z PDQ calculations. Table II compares the calculated eigenvalue differences from Table I with, first, the predicted ΔK values for the SL-1 quoted in the "Hazards Summary Report on the ALPR," ANL 5744, and, second, reactivity values inferred from the change in rod bank position for the various conditions. An incremental rod worth of 0.55% $\frac{\Delta K}{K}$ per inch of motion was used to obtain the comparison. This was determined for hot operating conditions in the lifetime calculation described below and is being used somewhat arbitarily to provide this comparison.

TABLE I

PREDICTED EIGENVALUES AND REACTIVITIES FOR SL-1

Condition	K	ΔK	$\frac{\Delta K}{K_1 K_2}$
<u>Beginning of Life:</u>			
Cold (83°F) Rods Out	1.078	.027	2.4%
Hot, Zero Power, Rods Out	1.051	.012	1.1%
Operating, 2.56 MW, No Xenon	1.039	.013	1.2%
Operating, 2.56 MW, Equilibrium Xenon	1.026		
Cold (83°F) Rods In	.957	.121*	11.8%
<u>932 MWD</u>			
2.56 MW, Equilibrium Xenon	1.036	.010**	.94%

* ΔK = K (rods out) - K (rods in)

** ΔK = K (931 MWD) - K (0 MWD)

TABLE II

COMPARISON OF PREDICTIONS AND MEASUREMENTS

Condition	Results of SL-1 Evaluation at 1.56 MW(17) Δ K	Δ K/K	ANL Predictions at 3 MW(23) Δ K	Δ K/K Measured (inferred from rod positions
Temperature Defect 83 - 420°F	.027	.024	.015 - .020	.027
Vapor Defect	.012	.011	.013 - .020	.013
Equilibrium Xenon	.013	.012	.030*	.008 - .01
Maximum Xenon	.002	.002	.01 - .015	.002
Rod Bank Worth Cold	.121	.12	.15	.114**

* Includes samarium.

** See description of rod calibrations (section c)

With this analytical background information in mind, the reactivity history of the SL-1 reactor will now be presented. In particular, the shutdown margins will be discussed and also the inference of a mechanical loss of boron based on the difference between predicted and observed rod bank positions.

b. Lifetime Calculations

This section presents an evaluation of the methods used in the SL-1 lifetime calculations,[17] and an estimate, where possible, of the effects of various uncertainties on the calculated rod bank positions and reactivities. This is of special significance since the estimates to date of the mechanical loss of boron are based on the difference between the observed and predicted rod bank positions.

1) Methods of Analysis

The lifetime behavior - excess reactivity, rod bank position, fuel and poison depletion, power distribution, etc. - was calculated by two methods: (1) one-dimensional (axial) "window-shade" technique, and (2) a simplified three-dimensional (cylindrical) synthesis.

Window-Shade Calculations - In the "window-shade" method the CANDLE depletion code (in the slab form) was used, which automatically moves the boundary between the rodded and unrodded region until criticality is achieved at each time step. The code also recalculates the core composition at each time interval. The initial homogenized core composition for all materials but control rods was obtained from the beginning of life R, Z calculation. The control rods were represented by a uniform poison above the bank position and a constant radial leakage was used throughout the core. The control rod poison and radial leakage were selected to give criticality at the observed beginning of life bank position and the same axial split in power between the rodded and unrodded regions of the core as in the beginning of life R, Z calculations. From these calculations, it is estimated that at 932 MWD (core life at the time of the incident), the average U-235 depletion was 8.3% and that the average B^{10} burnup was 36.7%.

Synthesis Depletion Calculations - To take into account the non-uniform radial depletion, a first order synthesis of radial and axial calculations was performed for various times of core depletion. The rod bank positions from the synthesis calculations are in good agreement with the "window-shade" positions up to about 900 Mw days. Thereafter, as expected, the rod bank was predicted to come out faster.

2) Uncertainties in Self-Shielding Factors for Boron

There are, of course, the usual uncertainties in reactor physics calculations which are common to most water-moderated reactors. These include the uncertainties in cross section, in the general approximation by a few neutron energy groups, and in the three-dimensional analysis techniques. The cross sections and analysis methods which were used for the SL-1 evaluation study are in fairly general use and their validity has been demonstrated from time to time by comparison with many critical experiments.

There are, however, uncertainties in the analysis which are peculiar to reactors with localized self-shielded burnable poisons such as are present in the SL-1. These uncertainties increase with the complexities introduced by the two-dimensional arrangement of aluminum-boron strips, strips of different lengths, and the variety of environments for the aluminum-boron strips (adjacent to control rods, to water channels, or to control rod followers).

One-Dimensional Approximation - For the SL-1 evaluation the calculation of self-shielding factors was made using one-dimensional slab geometry to make the calculation relatively simple and straightforward. The self-shielding factor for any material was there defined for calculation convenience as the ratio of the flux in the material to the average flux in the fuel cell. Since reactivity is more sensitive to the ratio of neutron absorption in boron than to that in fuel, we will here concern ourselves with a definition of self-shielding factor, which is the ratio of the flux in the boron to that in the fuel. Essentially, a very detailed picture of the core was constructed, and flux distributions were found. In the case of black burnable poisons, and for very small regions, diffusion theory is inadequate, and transport theory, or higher order approximations than diffusion theory calculations were used. Even with simple geometry, if there are more than two or three regions associated with the boron, moderator, fuel and structure, hand calculations become too involved, and digitial computer codes were used.

Two-Dimensional Effects - The actual geometrical configuration of the aluminum-boron strips in the SL-1 is more complex than the above dimensional model. These strips are perpendicular to some fuel plates, and parallel to others (see Fig. 21). At the time of the SL-1 evaluation report, sufficient time and methods to perform two-dimensional calculations were lacking; therefore, reasonably precise one-dimensional, self-shielding factors were calculated using double P_1 transport theory. A rough check was made of the validity of this one-dimensional representation by comparing a two-dimensional with a one-dimensional diffusion theory calculation with homogenized fuel and water and explicit aluminum-boron strips. Although this check showed only one half percent difference in reactivity, a further check is now planned using a two-dimensional P_3 transport calculation in completely explicit geometry.

Combinations of Different Self-Shielding Factors - Aluminum-boron strips are placed throughout the core in varying amounts. Some are next to control rods, rod channels, or rod followers, while others are more or less surrounded by fuel. There are more strips of boron in the lower central region of the core than elsewhere. In addition, during operation, the hydrogen density is spatially dependent. There are, therefore, many different representative regions of the core, and many different self-shielding factors. Having obtained the appropriate self-shielding factor for each strip of aluminum-boron in the core, one is faced with the problem of combining these numbers into a smaller number, or group, which could be used in lifetime calculations.

For the lifetime calculations in the SL-1 evaluation study, the boron was homogenized over two characteristic regions, each with an appropriate quantity of boron and self-shielding factor. These were the beginning of life rodded and unrodded sections of the core.

The problem of coalescing different self-shielding factors quickly becomes very complicated. Consider a simple example of two pieces of boron which are identical except for self-shielding factor. Suppose for simplicity one assumes, as was done in the SL-1 evaluation, that the self-shielding factors remain constant through life and he then uses the average value for the two strips. It is easy to show that, even with the assumption that the individual factors remain constant, the average value through life would not be constant but should approach that of the strip with the lower self-shielding factor because it burns up more slowly.

Time-Dependent Self-Shielding Factors - It is, of course, also true that, as the fuel and boron are depleted in the core, the neutron flux distribution changes and, with it, the disadvantage factors for the aluminum-boron plates. In the Evaluation Study, the factors were taken to be constant through life, for simplicity; however, the effect of this simplification was then checked by recalculating the multiplication factor at end of life with appropriate disadvantage factors and the reactivity gain was only one half percent $\Delta K/K$. There would probably be an additional one half percent gain if the boron had been allowed to burn out faster during life with increasingly higher disadvantage factors. It is not possible to predict without considerable additional detailed analysis what overall effect on reactivity in the SL-1 arises from the treatment of boron by coalescing the time independent disadvantage factors.

3) Effect of Changing the Self-Shielding Factor

In view of the above uncertainties in the boron disadvantage factor, it is interesting to examine the sensitivity of the reactivity-lifetime relationship to small changes in disadvantage factor.

Hand Calculations - Two sets of hand calculations were carried out on the basis of a uniform core with the same average composition as the bottom half of the SL-1 core. Since xenon builds in very rapidly, the beginning of life xenon number density was assumed to be equal to the equilibrium xenon number density. For simplicity, percentage changes in thermal utilization rather than reactivity changes were calculated, since to a very good approximation the two are proportional. The effects of samarium were not included.

In one set of calculations, the disadvantage factor was assumed to be time independent; in the second, the factor was assumed to be time dependent, such that its initial value and shape are determined by the value at beginning of life and the asymptotic value is unity.

The results of the hand calculations are presented in Figures 22 and 23. For comparison purposes, changes in the unrodded effective multiplication factor obtained from the lifetime no-control eigenvalue curve given in the SL-1 evaluation report are included. It is apparent that relatively small changes in disadvantage factor can produce significant changes in the magnitude and shape of the reactivity curve.

Window Shade Calculations - The lifetime window shade calculations were rerun using smaller time steps with the original boron disadvantage factors and also values 10% larger and 10% smaller. For further comparison, a calculation was also made with a disadvantage factor of unity corresponding to a homogeneous distribution of boron in fuel. The use of shorter time steps, as discussed later, allows a more detailed representation of the samarium buildup. The critical rod bank positions during life are shown in Figure 24. Again, one may see that a 10% change in boron self-shielding factor (which, considering the complexities involved, might not be too large a change to expect) produces significant changes in the shape of the reactivity curve.

It is interesting to speculate that if the boron self-shielding factor in the Evaluation Study had been 10% higher - and at this time there is no basis for assuming this - the rod bank would have been predicted to go in about $1\frac{1}{2}$ inches further by 700 MWD, which is the approximate time when the loss of aluminum-boron strips was discovered. On this speculation, the difference between the observed and predicted rod bank at that time would have been only one inch instead of $2\frac{1}{2}$ inches. The main conclusion is that it is not possible to reliably estimate from the difference between the observed and calculated rod bank positions the amount of boron that may have been lost from the core by corrosion, or by mechanical means. Further study is needed to tie the matter down more quantitatively.

c. Reactivity Worth of Control Rods

1) ANL Rod Calibrations

Chronologically, there were three periods during which rod calibrations were conducted. First there were calibrations made by Argonne National Laboratory personnel early in core life. The second and third were conducted by CE personnel, one shortly after CE began operation of SL-1 and one following the discovery of the mechanical loss of aluminum-boron plates. The ANL rod calibrations shown in Figure 25 were performed in the cold reactor prior to any power generation. The reactor was brought to critical at various rod bank positions by dissolving suitable amounts of boric acid in the water. Rod No. 9 was withdrawn a small amount, period measurements made and an incremental rod worth inferred. Using a similar procedure the incremental worth of the four remaining rods in a bank was obtained. The curves shown in Figure 25 represent the integrated rod worth as a function of position. The curve for the five rod bank was obtained by addition of the rod No. 9 and the four rod bank data. This curve is the bank calibration curve which is included in the SL-1 Operations Manual. An effective delayed neutron fraction of .007 was used in converting reactivity in dollars to Δ K/K units.

2) Early CE Rod Calibrations

A comparison of the relative worths of the otherwise identical four side rods was made at 1.4 Mw(th), on July 1, 1959, following 160 MWD of core operation. In carrying out the measurements, the bank was maintained at a constant withdrawal of 21 inches and the movement of each rod was balanced by movement of rod No. 9. The results shown in Figure 26 indicate that rod No. 5, nearest the antimony-beryllium source, is worth 0.2 to 0.3% more in reactivity than rod No. 1, which is farthest from the source. Rods Nos. 3 and 7, which are equidistant from the source, appear to be worth 0.1 - 0.2% less than rod No. 5. The difference in rod worths was attributed to the relative proximity to the beryllium in the source. Reactor water temperature was measured by means of thermocouples located at the inlet to the purification system, as was normally done when instrumented fuel assemblies were not in the core.

A calibration of rod No. 7 at $120^{\circ}F$ was performed on August 31, 1959, after 200 MWD of operation. Period measurements were taken for incremental motion of rod No. 7. Rod No. 3 was moved to maintain the reactor critical for various positions of No. 7 while the remaining rods were maintained 11.6 inches withdrawn. The integrated worth of rod 7 from these measurements was \$2.60 or 1.8% in reactivity (using a β eff of .007).

3) Latest CE rod Calibrations

The most recent set of rod calibrations is reported in CEND 1005 (Evaluation of the Loss of Boron in the SL-1 Core 1), The calibration of rod No. 5 was conducted on September 13, 1960, after about 715 MWD operation. The measurements of differential worth were hampered by the high source power level (0.1 to 1 watts) and difficulty in controlling the temperature of the water which was being heated by decay heat. Thus, period measurements were made only in the 10 to 100 watt range. The cooling system was turned on and off occasionally to keep the water temperature in the range of 99° - $114^{\circ}F$. Water temperatures were measured by thermocouples in an instrumented fuel assembly. As the rod No. 5 was withdrawn from fully-in to 24 inches out, the other rods were inserted as a bank from 11.25 inches. The differential worth values exhibited considerable scatter; however, when smoothed and integrated a worth of 1.5% is obtained for the No. 5 rod. The integrated worth curve is shown in Figure 27.

The other calibration of No. 5 rod quoted in CEND 1005 was carried out on August 25 with control rod No. 7 fully inserted. The water temperature of $155^{\circ}F$ was read at the inlet to the purification system. The differential worth curve for No. 5 obtained in this calibration showed an unusually high peak and a consequently high worth of $\sim$ 2.5%. The high worth obtained here for rod No. 5 is probably due primarily to the complete insertion of No. 7. This calibration was repeated on September 13, 1960, as described above, with the remaining control rods moving as a bank.

The calibration of rod No. 9 (August 25, 1960) reported in CEND 1005 was carried out only for the withdrawal range of 0 to 12 inches of No. 9 with the other rods in the following positions: No. 1 at 9.2 inches, No. 3 at 16.8 inches, No. 7 at 9.2 inches, and No. 5 rod inserted from 22 inches to 3.2 inches to compensate for the withdrawal of No. 9. In this range, the differential worth of No. 9 was obtained from the worth of No. 5 obtained on September 13, 1960. Beyond 11 inches of No. 9 withdrawal, the shape of the No. 5 differential worth curve was used. Integration of this fabricated curve gave a total worth of about 5.3% for No. 9. The extreme uncertainty of this extrapolation procedure for the total worth should be recognized.

For the determination of shutdown and worth of the cadmium Tee rods, a worth curve for the entire bank was needed. This curve was synthesized by adding four times the worth of No. 5 rod from the September 13, 1960 calibration to the worth of No. 9 rod. There are many configurations of the rods in which this can be checked, several of these (see Figure 34) are observed to give good agreement between the shutdown, as measured by No. 9 withdrawal, and the shutdown as indicated by the bank position. General agreement has not been established nor is it clear that this method of inferring the rod bank worth is valid. The inferred worth for the entire bank is shown in Figure 17.

The calculation of the rod bank worth in the cold SL-1 reactor at beginning of life was carried out by means of two one-dimensional axial criticality calculations with rods fully in and fully withdrawn, respectively. From calculations in the transverse direction in which the control rods were represented explicitly, the rods were homogenized and treated as an effective homogeneous poison uniformly distributed in the core. The resulting rod bank worth was 11.8% in reactivity, at the beginning of life, which may be compared with the Argonne measured value of 14.5% and the value of 11.4% inferred by CE from limited rod calibration data. In addition to the cold rod worth, an incremental rod worth for the hot operating reactor was calculated with the rods located at the critical position. This was obtained in the course of the core depletion calculation described above. The calculation indicated that the rod bank was worth about 0.55% $\Delta K/K$ per inch of motion near the critical position at the beginning of life. The rod bank in this case was withdrawn about 20 to 22 inches. Although this value for the incremental bank worth is applicable only to the conditions for which it was computed, it has been used as an approximate indication of the rod bank worth in the hot core in those cases where the rod bank is not too far from the critical positions for which the value was computed.

d. Core Fuel Loading History

All but three of the fuel elements which were in the reactor at the start of CE operation (see Fig. 21) have been in the positions shown for the entire 932 MWD. Figure 21 shows the arrangement of fuel assemblies as of February 5, 1959, but does not show the position indices (which are necessary to describe the changes made in the arrangement). These indices are two-digit numbers, the first digit giving the row, the second the column of the position in the 8 x 8 array, starting at the upper left hand corner of the drawing. The four corner positions 11, 81, 18, and 88 are counted in this numbering system, but contain no fuel or dummy elements.

Subsequent to CE taking over the operation of the reactor, some rearrangements of the fuel were made for inspection and for the installation of instrumented fuel elements. During one of the periods, the reactor was operated briefly with an extra fuel element in place, making a 41-element core assembly for a period of 14 days (16 MWD).

The following information from the fuel log was obtained to provide a record of the fuel element changes made up to the time of the incident:

September 23, 1959 (213 MWD)

Fuel element No. 6 was moved from position 45 to position 87, which was previously unoccupied.

Instrumented fuel element No. 63 was placed in position 45.

October 7, 1959(229 MWD)

Instrumented fuel element No. 63 was removed from position 45 and placed in a fuel storage well.

Fuel element No. 6 was moved from position 87 back to position 45.

Fuel element No. 38 was moved from position 55 to the storage well for later inspection.

Fuel element No. 42 was moved from position 66 to position 55.

Fuel element No. 62 was placed in position 66. This was a new element with full boron strip No. 73 containing 0.41 gm B-10 and a half boron strip No. 75A containing 0.19 gm B-10

August 21, 1960 (680 MWD)

Fuel element No. 42 was removed from position 55 and placed in the fuel storage well.

Instrumented fuel element No. 1 was placed in position 55. This element had only a half strip of boron.

Fuel elements Nos. 19 and 59 were removed from position 85 and 75 respectively, examined, and returned to their positions.

Thus, in Figure 21, elements 38 and 42 have been replaced by elements 1 and 62 respectively. These latter two elements have been in the reactor for 253 and 703 MWD of operation respectively.

e. Rod Bank Position Measurements Throughout Core Life

The reactivity history of the reactor can be inferred from the control rod positions measured under various conditions during life. These data fall into two general categories. The first category is made up of physics test data which include those measurements made under carefully controlled conditions. For these measurements, care was taken to insure criticality, rather than some long period, to insure that the rods are banked and to insure that the reactor is at the desired power level and in the desired xenon condition. These measurements were made periodically during the SL-1 operation. The second category includes those data taken on a routine basis (once each shift) by the operating crew, and recorded in the operations log. In this case the rods were often not in a bank, the xenon history was either very complex or not known, and plant conditions were often changing.

1) Physics Test Date

The major results of the physics tests are shown in Figures 28, 29 and 30. Figure 28 shows the variation in rod bank position with temperature, taken after 200 MWD of operation during a reactor cooldown from operating temperature. For temperatures above 200°F, temperature values were inferred from the pressure indication on the reactor. At that point during cooldown at which atmospheric pressure was reached, a port in the vessel head was opened and a thermocouple inserted which provided further temperature indication. Reference to Figure 28 shows that the rod bank position appears to be linear with temperature; however, a small discontinuity appears at the point of change between the two methods of temperature measurement. Using the .55%/inch reactivity worth described above for the control rods the temperature coefficients shown in the figures can be inferred.

Figure 29 shows the result of several measurements early in core life of the variation in rod bank position with steam flow rate. A steam flow of 8000 lbs/hr is equivalent to a 2.56 Mw power level. The vapor defect inferred from the rod bank motion is 1.3% in reactivity which with the calculated 7.1% vapor fraction at 2.56 Mw yields an average vapor coefficient of .18% Δ K/K per percent vapor. A local vapor coefficient which seems to apply in the 2000 to 8000 lb/hr range is .22% Δ K/K per per cent vapor.

The physics test points taken periodically during core life are shown for various operating conditions in Figure 30. In some cases, especially the cold and hot zero power cases, it was necessary to correct the rod bank position so that the plotted data corresponded to the same temperature. This was done using Figure 28. Corrections for power level were also made, using Figure 29. The uncorrected data from which most of the points plotted were taken are given in Tables III and IV together with the sources of the information. The points were connected by means of straight lines merely for ease in reading. No trend between observed points is implied. The jump at 853 MWD corresponds to the insertion of the cadmium strips in the two Tee slots and the rod bank measurement at 180°F shortly thereafter. A horizontal line was drawn from the last data point to the jump, and the magnitude of the jump was based on the first subsequent data point.

2) Operations Log Rod Bank Data

As stated above there exist, in addition to the physics test points, a large number of control rod positions, recorded in the operations log on a routine basis. Starting on September 9, 1959, after 206 MWD operation, the indicated position of each rod was recorded at the beginning of each shift, along with the main steam flow, reactor water temperature and pressure and other pertinent variables. Prior to that date, these data were recorded several times daily but not on

TABLE III

SL-1 CRITICAL ROD BANK POSITIONS TAKEN DURING PHYSICS TESTS
INCHES WITHDRAWN

MWD Operation	0 (24)	68 (17)	200 (19)	320 (25)	711 (20)
Cold Critical					
5 Rod Bank	12.3/12.8 (94°F/70°F)	12.8 (83°F)	13.7 (120°F)	14.9 (188°F)	10.5 (95°F)
4 side rods**		16.8 (83°F)	18.2 (120°F)		
Center Rod***	19.1 (94°F)	19.2 (83°F)	20.9 (120°F)		14.3 (95°F)
Hot, Low Power	17.3	17.4	18.0	18.4	14.2
Hot, High Power, No Xe	20.2 (2.56 Mw)	19.8 (2.56 Mw)	19.9 (2.2 Mw)	20.7 (2.7 Mw)	16.6 (2.4 Mw)
Equilibrium Xe	21.1 (2.56 Mw)	21.2 (2.56 Mw)	21.7 (2.2 Mw)	22.8/23.2 (2.7/3.0 Mw)	17.8* (2.5 Mw)
Maximum Xe				23.3 (3 Mw)	
Low Power, Maximum Xe				20.2	

* For 735 MWD.

** Center rod completely inserted.

*** Four side rods completely inserted.

TABLE IV

SL-1 CRITICAL ROD BANK POSITIONS
PHYSICS TEST DATA

Date	MWD	Conditions	Indicated Rod Bank Position
Sept. 16	711	Hot 407°F, zero power	14.3*
Sept. 16	711	2.5 Mw, no xenon	16.6
Sept. 25	736	2.5 Mw, equilibrium xenon	17.8
Nov. 6	848	2.56 Mw, equilibrium xenon	17.6
Nov. 15		CADMIUM STRIPS INSERTED	
Nov. 16	853	180°F, zero power, no xenon	13.2
Dev. 5	888	2.56 Mw, equilibrium xenon	19.3**
Dec. 23	932	2.56 Mw, equilibrium xenon	19.4

* Rod #9 was at 14.4"

** Rod #9 was at 19.2"

as regular a basis.* In collecting these data from the logs it was extremely difficult to ascertain the reactor power history which accompanied each point in order to estimate the xenon condition of the reactor. This was due to frequent changes in power level, and shutdowns and startups between data points, as required by the training and test program. In addition, the reactor power level associated with each point is a function of reactor pressure, feedwater temperature and main steam flow, the last of these being the most significant. Since for comparison these data points must be put on the same basis a correction for power level and xenon history is necessary. It was felt that the labor involved in an exact correction of each or even some of the points for power and xenon would be prohibitive. Therefore, the following procedure was established for measurements during power operation of the reactor:

(1) All the data taken over the period from February 5, 1959 to December 23, 1960 were tabulated.

(2) All points corresponding to main steam flow rates less than 4000 lbs/hr ($\sim$1.3 Mw) were discarded.

(3) Those remaining points for which two full days (48 hours) prior operation between 4000 and 8000 lb/hr is recorded were retained. All others were discarded.

(4) Only those points were retained for which the rods in the bank were within three inches of each other.
Using the calibration curves in Figure 27 and assuming the calibration curves apply to each rod independently over the small correction range, the individual rod positions were so corrected as to give a common bank position.

(5) The resulting rod bank positions were corrected for vapor fraction to a common steam flow of 8000 lbs/hr by means of Figure 29.

With this procedure there is reasonable assurance that the resulting data with some small (compared to the original data) margin of uncertainty can be considered the critical rod bank positions corresponding to 8000 lbs/hr steam flow and equilibrium xenon. The major shortcoming of this procedure is the fact that the correction to 8000 lbs/hr steam flow was based on

---

* More detailed and complete data also exist in the hourly log sheets, however, these would require an extensive amount of time to analyze.

measurements made with no xenon, and that the two days prior operation was not necessarily operation at the same power level. Considering that the rod bank motion during buildup of equilibrium xenon at 8000 lbs/hr is under two inches, and considering that the large majority of the points were above 6000 lbs/hr, this procedure could result in a scatter of one half to oneinch in the bank positions with most points within one half inch of the general trend. Other possible sources of scatter in the points are the effect of deviations in reactor pressure and feedwater temperature on the power level (which was inferred from the steam flow), the uncertainty in reading of the instruments and possible changes in reactor conditions prior to taking the data.

The results of this collection of rod bank positions are shown in Figures 31 and 32 as a function of calendar time, and in Figure 33 as a function of megawatt days of operation. Gaps in the data are due to shutdown or of less than two days continuous operation at power. The data as a function of megawatt days operation is more useful for careful examination and is, therefore, plotted in Figure 33 on a larger scale. Figure 33 also shows the physics test data at equilibrium xenon for 2.56 Mw. The first conclusion to be drawn from Figure 33 is that the trend implied by the few physics test points is fairly well borne out by the larger accumulation of data. The steady inward motion of the rod bank from about 300 to 700 MWD and the apparent leveling off near 700 MWD is indicated by both sets of measurements. Even the slight outward motion of the rods near beginning of life appears in both sets.

Although the differences are small, the observed variation of rod bank position in time differs in two ways from the prediction. First, during the period prior to about 300 MWD the rod bank appears to be coming out. The second noticeable difference is that, subsequent to 300 MWD, the rod bank is observed to go into the core considerably faster than predicted in a rather systematic fashion up to about 700 MWD.

There are two possible reasons for the rod bank coming out early in life. First, the buildup of equilibrium samarium would result in a maximum rise in rod bank of 0.7 inch over the original window shade prediction with a peak in the neighborhood of 100 MWD, as shown in Figure 30. The buildup of equilibrium samarium was recently calculated using small

($\sim$200 hr) time steps in the original window shade calculation. The result is shown along with the original window shade calculation in Figure 30. This accounts for a rise in rod bank, but not over a 300 MWD period. The picture might be modified some if the change in self-shielding factors of boron with depletion is accounted for. This can only be determined by more detailed calculations for the core depletion.

A second reason arises from the method of zeroing the rods; i.e., the method of positioning the rod relative to the core during mechanism assembly at the proper point while the indicator is at zero. Prior to May 14, 1959 (88 MWD) there was no well defined method for measurement of the control rod position at indicated zero. On that date measurements on a disassembled mechanism with reference to construction drawings revealed that at indicated zero the bottom of the cadmium should be 3-1/8 inches below the bottom of the core. Following this measurement a tool was fabricated consisting of a pipe with a gage mark which enables pre-setting the rod position while the mechanism is connected to the indicator. It is, therefore, reasonable to expect considerably more uncertainty in the measurements prior to 100 MWD (the first zeroing following the position measurement). Reference to Figure 33 shows that a small shift in rod bank position could have occurred, but this is not clear in view of the scatter of the points. The details of the rod zeroing procedure, and an estimate of the uncertainty involved are discussed in Section II C3.

The inward motion of the rods subsequent to 300 MWD burnup was at first not considered surprising in view of the simplicity of the lifetime calculation. By the time core burnup reached 700 MWD, and the disparity was close to three inches, some concern was felt about the growing discrepancy. When the mechanical loss of aluminum-boron plates was observed during inspection of the fuel elements, it was at least consistent with the unexplained gain in reactivity of the core.

An investigation was made for evidence of any sudden increases in reactivity which might be indicative of the sudden mechanical loss of boron poison. It is difficult to identify in Figure 33 any clearcut evidence of sudden inward jumps in the rod bank. What may appear as jumps at 344, 470, and 590 MWD, for example, may be no more than scatter of the data. Since it is clear that the rod bank is steadily coming in, in the 300 to 700 MWD interval, it is difficult to distinguish between the steady inward trend and a so-called jump. The notation of the times of rod assembly and zeroing and the times of fuel assembly motion show that no correlation between any apparent discontinuities and fuel assembly disturbance can be made. However, some of the possible jumps can be correlated with the rod zeroing time, for example, at 470 and 590 MWD. In the light of this discussion it is concluded that there are no clear-cut indications of a sudden increase in core reactivity.

As has been mentioned earlier, it is not justifiable to attribute the entire deviation of the observed rod bank position, from that predicted, to mechanical loss of boron for two reasons. First, there is no indication of just how much boron has physically been lost from the core, or of the spatial distribution of the loss. Second, there remains the uncertainty in the predicted lifetime rod bank position curve which arises mainly from the treatment of the self-shielding of the boron strips in the complex SL-1 configuration, as discussed earlier.

f. Indicated Shutdown During Life

As a result of the discovery of the loss of aluminum-boron strips, immediate concern was felt for the shutdown margin of the reactor. Calculations indicated that if all the boron were lost at 700 MWD burnup the cold reactor would be supercritical by 3.2% Δ K/K with all five control rods in. It was decided to insert cadmium strips in the two unused Tee slots in the core to provide additional reactivity shutdown. This was done on November 15, 1960, and resulted in the control rod bank moving out as observed in Figure 30 at 850 MWD. The worth of the cadmium is estimated to be 0.8% Δ K/K based on the cold rod bank motion observed. The cadmium inserted in the reactor comprised six full length strips, each 4-13/16" wide and placed in two Tee slots. Calculations made for the insertions of four full Tee control rods (14" in the full span and 7" in the single arm) as fabricated for SL-1 indicated a worth of 3.9% Δ k or 3.3% in reactivity. Adjusting this calculation for the narrower cadmium strips and for the fact that only two out of four Tee slots were used, the best analytical estimate that can be given, without recalculation for the cadmium actually placed in the reactor, is 1.1% in reactivity.

The best indication of the shutdown margin at the time of the incident that can be obtained at the present time comes from the observed rod bank positions shown in Figure 30, and the rod calibration curves in Figure 27. As described above, there is some question concerning the applicability of these calibration curves to situations far different from those for which the measurements were taken and further analysis would appear worthwhile to assess this. These curves constitute the only measured rod worth data close to the time of the incident. From the observed rod bank position and from the calibrations curves, the amount of reactivity held down by the rod bank can be determined. From this, the shutdown margin is inferred. This was done for the cold rod bank positions shown in Figure 30 and for the rod No. 9 critical positions. The results are shown in Figure 34. It will be noted that at those times where data are available both for the bank and for rod No. 9, the two imply almost the same shutdown margin, thus increasing the confidence in the use of these calibrations.

Figure 35 shows estimates of shutdown margin based on the Argonne rod calibrations described above (Fig. 25). These calibration curves imply still more shutdown than the ones taken recently by CE. Also, with these calibrations a worth of 1.1% $\Delta K/K$ for the cadmium strips is inferred, as compared to the 0.8% implied by the CE calibration.

On the basis of these estimates of the shutdown margin at the time of the incident, and the calibration curves for rod No. 9 given in Figures 25 and 27, the position of rod No. 9 required for any given core reactivity can be estimated. At $83^{\circ}F$ the indicated rod 9 positions with all other rods inserted are 17.3, 19.5 and 24.3 inches for critical, prompt critical, and 1.8% $\Delta K/K$ supercritical respectively. These values are based on the CE calibrations (Fig. 27). Values of 18.1, 19.6 and 22.8 inches for critical, prompt critical and 1.8% $\Delta K/K$ supercritical are obtained by use of the ANL rod calibrations (Fig. 25).

3. Operational History of Control Rod Drive Mechanisms

a. Design Description

Vertical linear motion is imparted to the SL-1 control rods by a rack and pinion drive mechanism. The rack and pinion gears, the pinion support bearings and the back-up roller operate in saturated steam and water in a housing mounted above the reactor vessel (Figs. 36 and 37). A set of concentric springs located in the upper portion of the mechanism housing aids in absorbing the shock imposed upon the mechanism components during scram.

A rotary shaft pressure seal is used where the pinion drive shaft penetrates the mechanism housing. The pressure seal is of the positive clearance, break-down type, which has controlled leakage (Figs. 36 and 37). Water provides cooling for the seals, prevents outward steam leakage and provides a flow of water into the mechanism which bleeds down into the reactor vessel. Leakage from the seal is collected by a lantern ring and returned to the condensate tank. The seals each require approximately .01 gpm of water bled continuously from each control rod drive housing.

The control rod drive motor and position indicator assembly (Figs. 36, 37, and 38) are located outside the concrete biological shield above the reactor vessel. A universal coupling and extension shaft connect this assembly with the pinion drive shaft. The electric drive motor is engaged with the pinion shaft by means of a magnetic clutch (Fig. 37). Failure of the clutch current automatically results in rapid insertion of the rods into the core by the force of gravity. The mechanism is so designed that a scram signal will not only release the magnetic clutch, but also provides a back-up by energizing the drive motor to give a downward drive to the control rod. This is by positive action through a mechanical overriding clutch which free-wheels on a rod withdrawal but engages when the rod is driven in. In the event of power failure, the control rod motor current is supplied by an emergency power system.

Since the internal spring is unable to absorb all of the control rod free fall energy, two negator springs were attached to each pinion shaft. A gear on the negator spring drum drives the gear train that is coupled directly to the position indicator synchro-transmitter and micro-switches. This synchro arrangement assures the operator of positive rod position indication at all times during operation. The micro-switches (Fig. 36) are used to operate the upper and lower limit switches, control panel indicating lights, and electric motor interlocks.

The control rod drive mechanism, and pressure breakdown seals, were designed and developed by ANL and Alco Products, Inc. who also tested a lead mechanism to successfully demonstrate design performance. Over 8000 cycles and 250 scrams were made after which time visual inspection indicated satisfactory performance.

b. Normal Control Rod Positions with the Reactor Shut down

The nominal vertical location of the cadmium absorber section in the control rod blades relative to the nominal location of the fuel for three normal control rod positions with the reactor shut down (illustrated in Fig. 39) are as follows:

(1) When the scram stop washer and nut are removed from the control rod rack, the control rod hub rests on top of the control rod channel shroud. The cadmium section of the control rod extends 6-15/16 inches below and 1-3/16 inches above the fuel region.

(2) It is necessary to raise the rack 5-45/64 inches ($\sim$1 inch for attaching a C-clamp) in order to install the scram stop washer and nut. The cadmium section of the control rod extends 1-15/64 inches below and 6-57/64 inches above the fuel with the rod raised 5-45/64 inches.

(3) When the control rod is in its zero position, the scram stop washer is resting on the spring seat and the springs are deflected 5/8 of an inch due to the weight of the control rod assembly, and the cadmium section of the control rod extends 3-1/4 inches below and 4-7/8 inches above the fuel.

The zero position of the rods is checked when the mechanisms are reassembled. On at least one occasion, it was found that the actual position of the rods was at variance with that shown by the rod position indicators on the control console by as much as 5/8 of an inch. It is possible to have as much as $\pm$ 1/8 of an inch error from a true position in zeroing the rods due to backlash in gears and couplings and an inherent error in the zeroing procedure. (Appendix B) It would also be possible for a zero position to be off an additional 9/32 of an inch as the result of an operator error in locating the top of the rack with the measuring tube.

The rubber coupling which joins the shafts of the selsyn motor and limit switch cams could introduce an appreciable error in rod position during reactor operation. Coupling rotation relative to each shaft is prevented by two (No. 8) cup point Allen set screws bottomed on flats of the shafts. Inspection of two SL-1 selsyn-limit switch units at Windsor shows the cup points of the set screws bearing on the cylindrical surface instead of on the flats. Coupling movement relative to the shaft of .001 inch is equivalent to .053 inch of rod movement. The set screws could move if they were not adequately bottomed on the shafts.

c. Disassembly and Assembly Procedures

The "Nuclear Power Plant Operators Course - Mechanical Specialty Training - Control Rod Drive - SL-1 - Chapter II" issued by the Training Branch - Nuclear Power Field Office, describes the assembly and disassembly of the control rod drive mechanisms and is the document used for the training of the Cadre personnel. The training manual was written in lieu of a manufacturer's manual which was not available. An excerpt is given below with only figure numbers changed to match this report.

"Removal of Control Rod Drive

1. Conditions to be satisfied before the unit can be removed

    a. Reactor scrammed and brought to atmospheric pressure

    b. Reactor water level raised to bottom of plug nozzle in reactor head.

"Removal of Motor and Clutch Assembly (Reference Figure 37)

1. Disconnect electrical connection (#1) to isolate unit electrically.

2. Loosen 2 set screws (#2) and slide coupling off spline.

3. Remove 4 hold down bolts and remove motor and clutch assembly.

4. Manually slide control rod drive shaft from concrete shield block.

NOTE: This procedure is identical for all rods.

"Remove Biological Shieldings

1. Remove top shield plug utilizing a spreader bar and the overhead crane. This plug is constructed of laminated steel and masonite.

2. Remove the four key blocks using the overhead crane

3. Move the five concrete blocks away from the reactor vessel using chain sling and overhead bridge crane.

"Remove Rod Drive Mechanism (Reference Figure 38)

1. Secure feedwater valve to isolate rod drive seals from feedwater pump pressure.

2. Disconnect inlet and outlet lines to rod drive seal assemblies. (#1 and #2) respectively.

3. Remove tie rod studs (#3).

4. Remove seal assembly and place on a clean blotter paper.

5. Remove pinion shaft extension (#4) from thimble (#5). Place on clean blotter paper.

6. Remove socket head nuts (#6) using Allen wrench and soft hammer.

7. Lift off thimble (#5). Caution: this item is very heavy and cumbersome and must be carefully balanced during removal.

8. Remove two retaining rings (#7) and remove pinion and bearings (#8)

9. Secure special tool CRT #1 (Fig. 40) on top of rack (#9) and raise rod not more than 4 inches. Secure "C" clamp to rack at the top of spring housing (#10)

10. Remove special tool CRT #1 from rack and remove slotted nut (#11) and washer (#12)

11. Secure special tool CRT #1 to top of rack and remove "C" clamp, then lower control rod until the gripper knob located at upper end of [control rod] makes contact with the core shroud.

12. Remove 8 socket head cap screws (#13) and lift off buffer spring housing and pinion support assembly (#14) and place on clean blotter paper.

13. Secure two 3/8 inch eye bolts into spring housing [extension tube] (#15). Lift off spring housing and place on clean blotter paper.

14. Place special tool CRT #2 (Figs. 41 & 42) over rack and extension rod (#16) and secure special tool CRT #1 to rack. Connect special tool CRT #2 to hook of overhead crane and take up the weight of rack and extension rod. Rotate special tool in counter-clockwise direction; this action disconnects the split coupling (#17) from the control rod gripper (#18) located at the lower end of the extension rod. The special tools and extension rod are then lifted out by the overhead crane as a single unit.

"<u>Installation of Control Rod Drive</u>

1. Assembly of the rod drive mechanism, replacement of concrete shield blocks and installation of motor and clutch assembly are the reverse of disassembly. Replace all flexitallic gaskets insuring that all mating surfaces are wiped clean with alcohol or other cleaning agent. Particular care should be taken when

securing the rod drive seal cooling lines and fittings. If not properly fitted up considerable leakage will occur and result in a loss of feedwater and pressure.

"Disassembly and Assembly of Components

Seal Disassembly. (Reference Figure 38)

a. Remove snap ring (#19) and coupling (#20). Tape snap ring and key (#21) to coupling to prevent loss of these items.

b. Remove five socket head cap screws (#22) and bearing retainer (#23).

c. Remove bearing locknut (#24) and 5 socket head cap screws (#25) and remove water gland seal (#26).

d. Remove seal shaft (#27).

e. Remove lantern ring (#28).

f. Remove 5 seal diaphragms (#29) and floating ring (#30).

g. Remove retaining ring (#31) and stellite bushing (#32).

NOTE: The seal diaphragms and floating ring must be kept in pairs and in the order of their removal from the seal housing as they must be replaced in their original order. All parts of this assembly will be cleaned using acetone or alcohol and dried with soft lint free material.

NOTE: The assembly of this unit is the reverse of disassembly.

"Spring Housing and Pinions Support Disassembly

1. Remove 4 socket head cap screws (#33) and remove backup roller (#34).

2. Remove 6 socket head cap screws (#35) and remove spring housing (#10).

3. Remove spring seat (#36) and two compression springs (#37) and (#38).

NOTE: Assembly of spring housing and pinions support assembly is the reverse of disassembly.

"Clutch Unit Disassembly (Reference Figure 37)

1. Remove motor from base

2. Disconnect and tag clutch power wires.

3. Remove change gear (#39)

4. Remove instrument pad.

5. Remove 2 socket head cap screws (#40) and bearing cap (#41).

6. Remove spline (#42), bearing (#43), and shaft assembly (#44).

7. Remove 2 set screws (#45) in cam clutch (#46) through hole (#47) in cam clutch cover (#48) and remove drive shaft (#49) and bearing (#50).

8. Remove negator spring drum (#51), cam clutch (#46), and magnetic clutch (#52).

NOTE: Assembly of this unit is the reverse order of disassembly. The refacing of the magnetic clutch is accomplished in the same manner as described in Chapter I, pages 11-13.

"Installation of Negator Spring (Reference Figure 37)

1. Loosen set screw and remove coupling from motor and clutch assembly.

2. Drive rod out until the position indicator in the control room reaches approximately 28 inches.

NOTE: Limit switches must be by-passed.

3. Remove socket head cap screws. (53-54)

4. Install negator spring (55 or 56) in slot on negator spring drum (51) and replace socket head cap screws (53-54).

NOTE: Removal of negator spring is accomplished in the reverse procedure described above."

d. Operating Procedures

A series of three manuals have been written for use in operating the SL-1 facility. The first manual was prepared by ANL and the second and third revisions of the manual were prepared by CEND.

A summary of the ANL manual "Standard Operating Procedures for SL-1 Reactor" included the following control rod operational checks after determing that the nuclear instrumentation was in satisfactory operating condition:

(1) Before start-up the control rods shall be checked for satisfac- operation by raising each rod in turn 10 inches, checking that

the rod will drive in, and then dropping the rod from 10 inches. Drop time should be one second. In this test, each rod is dropped by one of the following scram devices: test push button on channels I and II; the period trip on channel III; the reactor scram buttons; the center control rod scram button. Each of these scram devices is used to test drop one of the five control rods.

(2) When the pressure in the reactor reaches 300 psig, pressurized rod drop tests are performed using the central and selected rod scram buttons. Each rod with the exception of the center control rod (control rod No. 9) is withdrawn 30 inches, dropped and timed. The central control rod is withdrawn 22 inches, dropped and timed. All rods must drop in two seconds or less to continue reactor start-up.

(3) When the reactor is operating during a sustained power run the control rods shall be exercised once each day through a travel of at least one inch up and one inch down. Once each week the rods shall be exercised through the maximum travel possible without reducing power.

The approved CEND operating manual SL-1 (ALPR) Plant Operating Procedure of March 19, 1959 included similar control rod operational checks specified by the ANL procedures listed above with the following exceptions:

(1) When the reactor is at pressure the rods may be dropped from a 30 inch position. If the reactor has been satisfactorily scrammed at pressure within the last 14 days it is not necessary to perform the hot rod drop tests.

(2) Each day at 1000 hours (10 A.M.) the rod bank will be adjusted to maintain the four outside rods within one inch of the center rod.

(3) Once a week each of the outside control rods will be moved through the maximum travel possible maintaining the center rod in automatic demand control.

CEND revised manual "SL-1 Operating Manual, Vol. II - Operating Procedures" dated September 1, 1960 and submitted for comments and approval to the AEC on September 16, 1960, includes similar routine start-up procedures as listed in the ANL manual, with exceptions that pressurized rod drop tests are conducted no more frequently than once per week, and that during a pressurized rod test, No. 9 rod is withdrawn 20 inches for its drop test.

Vol. I, which has not been completed, describes the SL-1 reactor and associated plant equipment. Each system is written in sufficient detail to adequately explain the operation of each system and its relation to the rest of the plant.

e. Rod Sticking Summary

During the operation of the SL-1 reactor there were sporadic instances of slow scram time and control rod sticking which increased in difficulty with time. In order to more clearly understand the nature of these sticking incidents the term "sticking" shall be divided into three types, defined as follows:

Type I - Sticking of a control rod resulting in failure to meet the drop time requirements (one second for 10" drop; two seconds for a 30" drop) and which did not require a power assist from the drive assembly.

Type II - Sticking of a control rod in which the control rod stopped and required a power assist to enable the control rod to reach its zero position (even if it subsequently fell free at a lower level).

Type III - Sticking of a control rod in which it was not possible to drive the control rod in a desired direction, e.g., clutch slippage during a rod withdrawal, or failure of a drive assembly shear key or gears resulting in failure to drive a control rod.

The earliest record of rod sticking incidents is listed in ANL-6084, Initial Testing and Operation of ALPR, dated December 1959 (during ANL operation of the reactor) in which difficulty in meeting the scram time requirements are encountered. Specifically, No. 7 drive mechanism was replaced, and one of the two constant velocity negator springs was removed from each of the five drive assemblies in order to meet the scram time requirements.

A listing of all rod sticking incidents* that took place since CEND has been operating the SL-1 facility is included in Appendix A. A

*Data accumulated for this listing was taken from the Operating and Rod History Logs summary of this listing is shown in Table V and it includes all rod sticking incidents experienced in rod withdrawals, rod drop tests, rod exercising and rod scrams.

Of sixty-three (63) rod sticking incidents during either a drop test (10")or a 5-rod scram, forty (40) of these incidents took place over a twenty-two (22) month period. The percentage of rod sticking incidents considering the number of scrams and the number of rod drop tests** was approximately 2.5% over this same period of the SL-1 operation. In the one month period prior to shutdown (November 18 to December 23, 1960) the incidents of sticking during scrams and rod tests** increased markedly to approximately 13.0%.

TABLE V

CONTROL ROD STICKING SUMMARY

Rod Sticking Incidents - Scram and Drop Tests > 10"
From February, 1959 to November 18, 1960

	Rod Numbers					
Type of Sticking	1	3	5	7	9	Total
I	2	6	3	2	1	14
II	4	3	3	9	4	23
III	-	-	-	2	1	3

Rod Sticking Incidents - Scram and Drop Tests > 10"
From November 18, 1960 to December 23, 1960

	Rod Numbers					
Type of Sticking	1	3	5	7	9	Total
I	1	-	1	2	-	4
II	5	2	1	10	1	19
III	-	-	-	-	-	0

Rod Sticking Incidents Other than During a Scram
or Drop Test > 10"
From February, 1959 to December, 1960

	Type of Sticking			
Rod Operation	I	II	III	Total Sticking Incidents
Rod Withdrawal	1	1	8	10*
Rod Exercise ± 1 inch	-	4	1	5
Rod Exercise - Max. Travel	-	4	-	4
Rod Drop Test - 10"	-	1	1	2

* Four of these incidents occurred during last month of operation.
**Does not include 10" rod drop tests

It can be seen from control rod sticking history that of all the rods, the central rod No. 9, had the best operational performance record even though it was operated more frequently than any of the other rods. It had been successfully scrammed 130 times during the six month period prior to the last shutdown, with only one instance of sticking where it hesitated momentarily at the start of a scram.

In instancesof rod sticking, the Operating or Rod History Logs show that rod drop tests were performed to insure that rods could be operated satisfactorily before continuing further in reactor operation. In those instances where the logs do not list a drop test after a sticking incident, it was an operator error. Although the drop tests were not performed it may be noted that the rods in question did perform satisfactorily in their next operation, e.g., a stuck rod condition during a one inch rod exercise which could later be moved to a desired rod bank position. In instances of a Type III rod sticking condition, the condition was remedied, or the operation of the rod in question limited, as in the case of limiting the withdrawal height of No. 1 rod before continuing further. On September 7, 1960, and again on September 28, 1960, the withdrawal height of rod No. 1 was limited to 20 and 18 inches respectively to avoid rod withdrawal sticking which was known to happen above these elevations. It was also decided to check rod No. 1 at the first convenient plant shutdown period to determine the cause of sticking. On November 9, 1960, a burr was removed from the upper inside edge of the shroud of control rod No. 1 to allow free rod movement above eighteen (18) inches. In the only case of Type III rod sticking where further operation continued, rod No. 7 could not be withdrawn beyond 25.2" for a 30"drop test (June 1, 1959) although it was fully withdrawn in the next test.

In carrying out the assembly operation of raising the control rods for installation of the scram stop washer and nuts, there is no indication either in the logs, or as stated by military and CEND operating personnel, of any rod ever sticking. One of the military crew members stated that he performed this operation approximately 300 times, and another, 250 times with no sticking. Others have performed this operation fewer times but have never felt the rods stick.

f. Component Performance

1) Drive Assembly

The drive assemblies are located outside of the shield blocks on the operating floor. The drive assemblies consist basically of a geared-down electric motor coupled to a shaft containing a pair of clutches. One clutch is a magnetic clutch and the other is an over-riding clutch. The shaft connects to the pinion shaft through a flexible coupling and extension shaft. The shaft further connects to a synchro-position indicator and a series of limit switches through additional gearing.

Tests conducted on various clutches have indicated the following:

(1) A new clutch with properly burnished face is rated at 240 inch-lbs. but tests have indicated it can carry up to 300 inch-lbs.

(2) A clutch that has seen light service (approximately two months of operation) indicated a carrying capacity of only 165 inch-lbs. (69% of rating).

(3) A clutch that has seen medium service indicated that it could carry up to 135 inch-lbs. (56% of rating).

It is believed from the above tests that the torque delivering capacity of the two used cluthces is representative of other SL-1 clutches. The operating logs list four instances in which manual assists were applied to free a sticking rod. In reviewing these cases with the Cadre, they have stated that only one hand was used to apply torque and free the rods. The other hand was used to hold a phone so as to maintain contact with an operator at the nuclear console regarding rod position. Recent tests (March 6, 1961) were conducted to determine the amount of torque that could be applied using a hand assist. The tests were run by three different people and the following results were obtained:

Maximum Torque Using a Single Hand	Maximum Torque Using Two Hands
147.5 inch-lb.	295 inch-lb.
125.5 inch-lb.	266 inch-lb.
140.5 inch-lb.	250 inch-lb.

It can be seen from the above results that the manual assists reported in the Operating Log supplied additional torque to the pinion gear. The amount, however, would either be below the torque value that could be supplied by a new clutch or would not grossly exceed its capacity. Therefore, a hand assist would not do more than apply a torque value that could be delivered by a new clutch.

The Operating Log states that on December 19, 1960, a pipe wrench was used to withdraw control rods 1 and 5 which were sticking on withdrawal to 28 inches. A review with Cadre personnel performing the task indicated that the pipe wrench was attached and under its own weight caused the coupling to be rotated sufficiently to enable the clutch to pick up the load after the sticking spot had been passed. Hand operations prior to and after this operation prove that a hand assist was sufficient to aid the withdrawal.

In summary, the rod sticking phenomenon observed in the withdrawal direction was probably due to the clutch, with its low torque carrying capacity, being unable to overcome the system friction plus the additional forces such as misalignment and corrosion product build-up. It is possible that the center mechanism had fewer problems than any other mechanism simply because of better alignment in its rod mechanism system due to its central location.

The limit switches are mounted on the top of the drive assembly. The limit switches are provided for rod-in and rod-out indication and rod low indication.* They are geared to the shaft which is between the drive motor and the coupling.

In operation it has been found that the limit switch assembly is quite flexible. Consequently, at an occasional scram, the rod-in limit switch would by-pass its end point and reactuate the drive motor and

*Indicates rods below 3 inches. Following a reactor shut-down or scram all control rods must be below 3 inches before any control rod can be raised.

attempt to drive the rod further against its bottom hard stop. When this occurred, there would be a failure of either the shear key or the gear teeth at the drive motor. Also the cams do not always maintain their adjustment during operation.

The over-running clutch is a unidirectional "Sprague" clutch which is disengaged in the withdrawal direction and free wheels on scram but engages on drive in. In operation this clutch has worked very satisfactorily and tests performed show no problems with this item.

The Selsyn generator is mounted on top of the drive assembly and is connected to the gearing for the limit switches. The Selsyn provides a signal for rod position indication. In operation these items have worked satisfactorily.

2) Seals

The control rod drive mechanism seal is a five-element, controlled leakage, labyrinth pressure, break-down seal. It consists of five floating rings and five stationary rings all made of stellite. The faces of the seal rings are lapped to a very fine finish and to a very flat surface.

The bore of the seal is accurately controlled to keep leakage at a minimum. Cooling water is fed in between the pinion shaft bushing and the seal. Leakage through the seal is accumulated in a lantern ring and drained to the hot-well. When the system is shut down the seal rings rest on the pinion shaft. When the system is pressurized the pressure difference across each ring seats each floating ring against its mating stationary ring. As the pinion shaft rotates it attempts to center the floating ring and must overcome the frictional force between the stationary and floating rings. Water is allowed to leak between the shaft and floating rings and is drained from the seal assembly through the lantern ring. This water contains corrosion products (crud) from the primary loop, some of which then deposit out in the seal assembly. As the shaft rotates the abrasive action of the crud and the frictional resistance of the floating ring cause wear to occur on the pinion shaft. Since wear is associated with friction this means that there are retarding forces which hinder complete freedom of operation of the pinion shaft.

It has been shown by tests that when the seal water flow is increased above about 120 GPH (for 5 mechanisms) a sufficiently high frictional force could be obtained to prevent scramming of the rods from occurring. The normal flow rate of seal cooling water is approximately 50 GPH. Whether this phenomenon is due completely to the seal or to the manner in which the seal water enters the seal housing and impinges upon the pinion shaft is unknown at this time, however, initial tests on a seal having a greater clearance indicates that it might be a seal water entry problem.

Tests have been conducted on a new carbon face seal. The preliminary data obtained indicate that this seal shows a great deal of promise both in low leakage rate and in low resistance to scramming.

In summary, it is believed that the seals caused additional frictional retarding forces to be supplied to the control rod system. It is not believed that this could cause sticking at a finite location if the system had been moving just prior to the sticking.

3) Bearings

The SL-1 control rod drive mechanism has three different types of bearings. These are ball bearings, sleeve bearings and graphitar bushings.

There are two different type ball bearings. One is the grease packed, double row bearing on the outboard side of the seal housing and the others are water lubricated, single row ball bearings located on each side of the pinion gear. The grease packed, double row bearing has performed very well with only one instance where grease leaked out of the bearing. The ball bearings on each side of the pinion gear were originally made of stellite. Early in life (August 1949) it was found that the bearings were performing in a rough and somewhat erratic manner. Inspection of the bearings indicated that wear had caused this rough performance. Since replacement bearings of the same type were not available, and delivery time was long, standard alloy carbon steel bearings were installed. These bearings had a high corrosion rate in themselves, in addition to being continually exposed to abrasive corrosion products from the seal cooling water which passes through the bearings on the way to the reactor.

Replacement stellite bearings were ordered and are now on hand.

There are two sleeve bearings in the SL-1 mechanism. One bearing is adjacent to the seals. This bearing in addition to being a support for the pinion shaft also metered the coolant water flowing to the reactor. No problems have ever been reported with the pinion shaft sleeve bearing. The second bearing is mounted into the lower part of the shield plug and it is used to guide the control rod extension shaft connector between the pinion gear and the control rod. This bearing also acts as a means for restricting the flow of cooling water into the reactor vessel. No problems have been reported with the extension shaft guide bearings.

The graphitar bushings are pressed into the control rod mechanism rack back-up roller. These bushings ride on a 17-4PH shaft. The only problem encountered with this bushing was in the one instance when it was found to be tight on the shaft and did not rotate freely and, therefore, restricted the rate of rod scram. The back-up roller was removed and the bore of the bushing reamed out to bring it up to design dimensions.

4) Pinion Gear, Rack and Rod Connector

Both the pinion gear and rack are made of 17-4PH stainless steel. These components have performed satisfactorily during their entire operating period and no problems have been reported. The pinion gear and rack on the #9 rod drive mechanism are in almost continual motion during operation, since this is the regulating rod.

The control rod connector shaft attaches to the ball joint on the end of the control rod and to the control rod connector extension shaft. During assembly and disassembly it was found that some galling had occurred between the actuating thread and nut and also between the ball joint and collet. Some of the parts were reworked by polishing and replating to prevent further galling. No further problems have been reported with respect to these components.

5) Control Rod and Control Rod Shroud

The two most important factors influencing the operation of a control rod within its shroud, besides adequate clearance, are alignment of the blades relative to the shroud and the assurance that there are no restrictions to movement of the blades within the shroud.

The factors which influence control rod to shroud alignment are dimensional tolerances (thickness, width, bowing and twisting) of the control rod and shroud in addition to the center line misalignment of the control rod drive extension (at the pressure vessel head level) and the shroud. Although the drawings provided by ANL do not specify all dimensional tolerances on the control rods and shroud, there is a nominal clearance between each side of a control rod blade and the adjacent wall of the shroud of .140 inches (giving a total clearance of .280 inches). The ball joint connection of the control rod to control rod extension and the flexibility of the control rod shaft extension assembly could accept some misalignment and still operate satisfactorily.

Reviewing the control rod and shroud design for unrestricted movement of a control rod blade, it is noted that there are eleven 2.0 inch diameter holes in each shroud wall in the core region. Also, the control rod blades are fabricated by welding the cladding around the outer edge of each blade. If a control rod rides against the side of the channel it could be possible for a moving control rod to hesitate as the blunt bottom edge of the blade rides over the lower edge of a hole. This type of sticking would also be sporadic because the control rods do not necessarily follow the same downward path at all times. The probability of this happening would increase if a control shroud were distorted inward. It is also conceivable that the shroud holes would permit pieces of the aluminum-boron strip that broke away from fuel assemblies to project through the holes into the control rod channel and produce temporary rod obstructions.

Although no actual measurements of channel width had been made by Combustion Engineering, it is possible that the channel width has decreased from its original dimension. A decrease in channel width could be caused by the lateral distortion of the aluminum-boron strips which are tack welded on opposite sides of each of the sixteen centrally located fuel assemblies and one side of the remaining twenty-four fuel assemblies. One of the strips on each of the centrally located fuel assemblies is a half length strip located on the lower half. The distorted aluminum-boron strips press against an adjacent fuel assembly on one side and against an essentially perforated shroud wall on the other side. However, such inward distortion of the shroud could not be a prime cause for control rod sticking for it was possible in most cases to successfully scram a control rod right after it was found to be sticking on a scram. A review of the rod sticking summary (Appendix A) shows sticking incidents are not reproducible and sticking generally occurred above 15 inches. If inward distortion were a prime cause for sticking, it would be expected to occur more frequently in the region below 13 inches, because only in this region are there aluminum-boron strips on two sides of the centrally located fuel assemblies. Also, No. 9 control rod, the central rod, would have been affected more than any other rod because it is completely surrounded by fuel assemblies each containing two aluminum-boron strips, however, its operational performance was best. It is also known that the lateral distortion of aluminum-boron strips increased with time. In the six months period prior to the last shut-down, No. 9 rod was successfully scrammed 130 times* with only one instance of sticking. This instance occurred November 28, 1960, when it hesitated momentarily at its drop height of 18.5" on scram.

In summary, it is believed that the control rod shrouds could have been distorted inward and introduce some frictional resistance to the over-all system, but that distortion alone was not a prime cause for sticking rods.

g. Mechanical Evaluation and Redesign

Combustion Engineering's contract included the design of a replacement core and rod drive mechanisms for the SL-1 facility based on the design for PL type plants. This PL type rod drive mechanism is shown in Figures 43 and 44. The design is basically a modification of the existing SL-1 drive mechanism. The modifications were made to overcome the problems experienced with the SL-1 mechanisms, and those design features that have proven successful were retained. Major design changes are as follows:

Scram Shock Absorption - The use of a buffer spring has been eliminated entirely to produce a more substantial and reliable design and to reduce the height. Instead, scram shock absorption is now provided by an elastic system consisting of a long connector bolt, the

*Data taken from Power History Log.

extension shaft, and the pinion housing. Each of these components has been designed to utilize their energy absorption capabilities. A dynamic analysis has shown that the maximum scram energy produced by a cold, dry, frictionless scram from full rod withdrawal could be absorbed in the system without causing the failure of the control rod or any component in the mechanism.

The SL-1 mechanism was not designed on this basis and could not meet this same scram condition, however, shock loading during normal operational scram never reached the maximum level of a cold, dry, frictionless scram because of hydraulic dampening and frictional resistance. The SL-1 system has inherently high frictional resistance. An analysis of the shock absorbing components is given in IDO-19003, "SL-1 Reactor Evaluation Final Report."

Mechanism Installation and Removal - The PL mechanisms have been designed to meet the requirement that they be individually removable and interchangeable. In addition, the design is such that removal can be accomplished with a minimum of mechanism disassembly. The pinion housing, pinion gear, bearings and shaft seal assembly do not require disassembly for vessel head removal. The coupling between the mechanism extension shaft and a control rod blade is a 1-1/4 inch fluted Acme thread. The control rod blade is not raised for the coupling operation.

Pinion Bearings - The pinion housing is designed to take either ball bearings or carbon-graphite bushings. Development tests to be performed with a PL lead mechanism (presently being assembled) are intended to optimize ball bearing material selection between stellite and AISI 440-C stainless steel, or graphite bushings. SL-1 experience has shown heavy wear and crud build-up on the original bearings.

Shaft Seal - The SL-1 floating ring, controlled leakage shaft seals have shown shaft and ring wear. This may be detrimental in two ways: first, the small clearances between shaft and rings and the associated wear provides high friction in the seal; and second, increased seal leakage and a consequent change in cooling water flow split between the seals and actuator. PL seal design will utilize increased clearance rings and as a back-up design, preliminary tests have been started (and will continue) with various other seal types, such as a face type seal.

Drive Package - The PL drive package design improvement will include a limit switch assembly containing more rugged cams, so that cam adjustments can be made easier and that they will maintain their adjustment during operation. Negator springs, springs whose shock absorbing ability were not needed and subsequently removed from SL-1 drive packages, have been eliminated in PL design. In addition, an improved material choice of 304 stainless steel for the gear pair coupling the drive motor to the clutch assembly has been incorporated.

General - Other design improvements include a shorter shaft distance between the floating ring seals and the pinion gear, and a larger diameter pinion gear bearing shaft to reduce seal shaft deflections and consequent binding. The accumulated effect of parasitic loads imposed at the pinion gear bearings, seal shaft bearings, and floating ring seals may be a contributing factor in rod sticking on the SL-1 mechanism, particularly at the higher rod elevations.

4) Higher Power Operation

The SL-1 reactor was operated at higher than design power to test the PL-2 condenser. This condenser is rated at a steam flow of 13,000 lbs/hr. The SL-1 operating at 3 MW can only provide 9000 lbs/hr., thus, the PL-type condenser tests required reactor operation at 4.7 MW.

Preliminary testing was accomplished on the air-cooled condenser to check the design capacity and the over-all performance. The initial tests were run at part load and straight through air flow. The testing was limited since permission had not been granted at that time to operate the reactor at power levels over 3MW. In addition, the damper control system controlling the inlet air temperature to the condenser had not been installed.

The complete damper control system was installed and checked out early in December 1960. A number of full load tests were run on the air-cooled condenser which indicated that the condenser would perform as designed. Air velocity and temperature traverses were made at the intake ducts to the building, inlet face of the condenser and the exhaust dampers. These showed the extent of air mixing, and the capability of the by- ass air to control inlet air temperature to the condenser.

During the full load tests, the reactor power level was approximately 4.54 MW with a steam generation rate of 13,550 lbs/hr. The steam flow to the PL condenser was 12,100 lbs/hr. With the inlet air temperature to the condenser controlled at 41°F, condensing pressure at 14.5 inches of Hg absolute, 14 x 10^6 BTU/hr was removed by the condenser. With the inlet air temperature of 65°F, and a condensing pressure of 21.2 inches of Hg absolute, 12.75 x 10^6 BTU/hr was removed. These results indicate that the condenser design is satisfactory.

During the higher power runs oscillations of the reactor neutron flux were observed. These oscillations are usually referred to as boiling noise. In order to determine the variations in the boiling noise with power level, measurements were taken on October 1, 1960, at 1, 2, and 3 MW.

Between November 2 and November 4, 1960, the power was increased from 3 MW to 4.7 MW in 20% increments. On November 2 the power was raised to 3.5 MW, on November 3 to 4.1 MW and on November 4 to 4.7 MW. The boiling noise was measured at each of these power levels. The six points in Figures 45 and 46 marked by circles show the amplitude and frequency of the oscillations at 1; 2; 3; 3.5; 4.1; and 4.7 MW.

On November 15, 1960, the cadmium shims were installed in Tee slots of rod positions No. 2 and No. 6. On November 17 power operation was resumed and new boiling noise measurements were taken at various power levels. It was found that for power levels below 4 MW the boiling noise was lower in amplitude and higher in frequency and above 4 MW it was higher in both amplitude and frequency than prior to the insertion of the cadmium shims.

On November 23, 1960, the reactor was operated at 4.7 MW. At the start of the run the outside rods 1; 3; 5; and 7 were at 24" and rod No. 9 was at 17.7 inches. For the initial 1000 sec. of this run the center rod was withdrawn and the outside rods were inserted to approach a banked position. During this time the amplitude of the oscillations was increasing. After 1000 sec. the amplitude of the oscillations was approximately 1 MW. At this time the motion of the rods was reversed i.e., rod No. 9 was inserted and rods 1; 3; 5 and 7 were withdrawn.

The magnitude of the oscillations seemed to decrease for about 50 seconds and then started to diverge again. Seventeen seconds later the oscillations peaks were off scale on the Offner recorder and remained off scale for 11 cycles corresponding to 4 seconds. At this time the amplitude of the oscillations decreased again to approximately 1 MW for 3 cycles, or 1 second. The oscillations diverged again and went off scale for 9 cycles, or 3 seconds, at which point the reactor was scrammed on an over-power signal. The scram setting during this test was at 5.7 MW, however, it is estimated that the peak power achieved was between 6 and 8 MW. The reactor did not scram at 5.7 MW due to the short duration of the over-power cycle compared to the delay time in the scram system relay.

This series of oscillations should not have harmed the core, or any of its components, since the average power during this time was not more than 4.7 MW and the integrated power in any cycle is not sufficient to cause damage. Immediately after the scram, the effluent gas activity was checked and no increase observed. Following the startup of the reactor, control rods 1; 3; 5 and 7 were recorded at 20" and No. 9 at 18.2" withdrawal at a steam flow of 7700 lbs/hr. Using the rod calibration curves, (Figure 27) and the steam flow vs. rod bank curve, (Figure 29) these positions were corrected to a bank height of 19.2" corresponding to 8000 lb/hr. steam flow. Prior to the stability test the rod bank positions at 8750 lb/hr. steam flow were 19.2" for 1; 3; 5 and 7 and 19.8" for No. 9. These were corrected to a bank position of 19.2" corresponding to 8000 lb/hr. It was, therefore, concluded that there was no change in rod bank position as a result of the oscillations and hence no gain in reactivity, or loss of boron. This is further confirmed by the data plotted in Figure 32 which shows no change in the rod bank position on or subsequent to November 23.

This indicated that the reactor could not be operated stably with banked rods at 4.7 MW, therefore it was operated with the center rod down and controlling and the other rods fully withdrawn. As Figure 47 indicates, the amplitude of oscillations at 4.7 MW with programmed rods is appreciably lower than with banked rods at 4.2 MW.

The only conclusion which might be drawn from these boiling noise studies to date is that the noise appears to increase with both power level and radial peaking factor.

5. Coolant History

SL-1 water quality is maintained and adjusted by filter and ion exchanger purification. Incoming raw water passes through a filter into a mixed bed ion exchanger and then to a 1000 gallon stainless steel makeup storage tank. Makeup water is introduced into the reactor system through the hot well and feed pump.

A boiling water reactor acts as a concentrator of non-volatile impurities by evaporating pure steam. Three to five gpm of SL-1 reactor water is tapped off into a by-pass purification system which consists of a filter, a regenerative cooler, parallel cation and mixed bed ion exchangers and a return line to a feedwater filter. The conductivity and pH of the water from the by-pass purifier are measured continuously and the values are recorded on the control room panel board.

The reactor water specifications are as follows: pH 6.5 to 7.5 Resistivity, greater than 500,000 ohms.

There are no other requirements; however, the chloride and oxygen levels are kept to a minimum through proper operation of the feedwater and by-pass purification systems. A decrease in resistivity below 500,000 ohms indicates that the mixed bed resin is exhausted. An increase in pH indicates the cation resin is exhausted.

a. Checks

The following checks were initiated to insure that the reactor water met the required specifications at all times:

Water Activity in the Reactor - The purpose of this check is to keep track of buildup of long-lived activities due to corrosion, and to check on clad rupture by alpha-count level. Sampling is done daily during reactor operation. The sample is taken from sample Tap No. 2 in the purification system.

Decontamination Factor - The purpose of this test is to check for water carry-over and steam purity. Sampling is done weekly during steady state operation. Extra samples are run during startup, or if water level or power level is changed. The sample tap is on the main steam line.

Raw Water - This is tested to prevent impurities from being introduced into the reactor from the raw water supply either through the makeup system or the shutdown cooler. It also serves as a check on the probable life of the makeup water demineralizer. Sampling is done once a month at the raw water demineralizer inlet.

Makeup Water - Tested to keep track of demineralizer behavior and determine amount of impurities which might build up in the reactor water from this source. The frequency of sampling depends on makeup water requirements, but is ordinarily done every two weeks. The sampling point is the outlet of the raw water demineralizer.

Condensate Feedwater - Tested to keep track of solid carry-over and volatile impurities such as oxygen and chloride in the feedwater. Samples are taken at the hot well at least once a day.

Reactor Water - Tested in order to identify corrosion products in reactor water and to check on impurity trends. These impurities are determined in the same sample taken for reactor water activity measurements.

If for some reason the water quality was determined to be below standards, the reactor was run at reduced power in order to allow the bypass purification system to clean up the water. In one case, it was necessary to shut the reactor down in order to obtain high purity water.

The main influences on reactor water quality control are oxygen and chloride levels, suspended and dissolved solids in the form of corrosion products and fission products, and fission and corrosion product carry-over in the steam. These items are discussed below:

Oxygen and Chloride - The amount of dissolved oxygen is determined in the reactor water and condensed steam. Chloride content is determined in the feedwater and reactor water. The oxygen in the condensed steam averages about 23 ppm at 3 MW. This high value is primarily due to the radiolytic decomposition of water into a stoichiometric mixture of hydrogen and oxygen. The amount of decomposition is a function of reactor operating pressure, power level, pH and impurities in the water. As a result of radiolysis, sizable volumes of gas must be handled by the condenser system air ejectors.

The dissolved oxygen in the reactor water has been maintained under 0.5 ppm throughout operation by Combustion Engineering. During the PL-2 condenser tests, when the reactor power was raised to a maximum of 4.7 MW the reactor water dissolved oxygen was 0.48 ppm. Under normal operating conditions at 3MW the following data was obtained:

Sample Point	Chloride ppm	Dissolved Oxygen ppm
Reactor Water	0	0.23
Feedwater	0	0.16
Purification Effluent	0	0.13

Oxygen and chloride levels are extremely important from the point of view of the corrosion of X-8001 aluminum and other system materials such as 304 stainless steel.

Suspended and Dissolved Solids - Total and dissolved solids are determined in the reactor water and raw water.

Analysis of reactor water to date has shown a total solids content in the range of 4 to 5 ppm. Resistivity values indicate that dissolved solids are about 0.5 ppm. The remaining solids content is partly organic and party suspended solids. The suspended solids which are carried over during boiling or introduced in the makeup water are removed from the system in the feedwater filter. The remainder of the solids are removed in the by-pass purification system.

Tables VI and VII show the relative quantities of elements present in the feedwater and purification filters. The activity of the feedwater filter element is attributed to the presence of Cr^{51}, Zr^{95} and Nb^{95}. Cr^{51} contributes more than 98 per cent of the activity in the feedwater filter. Cr^{51} and Zn^{65} contribute about 92 per cent of the activity in the purification filter. These and the remainder of the isotopes contributing to activity in the purification filter are summarized in Table VIII.

TABLE VI

ELEMENTS PRESENT - FEEDWATER FILTER

Major	Minor	Trace	
Fe	Al	Co	Pb
	Cr	Cu	Sh
	Ni	Mg	Ti
	Si	Mn	V
		Mo	Zr
		Nb	

TABLE VII

ELEMENTS PRESENT IN PURIFICATION FILTER

Major	Minor	Trace	
Al	Cu	Co	Ni
Cr	Si	Mg	Pb
Fe		Mn	Sn
		Mo	Ti
		V	Zr
			Nb

TABLE VIII

ELEMENTS CONTRIBUTING TO PURIFICATION FILTER ACTIVITY

Isotope	% of Activity
Cr-51	77.6
Zn-65	13.64
Ce-141	2.5
Fe-59	2.6
Sb-124	1.05
Ru-103 or 106	0.58
Ba-140	0.36
La-140	0.36
Co-60	0.19
Zr or Nb-95	0.30

These isotopes are fission products and corrosion products from the aluminum and stainless steel. The high Zn^{65} activities cannot be attributed to either of these alloys.

Fission Products and Radioactive Corrosion Products - Radioactivity in the water due to fission products and activated corrosion products is one of the major problems in the purification of the water. Fortunately, the boiling process tends to confine non-volatile materials in the reactor vessel. The ability to thus confine radio-activity is expressed as the decontamination factor, DF = reactor water activity in c/m/ml divided by condensed steam activity in c/m/ml.

The decontamination factor measured during the 1000 hour test in June and July 1959 varied from 98 to 1.9×10^3 depending on steam flow and purification flow.[12] Main steam flow ranged from 5000 to 8950 lbs/hr.

The average decontamination factor was about 10^3 compared to about 10^4 for Borax III, Borax IV, and VBWR.

Later tests were performed in March, 1960, in which the decontamination factors between the reactor water and steam as well as across the ion exchange columns were determined as a function of steam flow rates. During this test the steam decontamination factor averaged about 10^4, more in line with Borax, EBWR, and VBWR[25]. This decontamination factor was confirmed during the power extrapolation studies performed in October and November, 1960.

The isotopes in the reactor water are non-volatile fission products and corrosion products. The major activities found in the reactor water are Mn^{56}, I^{131}, I^{132}, Na^{24}, and Cr^{51}. In all cases of analysis to date, Na^{24} has accounted for over 95 per cent of the total activity. A typical set of activity measurements is given below:

Isotype	d/m/ml	Per Cent of Total Activity
Na^{24}	1.3×10^6	97.2
Mn^{56}	3.0×10^4	2.2
I^{131}	4.2×10^3	0.3
I^{133}	3.2×10^3	0.2
Cr^{51}	0.92×10^3	0.02

The same trends noted above were observed during higher power tests. The major fission product activities were I^{131}, Sr^{89}, and Ba^{140}. The major corrosion product activites were Na^{24}, Mn^{56}, and Cr^{51}. An increase in fission products in the reactor water as a function of power level was noted, however, the I^{131} activity appears to remain constant and this isotope accounts for 60 to 70 per cent of the fission product activity in the reactor. The high Sr^{89} levels must be attributed to residual strontium in the makeup water. The well water at the S1-1 has a high concentration of strontium, and Sr^{89} may be produced through the (n,α) reaction on stable Sr^{88}. When the fission product yields of Sr^{89}, Sr^{90}, and Sr^{91}, are considered the activities of these three fission products should be of the same order of magnitude.

In addition to the carry-over due to entrained moisture and dissociated water, the activity carried over due to volatile fission products was determined. This work was performed during the period September 1 to December 30, 1959.

Activity in the steam consists of volatile fission products and some Na^{24} which was probably carried over with entrained moisture in the steam. The gross activity was 3.2×10^3 d/m/ml. The major activities were I^{131}, I^{133}, Xe^{135}, and Kr^{88}. The total activity of air ejector gases during the test period was 5.3×10^4 d/m/ml. The gas activity was a mixture of krypton and xenon isotopes. The major activities were due to Xe^{138}, Xe^{135} and Xe^{135m}, Xe^{133} and Kr^{88}.

During early Sl-1 tests, it was found that 66.5×10^{-2} curies per day of Xe^{138}, 0.9×10^{-2} curies per day of Xe^{133}, and 8.0×10^{-2} curies per day of Kr^{88} were being emitted from the air ejector. This gas released was probably due to surface contamination of the fuel plates. It has been calculated that a few tenths of a milligram of U^{235} on the fuel plates can account for the Xe and Kr activities. Three spare SL-1 fuel elements were analyzed for uranium surface contamination. The presence of alpha activity was confirmed. It is believed that this activity was not introduced during fabrication, since the fuel assemblies were inspected before shipping, but was probably from air borne material from the storage vaults.

Fission product activity in the water, the I^{131} activities, and air ejector gas activities point to a delayed release of fission products. Whether this delay is due to diffusion through leaks in the fuel element cladding, or some other mechanism, is not known. The ratios of short to long-lived gas activities are below the theoretical ratios for recoil of fission products into the water.

In spite of these early contamination problems the air ejector gas activity has remained fairly constant since Combustion Engineering has operated SL-1.

b. Operational Problems

<u>Oil in Well Water</u> - During the 500-hour acceptance test of the SL-1 reactor, difficulty was experienced in maintaining proper water

resistivity and pH. Later operation in January and February, 1959, led to resin break-through in 5 to 6 days of operation. It was observed that oil was present throughout the system. This oil was preventing proper operation of the raw water demineralizer and the by-pass ion exchanger system. The oil was traced to the deep well water pump. This pump was modified so that the oil would not drip into the water supply. Oil was cleaned out of all accessible parts of the system and purged by water flow from others. As an added precaution, a diatomaceous earth oil filter was installed in the plant makeup water line. Oil contamination of the water is now 2 ppm or less.

Ejection of Resins into Reactor Water System - In early June, 1960, the lower screen of the mixed bed resin containment vessel was ruptured during a routine resin change. The resin in the column was injected into the reactor water system. Two days of flushing and low power operation was necessary to clean the resin out of the system. During this clean-up, the feedwater filters, purification system filters, and feedwater pump strainers were changed. Glands on the feedwater pumps were repacked and gland seals were replaced. The mixed bed containment vessel was removed and repaired, and new resin was put into all purification columns.

Inadequacy of Pre-cooler to the Ion Exchange System - The purification system was limited to 175° due to the thermal stability of the mixed bed resin. The heat exchange cooler for the by-pass purification system was originally designed so that part of the feedwater was put through the cooler in order to cool the reactor water from 420°F to less than 175°F. The temperature of the feedwater was too high to achieve the desired cooling and the flow through the mixed bed resin was as high as 190°F at times which, of course, resulted in reduced resin life. In addition, it was necessary to limit purification water flow to 2 gpm.

This system was modified to allow raw water to flow through this cooler. The resulting increase in heat exchange capacity now permits the purification system to operate at its designed 5 gpm flow.

Boron Loss Evaluation - An inspection of fuel assemblies in the SL-1 indicated a severe corrosion of the aluminum-boron strips. In many cases the lower portions of the strips had dropped off, and insoluble material was released into the reactor. This material settled to the bottom of the vessel. Since the sintered boron powder is insoluble no boron has been noted in the purification systems. The material may be dense enough so that it is not picked up in the by-pass purification system filters, since the reactor water outlet to the purification system is several feet above the bottom of the reactor vessel.

One or two spectrographic examinations of reactor water indicated that boron was present in the part per billion range. The accuracy of boron analyses at these levels is highly inaccurate and no assumptions as to boron loss should be made on this basis.

Reactor Water Specifications - During the period of August 2 to 9, 1960, the reactor was operated at low power for two days for maximum primary water purification because the water quality had dropped below specifications after a series of training scrams. During the period June 7 - June 14, 1960, the water had to be cleaned up due to malfunction No. 27 (injection of resin into the reactor water). On April 21, the reactor was secured for 26 hours because of poor quality reactor water obtained after malfunction No. 23 (false high water level scram). The water quality fell below specifications during the down period and start-up period following this scram.

6. Reactor Equipment Operating Experience

a. Head Gasket Leak

On April 2, 1959, the reactor vessel closure seals developed a leak. Upon removal of the head, inspection of the two gaskets revealed that the outside retaining ring on the outer gasket was out of the gasket groove in a five degree arc[26]. This can be seen in Figure 48. Apparently, the outer gasket was oversized and did not seat properly during initial assembly.

In the process of removing the vessel head , the stud nuts were found to be tightened excessively. Elongation measurements revealed that the average stud elongation was .025 inches, rather than the design elongation of .006 inches. As a result of this measured indication of excessive initial bolt-down, it was decided to remove all forty-eight studs for inspection.

Removal of all the studs was very difficult and two of the studs had to be drilled and burned out. Stud removal problems caused the shutdown to extend to almost a month. New studs of an improved design were procured for all forty-eight bolt positions and these were used when the head was replaced.

New gaskets were installed on April 23, 1959. The head was bolted down to the design bolt load (bolt elongation of 0.006 in.) and hydrostatically tested at 600 psig. The gaskets showed leakage at the rate of 150 cc per hour, which was above the minimum allowable rate. To reduce the leakage rate, an additional bolt load was impressed (average bolt elongation of 0.010 in.). Leakage for this bolt load was as follows:

600 psi	-	12 cc per hour
400 psi	-	4 cc per hour
300 psi	-	No measurable leakage

Because there was no measurable leakage at the operating pressure of 300 psi, the replacement gaskets were left as installed. Their performance subsequently has been satisfactory.

b. Refueling

Fuel element transfer from the reactor core to one of three fuel storage wells is accomplished with the fuel transfer coffin, Figure 49. The coffin is a steel enclosed, lead filled cylinder 28 inches in diameter by 56 inches high. An integrally mounted hand operated hoist raises or lowers a single fuel element within the coffin cavity. A drawer-type gate slides open to permit entry to the cavity. The fuel gripper which is hung from the hoist cable and actuated by a gripper release cable, attaches to a fuel element.

The fuel transfer equipment was tested during the week of April 9, 1959. An unirradiated fuel element was transferred from the fuel well to the coffin and then returned. An irradiated element was transferred from the reactor core to the coffin and returned. These two operations revealed several problems. First, the fuel element could not be completely withdrawn into the coffin because of sharp corners within the coffin cavity. Second, alignment of the coffin over the reactor head openings and fuel well cover plate were difficult. Third, radiation streaming was monitored from around the gate and the bottom of the coffin when the fuel element passed these points.

The fuel element transfer coffin was modified as a result of the above tests. The shoulder between the fuel element cylinder and the gripper chamber was chamferred to prevent the gripper from catching when it is withdrawn into the coffin. A fixed locking pin was installed on the coffin hoist to prevent the gripper head from accidently unlatching a fuel element in the coffin. A steel funnel was fabricated for the fuel storage well to facilitate coffin alignment. The hoist control was replaced with a straight rod crank when the former unit broke in operation. The radiation streaming problem remained.

Subsequent fuel handling operations with the fuel transfer coffin continued to reveal difficulties. In one instance (week of June 20, 1960) a dummy element became disengaged within the coffin. It was found that the gripper hoist cable and release cable had kinked and coiled in the gripper receptacle. This prevented full insertion of the element into the coffin. Further, it disengaged the gripper when the load was slightly relaxed.

7. Plant Malfunction Report Summary

The following is a summary of malfunctions which occurred at the SL-1 facility. The summary was taken from malfunction reports, 1 through 38. It should be noted that the SL-1 was operated as a training facility and as such experienced a very large number of startup and shutdown cycles. This excessive cycling no doubt contributed to the frequency of some malfunctions. Reports are written on the basis of criteria provided by the Atomic Energy Commission for SL-1 malfunction reports[32], as follows:

(1) An occurrence resulting in a reactor accident or physical damage to the core or primary plant components.

(2) An equipment failure which causes a reactor scram, or plant shutdown.

(3) Repeated failure of equipment to remain in adjustment.

(4) An overexposure of personnel to radiation in excess of established tolerances.

(5) A fire or normal industrial accident that affects power plant operation.

Number of Malfunctions	Mechanical Equipment	Hours Shutdown
1	Head flange gasket leak	356
4	Control Rod Drive Mechanisms	48
	a. Mechanism binding caused No. 7 rod to hangup at 4 inches when dropped from 30 inches.	
	b. Negator spring broke loose and damaged limit switches.	
	c. Steam leaked from seal housing of one mechanism and from cooling water line fittings of another mechanism.	
	d. Seals leaked in three mechanisms.	
3	Ejectors	78
	a. Gland ejector leak-off system lost vacuum because of clogging of the ejector orifice.	
	b. Flanged fittings in after-condenser leaked.	
	c. Moisture froze in ejector discharge line.	
3	Valves	33
	a. Reactor venting valve froze open.	
	b. Pressure gage isolation valves in steam line leaked.	
	c. Main steam inlet isolation valve leaked.	
2	Turbine Governor	7
	a. Throttle valve had bent stem.	
	b. Turbine governor was improperly adjusted.	
1	Mixed bed resin screen ruptured	162
1	Three weld points in Main Steam System leaked	33
1	Condenser exhaust dampers slipped on shaft	
1	Condensate circulating pump shorted out	2
	Electrical Equipment	
1	Fan motor failed when insulation broke down due to excessive ambient temperature.	61
4	Station auxiliary breaker tripped out	5

Number of Malfunctions	Electrical Equipment (Continued)	Hours Shutdown
1	Fan Motor Breaker tripped	2
1	Utility Bus Breaker tripped	
	Control Systems	
2	Liquid Level Indicator	3-1/2
	a. Vacuum tube failed in Hayes Liquid Level Indicator.	
	b. Hayes transformer coil failed	
1	Vacuum tube and resistor in high voltage supply failed.	6
2	Insulation breakdown	17
	a. High temperature caused shorting of high voltage supply to Channel II.	
	b. High voltage line shorted when insulation broke down.	
1	Tube failed in Power Supply to Nuclear Channel I	1/2
	Operator Error	
1	Reactor water level dropped below top of core	
1	Circuit unintentionally shorted	1
1	Wrong fuse pulled	1
1	Turbine throttle valve not fully opened	3
1	Reactor water level indicator incorrectly installed	1
1	Purification pump incorrectly repaired	25
	Steam supply to turbine reduced	
	Resin introduced in hotwell	40
38	Total for all categories	898

8. Significant Events in SL-1 Operating History

Date	MWD	Events
2/5/59	68	CE assumes operating responsibility for SL-1 plant.
2/6/59 to 2/11/59	68	Selected fuel elements visually inspected in the reactor vessel by CE and ANL representative. Fuel element discolorations observed
2/5/59 to 3/29/59	68	(1 Performed routine maintenance and plant modifications (2) Oil in reactor water from deep well pump cleaned up 3 Interim Operating Manual prepared
3/6/59	68	Demonstration operation at power for two hours for CE Nuclear Safeguards Committee
3/30/59	68	Commission approval received for CE operation
3/31/59	68	Cold critical operation (a) Nuclear channel ranges checked (b) Relative cold critical rod worths determined
4/1/59	68	Begin power operation
4/1/59	70	Cold, hot, and operating critical rod bank positions measured
4/3/59	70	Plant secured because of head gasket leak on reactor vessel
4/3/59 to 4/23/59	70	(1) Head gaskets replaced (2) Hold-down boxes added to all but two outer clusters (3) Extension spool on Rod No. 9 removed
4/23/59	70	Begin five day shift operation at power
4/27/59	73	Equilibrium xenon (2.5 MW) rod bank position measured
4/30/59	80	(1) Relative control rod worth evaluation at full power (2) Reactor period recorded while heating to temperature and pressure to determine boiling effects and to measure the transient experienced during hydrogen venting operations (3) Decontamination test started
5/4/59	81	Control rod bank vs. power measurements with no xenon present

Date	MWD	Event
5/7/59	86	(1) Water temperature effect on control rod bank measured (2) Decontamination factor test continued (3) Analysis of stack effluent gases test started
5/14/59	91	Relative position of control rod cadmium and fuel checked by measurement. This measurement henceforth used to set control rods at zero position
5/19/59	91	(1) Various cold critical rod positions measured (2) Rod bank during heat up recorded
5/20/59	91	Control rod mechanism for rod No. 7 removed and shipped to Windsor for analysis of sticking operation
5/21/59	91	(1) Checked out new rod No. 7 mechanism (2) Continued water chemistry tests
6/1/59	102	(1) Various cold critical rod positions measured (2) Rod bank during heat up recorded (3) Rod bank for 22 hour xenon buildup measured
6/4/59	104	Begin 1000 hour test
6/5/59	105	Equilibrium xenon (2.5 MW) rod bank position measured
6/18/59	133	Rod housing operating temperatures measured on rods No. 3 and No. 7
7/1/59	162	Critical rod positions determined for various hot operating conditions
7/1/59	162	Intercomparison of side rods calculated
7/3/59	167	Feedwater temperature effect on reactivity measured
7/16/59	195	Ended 1000 hour test
7/20/59	195 to 200	(1) Various cold critical rod positions determined (2) Rod bank during heat up measured (3) Various hot critical rod positions measured and hot rod worth evaluations (4) Various rod positions vs. reactor power measured
7/27/59	200	Xenon decay measured
7/27/59	200	Plant secured for maintenance
8/10/59 to 8/11/59	200	22 hour demonstration run for the Military Liaison Committee

Date	MWD	Event
8/27/59	200	Visual inspection of fuel elements - boron side plate buckling discovered
8/31/59	200	Control rod calibrations - rods No. 3 and No. 7
9/8/59	206	Initiate two man crew operation
9/23/59	213	Add first instrumented fuel element - 41 element core (1) Element #6 moved from position 45 to position 87 (2) Instrumented element #63 placed in position 45
9/24/59	214	Instrumented fuel element test
10/8/59	229	(1) Shutdown to remove instrumented element and one additional element for future hot cell inspection- 40 element core (a) Instrumented element #63 removed from position 45 (b) Element #6 moved from position 87 to position 45 (c) Element #38 removed from position 55 (d) Element #42 moved from position 66 to position 55 (e) New element #62 placed in position 66 (2) Captive key bypass switches installed in scram circuits with new scram on Channel IV
10/12/59	230	Begin seven day shift operation
11/20/59	317	Xenon decay measured
11/20/59	317	Cold, hot, and operating critical rod bank positions measured
11/25/59	323	Shutdown for maintenance and inspection
12/7/59	324	Renew seven day shift operation
12/7/59	324	Equilibrium xenon measured
12/23/59	364	Shutdown for annual maintenance - all major items overhauled
2/3/60	364	Begin seven day shift operation
2/19/60	385	Shutdown for trainee testing
2/29/60	385	Renew seven day shift operation
2/29/60 to 3/29/60	385 to 463	(1) Hydrogen buildup test performed (2) Equilibrium xenon measured (3) Decontamination factor test continued (4) Steam Quality Test (5) Water Decomposition Test

Date	MWD	Event
4/1/60	469	Shutdown for plant maintenance
4/1/60	469	Rod drop performance test for design evaluation
4/8/60	469	Control rod mechanism for rod No. 9 disassembled and inspected for signs of wear for design evaluation
4/9/60	469	Renew seven day shift operation
4/9/60 to 5/26/60	469 to 587	(1) Decontamination factor test continued (2) Fission break monitoring test (3) Water decomposition test
5/26/60	587	Shutdown for decay heat test and plant maintenance
6/11/60	588	Commence seven day shift operation for Cadre training
6/25/60 to 6/26/60	608	Shutdown for NRTS open house
6/29/60	613	Hot criticals to determine best detector location for startup
7/11/60	639	Shutdown for maintenance (1) Checked grid plate bolt tightness (2) Inserted test coupons (Ag-In-Cd) and fluxwires
7/16/60	639	Cold rod drop tests for design evaluation
7/17/60	640	Resume power operation
7/31/60	660	Hot rod drop tests for design evaluation
7/31/60	660	Shutdown to remove test coupons (Ag-In-Cd) and two fluxwires
8/14/60 to 8/21/60	680	Shutdown (1) Inspect fuel elements, corroded boron side plates discovered and sections of plates from elements #42 and #8 removed from core (2) Inserted second instrumented fuel element (a) Element #42 removed from position 55 (b) Instrumented element #1 placed in position 55 (3) Fluxwires and test coupons (Ag-In-Cd) inserted into core

Date	MWD	Event
8/21/60	680	Control rod No. 1 stuck cold; was disassembled, inspected and reassembled, then found sticking above 20 in. Rod travel was then limited to 20 in. for subsequent operation
8/24/60	680	Resume operation
8/24/60 to 8/25/60	680	Control rod worth experiments
9/11/60	714	Rod drop tests for design evaluation
9/11/60	714	Shutdown to remove fluxwires and test coupons (Ag-In-Cd)
9/13/60	714	Cold critical and low power rod bank measurements
9/13/60	714	No. 5 rod calibrated
9/14/60	714	Rod bank vs. power measurements
9/16/60	718	Hot (Zeropower and 2.5 MW) critical rod bank measured
9/23/60	731	Equilibrium xenon (2.5 MW) rod bank position measured
10/3/60	755	Begin hot checkout of PL loop
10/28/60	822	72 hour test of PL loop at power
11/1/60	834	Boiling noise study at 1, 2 and 3 MW
11/2/60	837	Increase power to 3.5 MW
11/3/60	841	Increase power to 4.1 MW
11/4/60	845	Increase power to 4.7 MW
11/5/60	849	Equilibrium xenon measured
11/6/60	850	Shutdown for maintenance
11/9/60	850	(1) Test coupons (Ag-In-Cd) installed (2) Burr on shroud for Rod No. 1 smoothed with special tool
11/10/60	850	Dummy rod No. 4 installed
11/14/60	851	Seal test on Rod No. 4 performed
11/15/60	853	Inserted cadmium in Tee slots of rod positions #2 and #6

Date	MWD	Event
11/16/60	853	Critical rod bank position measured at 180°F
11/17/60	854	Resume power operation
11/21/60	863	3 MW stability test
11/30/60	879	Stability tests continued at various power levels
12/4/60	889	Equilibrium xenon measurement
12/9/60	898	Transient Test
12/15/60 to 12/20/60	912 to 925	Continuous operation at 4.7 MW
12/21/60	928	PL condenser performance test
12/23/60	932	Shutdown for maintenance
12/23/60	932	Equilibrium xenon 2.56 MW critical rod bank position measured

III. CHRONOLOGY OF ACCIDENT

A. REACTOR PLANT PRIOR TO DECEMBER 23, 1960 SHUTDOWN

On December 23, 1960, the reactor had been operating at 2.56 megawatts. The control rods were at their expected elevation for power and life conditions of the reactor and there were no unusual instabilities or malfunctions reported either in the operating log, or in later interviews with the Cadre and operating crew. For the last reactor shutdown it was required that each control rod be scrammed individually. With the normal cooling flow to the control rod seal housing, two of the five control rods (Nos. 5 and 9) dropped clean. The three remaining rods, which stuck at various elevations, required a power assist from the rod drive motors in order to go in. All control rod drive mechanisms were alike with the exception of the seal assembly on No. 9 which contained a face type seal installed in November 1960.

A detailed operational history of the control rod drives is covered in Section II C 3. Although the pre-installation testing of control rod mechanisms was satisfactory there have been sporadic instances of control rod sticking since early operation of the reactor in 1958. In the last month of reactor operation the incidence of control rod sticking had increased markedly.

In November, 1960 a sixth rod drive mechanism was installed for testing. This was located at the unused No. 4 Tee rod location. The dummy rod used with this mechanism was reduced in size in order to fit through the port in the head. The dummy rod was all aluminum with no neutron absorber content, thus having essentially no reactivity value.

It is not known to what extent the aluminum-boron strips had disintegrated and left the core, or whether there was any real loss of boron from those that had. The condition of the fuel assemblies and of the aluminum-boron strips was discussed previously under Section II C 1, "Metallurgical History of the Core."

Six cadmium strip assemblies (two sets of three each) had been installed in Tee rod positions 2 and 6 for added shutdown margin. These were also described in Section II C 1, "Metallurgical History of the Core."

The reactivity picture of the core has been presented in Section II C 2, "Reactivity History and Analysis of the Core."

The effects of distorted aluminum-boron strips on the core structure are not known. The greatest effect on control rod channels would be No. 9 since it is surrounded by fuel assemblies having two aluminum-boron strips on each. This control rod, however, had no record of sticking for six months prior to the incident either during operation or when lifting to assemble the control rod drive mechanisms.

Figure 8 is a plan view of the core as it appeared just prior to the December 23 shutdown. The forty active fuel assemblies, nineteen dummy assemblies, and one source assembly are shown positioned within the core structure. The thermocouple leads may be seen emerging from the centrally located, instrumented fuel assembly. The first overlay to the core plan view shows the positioning of the five active cruciform control rods, the Tee shaped dummy control rod, and the six cadmium shims. The second overlay provides a phantom view of the reactor vessel head. The head is shown with six open ports, corresponding to those presently open in the reactor and through which pictures have been taken. With this arrangement the pictures taken through the ports can be compared with this drawing which shows the position of equipment prior to the excursion.

B. WORK PERFORMED DURING THE SHUTDOWN PERIOD

The reactor was shutdown and the plant secured on December 23, 1960. The purpose of the shutdown was for the installation of 44 flux wire assemblies. A flux wire assembly consists of an aluminum support tube, an aluminum flux wire positioning rod and end plug, and a number of .020" dia. by 3/32" long, 0.5 weight per cent cobalt-aluminum wire segments. The flux wire segments are positioned in holes drilled through the aluminum rod at various elevations. The aluminum rod is contained within an aluminum support tube which is positioned between fuel plates in the core. These assemblies are the same as have been used before in determining neutron flux distribution. In order to install the flux wire assemblies it was necessary to remove control rod drive mechanisms and cover plates from the pressure vessel head ports to gain access to the core. In addition to the above, routine plant maintenance was also to be performed.

During the four day period from December 27 through December 30, 1960, the following work was performed: routine maintenance, instrumentation calibration, modification of the condensate pump with accompanying valving, piping and controls, addition of a new type of valve to the auxiliary steam system, and minor modifications to the plant load condenser system.

Shortly after midnight, the morning of January 3, 1961, a three man military operating crew started preparation of the reactor for the installation of flux wire assemblies. The work performed by this crew was recorded in the operating log and is as follows:

Time	Action
"0001	Placed system temperature, water level, reactor pressure, FW flow, purification temperature, indicating records in operation. CAM, H and F counting and RAM placed in operation.
0010	Started precoating oil filter. Started adding water to Hotwell and reactor.
0045	Started making demineralized water; placed shutdown cooler in operation.
0115	Removed rod drive units and extension.
0145	Removed rod #4 test rig.
0120	Jumpered purification pump interlock.
0140	Completed cold iron watch check list.
0145	Reactor water level 5' 0".
0200	Water sample tap #2 Ph -5.9; resistivity .65 $\times 10^6$.
0205	Purification system on line 4 gpm mixed bed.
0300	Top hat and shield block removed from around reactor head; established shoe cover area within blocks.
0400	Seal units removed from rods 1-3-4-7-9.
0600	Water sample tap #2 Ph -6.2; resistivity 1.4 $\times 10^6$.
0745	Thimbles removed from 1-3-4-7-9. Removed #1 thimble so plug #2 could be removed.
	Removed #3 thimble so #4 thimble could be removed.
	Housings removed from #4-7-9.
	Made 604 gallons demineralized water. Resins depleted.
	Removed plugs #2 and 6.
	Placed two lights in reactor.
	Placed finger tool and hook on operating floor."

The work performed between 0800 and 1600 on January 3, 1961, was not recorded in the log, however, the work performed on this shift consisted of the following:

1. Checked gaskets and clearances.
2. Checked in supplies and did necessary work in Mechanical Shop.
3. Changed #4 seal housing.
4. Removed #4 shield plug from reactor and exhanged with a spare.
5. Changed raw water demineralizer resins.
6. Moved two silver-indium-cadnium coupons and placed third in a ten gallon bucket of water and stored in low level room. Placed test samples of tubing in the reactor.
7. Inserted a total of 44 flux wire assemblies in their prescribed core locations.

C. STATE OF ASSEMBLY OF REACTOR ON JANUARY 3, 1961, AT END OF DAY SHIFT

The top of the vessel head at 1600 hours on January 3, 1961, was in the following condition:

Head Opening No.	Condition
No. 1	The control rod drive mechanism housing was removed and the shield plug assembly was in place. It is not known whether the scram stop washer and nut were in place.
No. 2	This head opening is a blank port which is sealed by a cover plate. A shield plug is welded to the underside of the cover plate. This port was open to enable insertion of flux wires.
No. 3	The control rod mechanism housing was removed although the shield plug assembly was left in place. It is believed that the scram stop washer and nut were in place.
No. 4	This head opening provides access for a test control rod mechanism. The mechanism had been completely disassembled and the port was open.
No. 5	The control rod drive mechanism had not been disassembled.

No. 6 This head opening is a blank port and is sealed the same as head opening No. 2. It was open to provide access for the insertion of flux wire assemblies.

No. 7 The control rod drive mechanism housing was removed and the shield plug assembly was in place. It is believed that the scram stop washer and nut had been removed.

No. 8 Thermocouples that were located in the instrumented fuel assembly (in position #38) were routed through the cover plate of this head opening. The cover plate was bolted down since it was not necessary for it to be removed.

No. 9 The control rod mechanism housing and shield plug assembly were not in place.

Two additional head openings contained water level indicators which were not disassembled. The reactor water level had been raised until the water level was just below the under side of the head.

Flux wire assemblies had all been installed in the core and the location of two silver-indium-cadmium coupons had been changed. One coupon had been removed by the crew and had been placed in a ten gallon bucket of water and stored in the low level room.

All parts of the control rod drive mechanisms and cover plates that were disassembled and tools used in process were on the operating floor outside the shield blocks. Small mechanism components were stored in containers. Further work was performed on equipment external to the reactor in preparation for reassembly and reactor startup.

D. DESCRIPTION OF THE INCIDENT

The instructions for the night of January 3, 1961, issued by the plant superintendent in the night order book were as follows:

"1. Perform a reactor pump down - procedure No. 54.
2. Reassemble control rods, install plugs, place shield blocks, leave top shield off.
3. Connect rod drive motors.
4. Electrically and mechanically zero control rods.

5. Accomplish control room and plant startup check lists.
6. Perform cold rod drops.
7. At 300 psi pressure check for leaks, replace top shield plug.
8. Perform hot rod drop tests.
9. Accomplish a normal startup to 3 MW operation."

The control room operating log book contains a single entry as follows:

"Pumped reactor water to contaminated water tank until reactor water level recorder came on scale. Indicates +5 ft. Replacing plugs, thimbles, etc., to all rods."

Presumably after this entry was logged the writer returned to the reactor operating floor to assist in the completion of these tasks.

The stage of reassembly of the various mechanism components and cover plates immediately before the accident has been estimated considering their condition and location after the accident as seen from photographs (Figs. 50, 51, 52 and 54) and as described by witnesses who participated in the recovery of the three crew members. The following is a comparison of estimated conditions in the vicinity of the reactor head immediately before and after the accident:

Head Opening No. 1

Before: The control rod drive mechanism housing had not been assembled. The shield plug assembly was in place. It is not known whether the stop washer and nut were in place.

After: The shield plug for this mechanism is no longer in the head. This shield plug may be the one that is presently stuck in the bottom of the fan floor in an approximately vertical projection from between openings No. 7 and No. 9. The rack for this control rod is protruding from the vessel head about 16½ inches. This would indicate that the cadmium in the rods extends 2 inches below the fuel. Due to the fact that the connecting rod is bent, it may be something less than this, however, it is believed that the cadmium section still fully covers the fuel. Approximately two-thirds of the threaded section at the top of the rack appears to be broken off. This indicates that the nut and stop washer may have been assembled.

Head Opening No. 2

Before: This port had been opened to insert flux wire assemblies and was closed and bolted down.

After: The cover plate with shield plug is still bolted in place.

Head Opening No. 3

Before: The control rod mechanism housing had not been installed. Although the shield plug assembly was in place, it is believed that the stop washer and nut were not in place.

After: The shield plug for this mechanism is not in place. It is estimated that the shield plug struck a fan floor wide flange beam in almost vertical projection above this opening. The location of the shield plug is presently unknown. The control rod rack is broken off immediately flush with the top of the nozzle flange. The rack could have been broken during the ejection of the shield plug or by another shield plug striking it in falling back across the head. The control rod blade is in the core and apparently fully, or almost fully, inserted.

Head Opening No. 4

Before: The control rod housing was not installed. The shield plug assembly was in place and the stop washer and nut were assembled. The shield plug used was a spare one, since the original plug was damaged in its removal during the previous shift. This spare shield plug had been used before for a short time during the early operation by ANL.

After: The shield plug for this mechanism is no longer in place. It is estimated that the shield plug struck a fan floor I-beam in almost vertical projection above this opening. The stop washer and nut may have been in place since the rack and extension shaft are no longer in view. The extension shaft-dummy control rod blade connection may have broken off when the blade struck the bottom of the vessel head. This blade may be lying across the top of the core.

The shield plug ricocheted off the beam and its present location is unknown.

Head Opening No. 5

Before: This control rod drive mechanism had not been disassembled during the shutdown period.

After: The mechanism appears to be intact, and thus the control rod is probably in the core.

Head Opening No. 6

Before: This port was closed and the cover plate with its shield plug bolted down.

After: The cover plate is still bolted down.

Head Opening No. 7

Before: The control rod drive mechanism housing was not in place. The shield plug assembly was in place. It is not known whether the stop washer and nut were installed.

After: The shield plug for this mechanism is no longer in place. It is believed that this shield plug impaled crew member number 3 and stuck into the bottom of the fan floor between two I-beams in an approximately vertical projection over head opening No. 6. The rack for this control rod is protruding from the vessel head about 17 inches. Based on this, the control rod is apparently fully in the core with the cadmium extending about $1\frac{1}{2}$ inches below the fuel.

Head Opening No. 8

Before: The cover for this opening contained the thermocouple leads from the instrumented fuel elements. It is believed that this cover had been bolted in place.

After: The cover plate is no longer in place and the threads on the holddown studs are stripped. One of the studs appears to be bent. The present location of the cover plate is not known although it is believed to have struck a wide flange beam overhead and ricocheted behind equipment on the operating floor. No evidence of the thermocouple leads has been seen.

Head Opening No. 9

Before: The control rod drive mechanism housing was not installed. It is believed that the shield plug was in place and that a control rod lifting tool had been attached to the control rod rack in preparation for the assembly of the stop washer and nut.

After: The shield plug for this mechanism is no longer in place. It is estimated that this shield plug struck a fan floor I-beam and fell back across the vessel head. The rod withdrawal tool was probably in place since the rack is no longer in view. It is felt that the connector shaft extension was broken off, and that the blade is probably resting on top of the core since the connector shaft is protruding into the nozzle opening.

The two water level indicators are still in place, however, it is felt that these indicators have been damaged by the incident and that the readings from these indicators can no longer be relied upon.

The 1/4" thick plate, which covers the dry mixture shielding material surrounding the nozzles on top of the head and which is intermittently tack welded at its outer edge to a shell section, was broken loose and flared up in the vicinity of ports 1, 2 and 8. The dry mixture is scattered over the reactor operating room with a predominant amount between ports 1, 2 and 8 and the adjacent shield blocks. Some pieces are seen on the fan floor I-beam flanges and on head opening flanges. Some of the shielding material may have dropped into the reactor.

Based on the medical evaluation of the men's injuries and the foregoing analysis, a reconstruction of the incident has been surmised. Figure 55 illustrates the probable positions and locations of the crew members immediately before the accident. Crew member #1 was standing in a space between two of the shield blocks which had been pulled away from the head. He may have been bringing tools or parts for use in the assembly of the rod drive mechanisms. Crew member #2 was standing near the outer diameter of the head in the vicinity of the instrument wells. He probably had his back toward the reactor and may have been waiting for crew member #1 to hand him some parts or tools. Crew member #3 was standing on the vessel head straddling the rod drive mechanism shield plug assembly No. 7 or was in a crouched position with his hands on the tool used to withdraw the rack for rod No. 9. The withdrawal of the rack enables attachment of a "C" clamp which holds the rack in a partially withdrawn position so that the scram stop washer, nut and cotter pins could be installed.

Presumably, crew member #3 inadvertently withdrew the No. 9 rack (and control rod) further than instructed which resulted in a nuclear excursion. The resultant sudden increase in reactor pressure forced all the shield plugs out of the vessel head with the exception of No. 5 which was a fully assembled and bolted down mechanism. At the same time rods 4 and 9 were completely withdrawn. Rod No. 4 was a dummy aluminum rod and did not contain any poison material. The No. 8 cover plate was also ejected, stripping the threads on its holddown studs. The five ports opened; water, steam and core material were ejected outside of the reactor vessel.

The #2 crew member was struck on his back and legs with water and/or steam causing him to be thrown against a shield block and landing in the vicinity of the instrument wells. The #1 crew member was also struck with water and/or steam and was thrown back against another shield block striking his head first. Simultaneously, the No. 7 shield plug assembly impaled the #3 crew member and pinned him to the bottom of the fan floor a distance of approximately 13 feet above the reactor head. Figure 56 illustrates the location of the three crew members after the accident as seen by the recovery team.

E. SUMMARY OF ACTIONS IMMEDIATELY FOLLOWING THE INCIDENT

After discovery of the accident the AEC-Idaho Operations Office Emergency Plan went into effect. The AEC report of events following the accident and during the emergency period contains important information. This release [27] covering the emergency period is included on the following pages.

"SEQUENCE OF EVENTS
RELATED TO THE SL-1 ACCIDENT
AT THE NATIONAL REACTOR TESTING STATION, IDAHO, ON
JANUARY 3, 1961

"First indication of trouble at the SL-1 (Stationary Low Power No. 1) reactor was an automatic alarm received at Atomic Energy Commission Fire Stations and Security Headquarters at 9:01 p.m. (MST) January 3, 1961. The alarm was immediately broadcast over all NRTS radio networks. At the same time, the personnel radiation monitor at the Gas Cooled Reactor Experiment gate house, about one mile distant, alarmed and remained erratic for several minutes.

"Upon the receipt of the alarm, which could have resulted from either excessive temperature or a pressure surge in the region above the reactor floor, the Central Facilities AEC Fire Department and AEC Security Forces responded. A Phillips Petroleum Company, operating contractor for some NRTS facilities, health physicist from the Materials Testing Reactor area was called at this time.

"The fire engines and security forces arrived at the SL-1 site, about eight miles from the central facilities area, at approximately 9:10 p.m. Security patrolmen opened the gates in the site area fence and later the south door of the SL-1 administration Building. Firemen equipped with Scott Air Paks and radiation survey meters went through the administration building and the support facilities building in search of the operators and evidence of fire.

"The initial penetration went as far as the entrance to the reactor building; however, unusually high radiation levels there caused the search party to withdraw pending health physics guidance. No fire or smoke nor any personnel were seen in the support facilities or administration building. The searchers did not enter the reactor building proper.

"At 9:17 p.m. the Phillips health physicist arrived at the SL-1 site. He and a fireman, wearing Scott Air Paks, made another trip through the administration and support facilities buildings and as far as the foot of the stairs to the operating floor of the reactor building, where they encountered a radiation level of 25 roentgens per hour, the limit of the survey meter they were using. They retreated from the reactor building and thoroughly searched the administration and support facilities buildings looking for the three men believed to be on duty. They saw no one, nor any smoke or fire. During this search they encountered radiation fields of from 500 mr per hour to 10 R per hour.

"By this time a radio check to other NRTS installations confirmed that the three SL-1 operators had not gone to any of them, so it was now presumed they must be in the reactor building.

"At 9:35 p.m. two more Phillips health physicists arrived, already in protective clothing. One of them, with two firemen and with a 500 R per hour range survey meter, went up the stairs of the reactor building until a 200 R per hour radiation field was encountered. This group withdrew from the building to plan a course of action based on radiation levels noted. Then, with AEC approval, the other Phillips health physicist and an AEC fireman went to the top of the stairs and took a brief look at the reactor floor. Observed radiation levels of the order of 500 R per hour forced their quick withdrawal. They saw some evidence of damage but no bodies.

"By 9:36 p.m. key personnel of AEC-Idaho Operations Office, Combustion Engineering, Inc. (operating contractor for SL-1), and Phillips Petroleum Company had been notified of the SL-1 accident. Following notification, many personnel who played key roles in the rescue efforts at SL-1 had to travel from Idaho Falls to the SL-1 Site, a distance of 41 miles. At 10:25 p.m. IDO designation of a Class I Disaster was broadcast over the NRTS radio network.

"When four Combustion Engineering personnel, including the SL-1 Plant Health Physicist, arrived, they decided to enter the 500 R per hour field. The four Combustion Engineering men, having verified that the three military men on duty had not left the site, prepared to enter onto the reactor operating floor.

"At approximately 10:35 p.m. the Combustion Engineering supervisors for plant operations and health physics, wearing Scott Air Paks and carrying two 500 roentgen scale Jordan Radectors, entered the reactor operating floor for less than two minutes. They saw two men; one moving. They withdrew and returned with two more Combustion Engineering men and an AEC health physicist.

"Two of the group picked up the man who was alive and put him on a stretcher at the head of the stairs. The other three of the group observed that the second man was apparently dead. The group got the stretcher down the stairs and out the west door within three minutes of entry, and put the stretcher in a panel truck. The man was taken in the panel truck to meet the ambulance, transferred, and taken to the junction of Highway 20 and Fillmore Blvd., where the AEC doctor was met. When the doctor examined the casualty at 11:14 p.m. he pronounced him dead and the ambulance returned with the body to the SL-1 site pending a decision on the temporary disposition of the body.

"At about 10:38 p.m. another group, made up of two military and two Phillips personnel, entered onto the reactor floor briefly to locate the third man. They located him and determined that he was dead and did not attempt to remove him.

"The recovery group went to the GCRE for preliminary decontamination. Gamma Exposures of the five-man group ranged from 23 to 27 roentgens. As the groups were returning from the GCRE, they stopped long enough to permit one military man and one AEC health physicist to go through the support facilities building and close doors to lessen the chance of a fire starting and spreading in the disaster area; the two men did not enter the reactor building on this trip. When the two men returned to the rest of the group, it proceeded on to the decontamination trailer set up at Fillmore Blvd. and Route No. 20. From here the group split up with part going to the Central Facilities Dispensary and the rest going to the Chemical Processing Plant for further decontamination.

"Having concluded that the remaining two operators were dead, the AEC-IDO health physicist suspended rescue efforts and ordered all personnel back to the roadblock established on Fillmore Blvd. at Highway 20.

"After the ambulance had been returned to SL-1 to await a decision on disposition of the body, personnel involved in the transfer of the body from the panel truck to the ambulance went to the Central Facilities Dispensary for decontamination. Between midnight and 3 a.m. on January 4 approximately 30 people who had been engaged in the emergency at the SL-1 area were admitted to the dispensary for secondary decontamination. These personnel included firemen, security patrolmen, and military personnel. Preliminary badge readings and urine sample analyses for these 30 people were received around 3:30 a.m. and indicated that all personnel could be released. To assist in the above-mentioned decontamination processes, four Phillips Petroleum Company health physicists came to the dispensary from the MTR and Engineering Test Reactor.

"At approximately 6 a.m. on the morning of January 4, a team of five men removed the body from the ambulance located in the SL-1 area. The body was disrobed in order to remove as much contamination as possible at the site. The body was replaced in the ambulance, covered with lead aprons for shielding purposes, and transported to the Chemical Processing Plant where surface decontamination was attempted. Individuals involved in the disrobing and transfer process received a maximum exposure of 770 millirems gamma. Prior to decontamination the reading from the first body was approximately 400 R per hour at the head region, approximately 100 R per hour at the feet, and from 200 to 300 R per hour over the remainder of the body. First efforts to decontaminate the body resulted in no significant decrease in the readings.

"Between 7 a.m. and 11 p.m. on January 4, the day following the incident, several entries into the reactor buildings were made. As a result of the entries, the second body was recovered, leaving one fatality to be recovered. Detailed events involved with removal of the second body are presented in a subsequent paragraph. A Hurst criticality dosimeter was recovered from just outside the door leading onto the reactor operating floor. Personnel history files were recovered from the Administrative Support Building. In addition, the reactor operating log book and all but one of the plant instrument charts were recovered from the Control Room Area. The instrument charts recovered are the following:

Condenser Air Temperature Inlet	Purification Water Temperature*
Condenser Air Temperature Outlet	Reactor Pressure
By-pass Steam Flow	Linear Power Level
Main Steam Flow	Log Power Level
Reactor Water Level	Feedwater Flow

"The linear power level and feedwater flow instruments are known to have been off at the time the charts were removed. The only chart not recovered was the Constant Air Monitor.

"During this same period investigation teams were organized by the AEC, Argonne National Laboratory and Combustion Engineering, Inc. Efforts continued on planning removal of the last victim, and assessing the damage incurred.

*This is the Feedwater Temperature Chart. Purification water temperature is not recorded.

"In addition to the normal continuous radiation monitoring stations which were operating at the time of the accident, radiological monitoring teams started intensive surveys of the adjacent areas and NRTS environs to evaluate any possible radiological hazard. These surveys are continuing. No radiological hazard to the public has been indicated.

"At approximately 4 p.m., January 4, 1961, preparations began to recover the second body from the reactor operating floor. The body was located in an area where radiation levels were estimated to be approximately 750 R per hour.

"A recovery team consisting of six military personnel and two AEC health physicists proceeded from the decontamination check point on Fillmore Blvd. near U. S. Highway 20, after having been extensively briefed, rehearsed, and attired in protective clothing, to the entrance of the SL-1 compound at about 7:30 p.m. Of this group, two military men and two health physicists entered the Support Facilities Building through the side entrance into the maintenance workshop area. A blanket was placed on the floor in the control room.

"Because of the high radiation levels to be encountered, the maximum permissible working time on the reactor operating floor was limited to one minute. One health physicist was assigned to hold a stop watch and time the actual entrance to the reactor operating floor, signaling the two-man recovery team when their time was up. The other health physicist remained in the support facilities building to check the body for radiation after its removal from the reactor building.

"Having been briefed as to the location of the body to be recovered, the two-man team entered the reactor operating floor and proceeded directly to the body. One man picked up the victim's legs while the other grasped the body around the shoulders and they moved rapidly out of the high radiation area and down the stairway. Their one minute limit in the reactor area did not expire until they were part way down the stairway. The two men continued down the stairs and placed the body on the blanket in the control room.

"The second two-man team entered the Support Facilities Building and went to the control room where they picked up the body by the four corners of the blanket and carried it out of the SL-1 compound. The work clothing or coveralls was removed from the body, which was then placed in an ambulance standing by for the purpose at 8:08 p.m. The ambulance proceeded with the body to the Chemical Processing Plant where facilities had been prepared to receive it. The third two-man military team proceeded into the Support Facilities Building and on to the reactor operating floor for the purpose of attempting to gain some more information about the status of the remaining body and the reactor.

"The short periods of time that these recovery teams were in the high radiation areas on the reactor operating floor resulted in gamma exposures of from 1 rem (roentgen equivalent man) to about 13 rems.

"On Thursday evening, January 5, an official photographer entered the radioactive reactor compartment to photograph the scene of the explosion. Radiation fields greater than 500 R per hour were reported by the accompanying health physicist. The photographer, wearing protective clothing and breathing apparatus, was allowed 30 seconds to complete his assignment. By entering the reactor compartment only long enough to trigger his camera and withdrawing to a less radioactive area to change film and make adjustments, the photographer was able to obtain the interior photograph needed. This photograph assisted AEC investigating teams in making plans to recover the third body and in evaluating damage to the reactor operating area. Maximum radiation exposure of these two men was less than two roentgens gamma of radiation.

"The third body had been observed to be lodged in the ceiling above the reactor. Because of the high radiation fields (above 500 R per hour) personnel could not climb onto a beam to free the body which itself was highly contaminated with radioactive material.

"The plan for removal of this third body was to position a large net (5' x 20') under it and attempt to lower the body onto the net. The net itself was fastened to the end of a crane boom. The large doors on the reactor building that are used for moving equipment in and out of the building were opened to permit the crane to position the net just below the body. A closed circuit TV camera had been placed in the reactor building to help position the net.

"When the net was in position, teams of two men each were to move in quickly and try to lower the body onto the net. Because of the radiation fields, each team had less than a minute to make their attempt at freeing the body.

"Due to a malfunction of the television equipment, it was necessary to use the first team of men to check that the net was properly positioned; they accomplished their mission in less than their allotted time.

"Four additional teams were used to accomplish the mission of freeing the body and lowering it onto the net. A sixth crew, outside the building was used to move the crane which held the net. The third body was removed from the building at 2:37 a.m. on January 9, 1961. The estimated doses received by the men entering the reactor building to free the body ranged from 2.5 to 7.5 rem.

"Recovery operations were completed at 4:42 a.m. January 9, 1961.

"Official photographers have made a permanent record of activities at the SL-1 area. Aerial photographs were taken Friday, January 6, 1961, to record the condition of the reactor building exterior, which appears undamaged.

"At 1:45 a.m. Sunday, January 8, 1961, a photographer, accompanied by a health physicist, photographed the reactor compartment. The photograph was requested by the Technical Advisory Committee which is assisting the Idaho Operations Office in planning the recovery of the third victim. A photograph of the control room was also taken. Readings of the high range gamma dosimeters worn by the men showed a maximum exposure of less than three roentgens.

"Entry to the reactor building continued to be a hazardous undertaking. To protect individuals from contamination, a detailed procedure is observed prior to entry. A detailed plan of action for each operation is established in order to obtain maximum benefit from the limited observation time of one to two minutes. AEC and Combustion Engineering health physicists personnel control the disaster field operations to ensure maximum safety for all participants. They determine who may enter, the radiation exposures to be tolerated, and the equipment to be utilized.

"The person assigned an entry mission and a health physicist are each dressed in two pairs of coveralls, shoe covers, and gloves. Around the wrists and ankles, tape is used to insure no skin remains exposed. Caps and respiratory protection equipment plus miscellaneous radiation detection equipment complete the outfitting of participants. Following exit from the contaminated area, clothing is removed and participants are decontaminated, if necessary, by scrubbing with soap and water.

"Since radiation effects are cumulative, each entry by an individual brings him closer to prescribed maximum permissible limits. Exposures to personnel are kept as low as possible by strict time limitations and careful planning. To prevent multiple high exposures to individuals the missions are assigned to different personnel, thereby requiring a larger number of persons.

"There have been 23 persons who have received radiation exposures during activities at the SL-1 site varying from three roentgens to 27 roentgens total body exposure. Of the total, 14 received exposures of three to twelve roentgens, six were in the 12 to 25 R range, and three above 25 R. Precautionary medical checkups did not disclose any clinical symptoms."

IV. INFORMATION OBTAINED FOR EVALUATION OF ACCIDENT

A. PHYSICAL CONDITION OF THE REACTOR AFTER THE INCIDENT

The primary evidence concerning the physical condition of the reactor and core after the incident of January 3, 1961 is contained in a series of photographs which have been obtained in a variety of ways. Shortly after the incident the entrance of a photographer onto the reactor floor was permitted so that a few survey photographs could be taken of the area surrounding the reactor head. These photographs which are reproduced as Figures 50, 51 and 52 show the distribution of debris in the immediate neighborhood of the vessel head, the damage to the top shield, the position of the disassembled bell housing and the control rod drive mechanism parts imbedded in the ceiling over the reactor. It was apparent from these pictures that the physical damage was highly localized in the neighborhood of the vessel head itself and the appearance of several control rod racks protruding from head nozzles indicate that at least some of the control rods might still be in the reactor. The No. 5 control rod drive mechanism which had not been disassembled prior to the incident appeared to be intact.

Further observations of the reactor were made by use of remotely controlled equipment in order to minimize the exposure of personnel to the high levels of radiation encountered in the building. A method of entry was devised through the freight doors opening onto the reactor operating floor. An Austin-Western hydraulic crane was equipped with a suitable extension boom for the insertion of lights and cameras into pre-determined positions over the reactor head. The cab of the crane was shielded with lead to reduce the exposure of the operator. A method of guiding the crane by remote operation into the proper position was developed. The crane is shown in Figure 53 in a typical entry operation.

In addition to the series of visual observations made possible by the use of this equipment, a number of physical measurements have been taken to indicate the water content of the reactor vessel, the gamma dose, neutron flux and temperatures encountered above and within the vessel.

Four entries have been made into the reactor with remotely controlled equipment to make visual inspections of the reactor in some detail. The purpose of these observations was not principally to shed light on the nature of the accident but rather to provide information of value in appraising the present safety of the reactor and for carrying out a deactivation plan. In the first remotely controlled entrance made on January 26, 1961, motion pictures were taken of the reactor vessel head (Fig. 54) to provide a basis for planning subsequent observations within the vessel; prior to these observations it was not known whether any of the head nozzles were sufficiently clear to permit the insertion of lights and cameras.

A first attempt to view the interior of the reactor with a TV camera was not successful. The equipment was not rugged enough at that time to meet the rough outdoor conditions which were further complicated by a snowstorm that added moisture problems. An entry was then made on February 22, 1961 in which a light was dropped into the reactor vessel and motion pictures were taken from over the open ports in the reactor vessel head. Although the quality of these pictures was fairly good, the small openings could not provide as complete a view as desired of what was found to be a decidedly chaotic condition in the reactor vessel. Consequently, two further entries were made on March 15-16, 1961 in which both a light and a ruggedized and maneuverable TV camera were lowered into the vessel. From various parts of these four motion picture films, the apparent condition of the core and its control system have been inferred.

The pictures obtained, particularly those from the latest TV camera entry, indicate the nature and disposition of debris on top of the reactor core. It should be emphasized, however, that the conclusions discussed below concerning the present condition of the reactor and its control system are, of necessity, somewhat speculative because little of the core itself is actually visible.

The most recent entry, April 15, with a shielded and remotely operated miniature camera provided a single picture of the core directly below port No. 8. This first picture (Fig. 61) with a still camera has greater resolution then previous pictures, thus, for the first time clearly defined fuel plates and core structure are seen. Further photographs are planned to obtain more extensive coverage of the core.

The overlays in Figure 8, particularly the phantom head, were prepared for comparison with these pictures. The phantom head has six clear ports corresponding to those presently open in the SL-1 head.

1. Summary of Observations

All the photographs obtained show extensive damage to the control rod shrouding and the visible part of the reactor core. The damage is not so great however, as to preclude making tentative identifications of some of the core components. On the basis of these photographic identifications, and the measurements mentioned before, some general statements about the state of the reactor can be made.

a. There is no evidence that the four peripheral control rods are withdrawn from their shrouds. The lower ends of the four peripheral control rod extensions appear to be near the appropriate control rod shroud.

b. A control rod, presumably the No. 4 dummy rod appears to be almost vertical above the core with one end briefly seen in (Fig. 58).

c. The core perimeter has expanded radially into the downcomer region, thus reducing the downcomer width considerably. This normally empty region between the core and thermal shields also contains debris from the core including loose fuel assembly box tops. Several of the cruciform control rod shrouds (Nos. 1 and 7) above the core have been deformed into an H shape and moved out radially from their original location.

d. Pieces of control rod shroud which were originally in the active core region are seen to be lying on top of the core along with what is apparently control rod No. 9.

e. From the negative results of the attempts to detect water in the reactor vessel it is clear that there is no water above the core. The slim probe introduced later actually penetrated through the core twice and apparently to the bottom of the reactor vessel with no indication of water, thus it appears that there is little, or no, water in the reactor vessel.

f. The insertion of the thermocouples onto the top of the core indicated a temperature at the debris lying on the core of 98°F.

A detailed reconstruction and identification of core components based on these photographs has been made by the U.S. Naval Photographic Interpretation Center; their report is reproduced verbatim in the following section.

2. Observation within the Reactor Vessel

Following photography of the vessel head an entry was made on February 22, 1961 to obtain a similar set of pictures from over the head with a light source lowered into the vessel through opening No. 8. A scanning pattern was developed for remote operation so that the motion picture camera would traverse head openings Nos. 8, 1, 2, 3, 9 and 7. Figure 57 shows selected frames from these photographs. During the scanning process the camera may have touched the rack protruding from opening No. 7, changing its position from roughly 12 o'clock to 3 o'clock.

To provide a view of a larger area of the core, entries were made on March 15 and 16 to insert a fixed focus, maneuverable TV camera and light source through head opening No. 8. Motion and still pictures taken of the TV screen provide significantly more information than had previously been obtained from the pictures taken above the head. In several frames from these motion pictures, large pieces of shrouding are visible which, judging by the circular holes and rivet holes apparently came from the fuel bearing region of the core (Fig. 5). A view of a control rod seen in these pictures is probably dummy rod No. 4 based on its apparent size. These pictures cannot be reproduced satisfactorily due to the loss of detail inherent in TV and subsequent photography of the TV screen. However, on May 11, 1961 a series of pictures were taken with a shielded miniature camera (Fig. 62) covering the same area of the core. These show considerable detail as shown in Figure 58.

Although many people have interpreted the pictures obtained, the most authoritative reconstruction of the photographic evidence of the situation in the vessel was made by G. Green of the U. S. Naval Photographic Interpretation Center in report N-PZ11, entitled "Detailed PI Study of the SL-1 Reactor Core and Vessel Damage, National Reactor Testing Station, Idaho." The text of his report follows, with his results as shown in the overlay (Fig. 59).

"<u>Position No. 1</u> - The shrouds have been folded flat and crushed against the thermal shield. The control rod appears to be in a completely down position. (The Fig. 9 appearing on the rod in the March 16-17 photos is an illusion made by the connection between the ball section and the upper, narrow end of the control rod.) The upper spray ring obscures a portion of the No. 1 area.

"Position No. 2 - Area is partially obscured by control rod No. 9. Several fuel elements are identifiable.

"Position No. 3 - The shrouds have been bent and twisted and moved toward the thermal shield. The control rod is in a down position and has moved about 8" toward the downcomer. A gaping hole remains at the former position of the control rod and the shroud. A probable cross-stanchion lies across the No. 3 area tilted at a 45° angle from the vessel wall, downward toward No. 9 position.

"Position No. 4 - The shroud, part of which is visible, has been smashed against the pipes at the vessel wall. The upper end of the 1-1/4" filler pipes to the lower spray ring has been ripped loose and twisted toward No. 3. Most of the area lies in the shadow of control rod No. 9.

"Position No. 5 - The rod extension appears to be in a full down position. Part of control rod No. 9 is crushed against it and obscures the shroud. It has been moved toward the downcomer, but how much is not determinable.

"Position No. 6 - Part of the shroud is visible. Most of area is hidden by control rods Nos. 9, 7 rod extension, and the upper and lower spray rings.

"Position No. 7 - The control rod is in the down position. The rod and shroud have been twisted and displaced toward the vessel wall about 6-8 inches. The rod extension and the rack have been bent or broken at the union joint. A probable fuel box top lies between the shroud and the vessel wall.

"Position No. 8 - The shroud has been twisted and warped and at the level of the fuel elements has been pushed against the thermal shield. Debris is wedged between the shroud and the wall, including a fuel box top. The hold-downs at No. 8 have been badly twisted. Six of the eight fuel boxes and spares have been identified. An unidentified item between Nos. 1 and 8 (annotation N) may be a spare box.

"Position No. 9 - The area No. 9 has been blocked from view by the No. 9 control rod blades, which appear to be lying almost horizontally from shrouds 1 and 2 to 6 and 7. The outer covering of the control rod blades have been torn, twisted and peeled from the center plate in sections of the rod. A section of shroud, possibly from No. 9, lies near No. 3, and another possible section of shroud lies in the No. 1 area.

"Other Comments: Little or no downcomer region remains according to the photographs.

"A badly twisted possible cross-stanchion appears to be lying across fuel element boxes between No. 8 and No. 9 positions.

"The lower spray ring has been ripped from the vessel wall and from No. 6 toward Nos. 7 and 8 has been twisted upward around the vessel wall. At a point above No. 8, it is approximately 6 feet above the core surface and there are two fuel box top sections resting on it. (See also annotation B on mosaic overlay, Fig. 59.)

"The upper spray ring has been torn loose at several points and has pulled away from the wall between No. 1 and No. 7, such that it passes above the core between No. 8 and No. 9.

"A total of 19 fuel element boxes in the core have been identified. Two others are possibles. These are in addition to the 4 top sections already identified."

3. Water Detection Attempts and Temperature Measurements

Since the presence of water in the SL-1 reactor vessel seriously affects the next stages of the recovery operation, two schemes have been devised and carried out for the detection of water.

In the first, an ultrasonic vibration probe was lowered on a cable into the reactor vessel to a depth of 11 feet 4 inches below the top of the No. 8 nozzle flange. The top of the active core is 11 feet 10 inches below the top of the No. 8 nozzle flange. The external dimensions of the probe approximate a cylinder two inches in length and 1/2 inch in diameter. During this attempt at water detection, no evidence of water was found.

In the second attempt to detect the presence of water, a long slender probe (1/4 in O.D.) was lowered through the No. 8 nozzle to a point at which the lower end of the probe should have reached roughly 15 feet 6 inches below the top of the flange. This probe was constructed in a number of separate sections (Fig. 60) one inch long each of which contained a cellulose fiber and a water soluble chemical (potassium permanganate) that dissolves rapidly and colors the fiber. The sides of the tube were pierced with small holes to permit entrance of the water into the individual sections. Those sections when immersed in water, color and give obvious indication in 2 minutes. No evidence of water was found with this probe. The indication that the probe penetrated a vertical hole through the reactor was obtained entirely from the apparent inertia of the probe as felt by the remote operator. A second probe entry penetrated 16 feet 3 inches, and pictures (Fig. 61) of the probe emerging from control rod channel No. 8 were taken with a shielded miniature camera (Fig. 62). On this entry the probe went through No. 8 channel and apparently to the bottom of the reactor vessel as inferred from the reactor dimensions shown in Figure 63.

One series of measurements have been made of the temperature distribution with a thermocouple probe. Above the reactor head the air temperature was reported as 47°F. After lowering the thermocouple into the reactor vessel to a depth of 7 feet above the core, the temperature was observed to rise to 90°F and when the thermocouple was subsequently lowered until contact was made presumably with debris at the top of the core, the temperature rose to 98°F.

B. SAMPLES OBTAINED FOR ANALYSES

The analysis of samples obtained from the SL-1 may be divided into several categories. Basically, information has been retrieved from the area in the form of activated metallic parts which formed part of the facility itself, activated items worn by the persons involved in the incident and soil samples. In all cases, these were analyzed to provide information indicating the nature and extent of the excursion.

The questions to be answered are:

a. Was the incident primarily a nuclear excursion?

b. Were other events such as a metal-water reaction involved?

c. Were fission products released and if so to what extent?

d. What was the total energy release?

Some of these questions remain unanswered at the present time. Only partial information is available in some cases since many of the samples removed from the SL-1 following the incident did not yield useful information. Results of analyses of all samples are included to provide a complete record.

1. Activated Material Removed from the SL-1

The results of the analysis of items removed in the form of activated material from the reactor and activated material on the persons involved in the incident are summarized in Table IX. The items which yielded information of some value are discussed in more detail below.

A wrist watch strap buckle taken from the second victim and a brass screw holding the flint in a Zippo cigarette lighter in the clothing of the first victim were analyzed for copper-64. The measurements made on these items are summarized in Table X. The buckle and screw were divided into two pieces each. One half of the buckle was counted, as is, after external decontamination. The other half underwent a copper sulfide separation. The neutron dose calculated from this data was 1.8×10^{10} nvt with no separation and 2.1×10^{10} nvt with separation. The neutron dose calculated from the head of the lighter screw was 9.3×10^{9} nvt. This information has been checked by the Chemical Processing Plant at NRTS and Combustion Engineering and has been found to be accurate within the limits of experimental error.

TABLE X

COPPER-64 DATA

(IDO HEALTH AND SAFETY DIVISION REPORT)

Sample Number	Date Time	Photo-peak Count	Wt. of Sample	% Cu	Count Time	d/m at 2100 1/3/61	Neutron Dose	Description of Sample
1	Jan. 5 0100	3059	0.812g	76	4 min.	2.37×10^{5}	1.8×10^{10}	1/2 Watch Band Buckle - No Chemical Separation
2	Jan. 5 0150	1639	0.406g	76	4 min.	1.62×10^{5}	2.1×10^{10}	1/2 Watch Band Buckle - CuS Separation
3	Jan. 4 1900	1066	0.366g	59.3	4 min.	5.42×10^{4}	9.3×10^{9}	Head of Screw Holding Lighter Flint

TABLE IX

SAMPLES TAKEN FROM SL-1 FOLLOWING ACCIDENT

Sample Description	Time of Analysis Date	Hour	Analyzed for	General Statement	Identification By	Data d/m	Remarks	Total Neutron Dose (nvt)
1. Cigarette lighter screw taken from first body recovered	1/4/61	1900	Copper 64	Copper 64 found	Gamma spectra	5.42×10^4		9.3×10^9 (thermal)
2. Brass pin from film badge case recovered from second body	1/5/61	0300	Copper 64	Copper 64 found	Gamma spectra		Insufficient activity for analysis	
3. Brass watch band buckle from second body	1/5/61	0100	Copper 64	Copper 64 found	Gamma spectra Copper chemistry Decay curve	2.37×10^{15}	Gross count on 1/2 buckle	1.8×10^{10} (thermal)
							Chemical separation other 1/2 buckle	2.1×10^{10} (thermal)
4. Copper wire and screws from control room telephone	1/7/61		Copper 64	None found				
5. NAD instrument taken from SL-1 (No. 270) position at top of access stairway	1/4/61	1100						
a. Bare gold foil	1/4/61	1100	Gold 198	Gold 198 found	Gamma spectra decay curve	2.2×10^3		0.6×10^8 (thermal)
b. Cadmium covered gold foil	1/4/61	1100	Gold 198	Gold 198 found	Gamma spectra	1.5×10^3		1×10^{11} (fast)
c. Sulfur pellet approx. 20 grams	1/12/61	1530	Phosphorus 32	Contaminated: Phosphorus separation made			Insufficient activity for analysis	
d. U-238, Pu-239, Np-237 fission foils	1/4/61	1600		No activity above background at time of counting				
6. Gold ring taken from third body recovered	1/10/61	1800	Gold 198	Gold 198 found	Gamma spectra	1.9×10^4	0.472 grams of ring 0.066 inch thick, 0.194 inch wide, 0.308 inch long	9×10^9 (thermal)
7. Zipper pull and button from clothing of first body recovered	1/4/61	1200	Copper 64	None identified: highly contaminated with aged fission products				
8. Flexatallic gasket from SL-1 reactor	1/19/61	1200	Cobalt 58	Cobalt 58 found	Gamma spectra Cobalt chemistry	1.1×10^3	15 grams steel (nominally 18% nickel)	2.5×10^{11} (fast)
	1/20/61	0830	Chromium 51	Chromium 51 found	Gamma spectra Chromium chemistry	2.0×10^3	15 grams steel (nominally 18% chromium, 8% nickel)	8×10^9 (thermal)
9. Samples shaken from clothing of first two bodies recovered								
a. Metallic appearing sample (25 R/hr at 1 foot)	1/6/61		Uranium		Mass spectrometer		3.4 micrograms per ml	
	1/6/61	0430	Strontium 91 On 10 ml (aliquot)	Strontium 91 identified and estimate made	Spectra on yttrium 91m milked from strontium fraction	2.5×10^4 d/m/ml ± 50% at 2100/1/3/61	1.5×10^{18} fissions	
b. Mass assay of uranium from metal from clothing of victims	Reported by CPP to IDO - Health & Safety						U-234 0.88% U-235 90.0% U-236 2.73% U-238 6.39%	
c. Rock and gravel sample (20 R/hr at 1 foot)	1/6/61		Uranium		Mass spectrometer		3.9 micrograms per ml	
	1/6/61		Strontium 91 on 5 ml (aliquot)	Strontium 91 identified	Spectra on yttrium 91m milked from strontium fraction		Insufficient for analysis	
10. Clothing sample from third body recovered								
a. Dissolved at CPP	1/10/61		Zirconium 97	No zirconium 97 identified				
b. Mass assay of uranium from coveralls from 3rd body	Reported by CPP to IDO - Health & Safety						U-234 1.02% U-235 90.93% U-236 2.06% U-238 5.99%	
11. Liver from first body recovered (1200 grams)	1/11/61	2330	Sodium 24	No sodium 24 identified	Gamma spectra	0.4 d/m/g		
			Sodium 23		Flame photometer		1.15 mg/g	
12. Liver from second body recovered (1570 grams)	1/11/61	2350	Sodium 24	No sodium 24 identified	Gamma spectra	0.3 d/m/g		
			Sodium 23		Flame photometer		0.95 mg/g	
13. Hair samples from all three bodies							Sent to Los Alamos for phosphorus 32 analysis	
14. 100 ml blood taken from first body	1/7/61	2200	Sodium 24	No sodium 24 identified	Gamma spectrum	5 d/m/ml		

In addition to these copper items, the Nuclear Accident Dosimeter (NAD #270) was removed from its position at the top of the stairway in the reactor operating room. A thermal neutron dose of 0.6×10^8 nvt was calculated from the activation data of the gold foil in this instrument. The data was supplied by IDO - Health and Safety Division, and calculations were performed by Combustion Engineering and Phillips Petroleum personnel at MTR. The cadmium covered to bare foil ratio is 1.46. This data is summarized in Table XI. In regard to possible previous activation of the gold foil, the saturation activation due to normal neutron levels in the operating room is 840 d/m. Activation seven days after removal of the instrument from the building for repairs was 138 d/m. The activity at 1110, January 4, 1961, 14 hours after the incident, was 2180 d/m; therefore, previous activation can be neglected. Three fission foils were also included in the NAD instrument; however, the activity on these foils was below background at the time of counting. As a result, these foils do not provide pertinent information.

TABLE XI

DATA FROM NAD #270

GOLD-198

	Wt. of Gold Foil gm	d/m at 1100 1/4/61	Neutron Dose (IDO-Health & Safety) nvt	Neutron Dose (CE & Phillips Petroleum) nvt
Cd Covered Foil	0.238	1494		
Uncovered Foil	0.238	2180		
Net		686	1.2×10^8	0.6×10^8

Estimated fast neutron dose from 1494 d/m = 1×10^{11} n/cm^2

A gold wedding ring was removed from the body of the third victim. The radiation level on the ring was 5R/hr when received. After decontamination, the level was 250 mr/hr. One quarter of this ring was dissolved and analyzed for Gold-198. A summary of the data on this sample is as follows:

Wt. of sample	= 0.472 gm
Dimensions	= .066" thick x 0.194" wide x 0.308" long
d/s	= 3.17×10^2 at 1830, on 1/10/61
d/s	= 1.88×10^3 at 2100, on 1/3/61
Neutron Dose	= 7.8×10^9 nvt (self shielding not considered)

A flexitallic gasket from No. 7 control rod thimble flange was analyzed for cobalt-58 and chromium-51. This was a new gasket installed during the assembly work prior to the incident. The analytical data from this sample is summarized in Table XII. A thermal neutron calculation on the basis of chromium-51 gave 8×10^9 nvt. A fast neutron dose of 2.5×10^{11} nvt was estimated from the cobalt-58 analysis.

TABLE XII

ACTIVATION DATA FROM FLEXITALLIC GASKET

	Cr^{51}	Co^{58}
Weight of Sample	15 gm	15 gm
Composition	18% Cr	8% Ni
Cross Section	11 barns	90 m-barns
% Abundance	4.49	67.76
Half-Life	27 day	72 day
d/m at 2100, on 1/3/61	2.0×10^3	1.1×10^3
nvt	8×10^9	2.5×10^{11}*
Reaction	$Cr^{50}(n,\gamma)Cr^{51}$	$Ni^{58}(n,p)Co^{58}$

* Fast neutron threshold = 4 Mev

Some metallic and silicious appearing materials were vacuumed out of the clothing of the first two victims removed. The sample weights and the results of the analyses for total uranium and isotopic uranium are summarized below:

Metallic - 0.16 grams - 25 R/hr at 1 foot

Sample dissolved in 35 ml of acid solution
Uranium - 3.4 μg/ml or 120 μg total

0.88 a/o U^{234}
90.0 a/o U^{235}
2.73 a/o U^{236}
6.39 a/o U^{238}

Silicous - 7.65 grams - 20 R/hr at 1 foot

Samples dissolved in approximately 35 ml of acid solution
Uranium - 3.9 μg/ml or 136 μg total

0.7 a/o U^{234}
84.6 a/o U^{235}
2.6 a/o U^{236}
12.1 a/o U^{238}

It was possible to identify strontium-91 in the metallic looking sample. Yttrium-91m milked from this strontium fraction was quantitatively analyzed by gamma spectrometry. The strontium-91 activity calculated from the yttrium-91m analysis was 2.5×10^4 d/m/ml $\pm$ 50% at 2100, January 3, 1961. A comparison of this activity with the uranium content of 3.4 micrograms/ml results in a value of 1.5×10^{18} for the number of fissions that occurred in the excursion. This number of fissions corresponds to a 50 Mw-sec energy release. The data and calculations from which this energy level is calculated are summarized below. The amount of Sr^{91} remaining from normal reactor operation has been calculated. The results of these calculations are included in the summary below.

Evaluation of Energy of Excursion from Uranium and Sr^{91} Analyses

		Uranium	Sr^{91}
a.	Concentration	3.4×10^{-6} g/ml	
b.	Activity		2.5×10^4 d/m/ml $\pm$ 50% at 2100, 1/3/61
c.	Total U^{235} in Core	1.17×10^4 g	
d.	Scale-up factor (c $\div$ a)		3.82×10^9
e.	Total Sr^{91} Activity d x b		9.55×10^{13} d/m
f.	Sr^{91} Half-life		9.7 hr.
g.	Sr^{91} Decay factor		1.19×10^{-3} min.$^{-1}$
h.	Atoms Sr^{91} (c $\div$ g)		8.04×10^{16}

	Uranium	Sr^{91}
i.	Sr^{91} Yield	5.9%
j.	Total Fissions (h ÷ i)	1.4×10^{18}
k.	Fissions/Mw sec	3.2×10^{16}
l.	Mw sec (j ÷ k)	45

Sr^{91} Residual in Core

N_{γ}/N_s = Saturation fraction = 1.0 at 3×10^5 sec or 3/5 days

N_s/N^o_{25} = Saturation level for 3 Mw operations -

$$\phi = \frac{(3)\ (3.2 \times 10^{16} \text{ fissions/Mw sec})}{N^o_{25}\ (5.8 \times 10^{-22})}$$

$$N^o_{25} = \frac{(1.3 \times 10^4 g)\ (0.90)\ (6.03 \times 10^{23})}{2.35 \times 10^2} = 3.0 \times 10^{25}$$

Therefore $\phi = 5.5 \times 10^{12}$ n/cm^2/sec, and

N_s/N^o_{25} $= 9 \times 10^{-6}$ at $\phi = 5.5 \times 10^{12}$ n/cm^2/sec

N_t/N_{γ} = Shutdown fraction = 10^{-8} for 11 day shutdown

$N_t N^o_{25}$ $= (1.0)\ (10^{-8})\ (9 \times 10^{-6}) = 9 \times 10^{-14}$ atoms Sr^{91}/atom U^{235}

$N_t\ (Sr^{91})$ $= 2.7 \times 10^{12}$ atoms

$A = N\lambda$ $= 3.2 \times 10^9$ d/m just before the excursion

Ratio of $\dfrac{Sr^{91}\ (\text{old})}{Sr^{91}\ (\text{new})} = 3.3 \times 10^{-3}\ \%$

2. Evidence of Fission Produce Release

Five soil samples were obtained on January 16, 1961 in the SL-1 area. The location of these samples is indicated in Figure 77 and the gamma scans of these samples are summarized in Figure 64. In general, the activities present follow a normal fission product spectrum. Iodine-131 and zirconium-niobium-95 constitute the primary contamination. The relative distribution of fission products varies from sample to sample. It may be noted in Samples 1, 2 and 3 that the amount of zirconium-niobium present, relative to the amount of ruthenium-cesium is considerably different. The ruthenium-cesium activity in Sample 2 is much greater than the zirconium-niobium activity, while the reverse is true in Sample 3.

Table XIII is a summary of the soil sample data. The gross activities and the strontium-90 activities are tabulated. While these results do not give an estimate of the total fission product release, they do verify the fact that there was a release of fission products from the reactor building.

TABLE XIII

SOIL SAMPLE DATA INSIDE SL-1 AREA (1-16-61)

Sample Description	Sample Size	Gross Activity γ c/m	Strontium-90 in d/m/sample* 1st milking	2nd milking
(1) Front Fence	20 grams	25,000	588 ± 25	240 ± 48
(2) Front Right Fence	20 grams	27,000	350 ± 25	100 ± 8
(3) Side Entrance S-F	20 grams	103,000	4100 ± 75	1010 ± 18
(4) Rear Training Bldg.	20 grams	6,300	713 ± 38	88 ± 8
(5) Middle Right Fence	20 grams	9,800	612 ± 25	65 ± 8

*The strontium results in the 1st milking reflect contamination from other isotopes due to the rush for data. The results from the second milking are valid figures for strontium-90.

Additional evidence for fission product release was obtained from smears taken from the reactor area and air samples collected in the vicinity of SL-1. These items are summarized in Table XIV.

C. RADIATION MEASUREMENTS AND SURVEYS

1. Inside the Reactor Building

a. Neutron and Gamma Ray Measurements

On most entries into the Reactor Building, gamma ray and/or neutron detectors have been used to attempt to measure and follow the decay of the dose levels inside the building. In each case, however, the measurements were carried out in conjunction with entries made for other, more important, purposes - recovery of bodies, viewing of the core, probing for water, etc. Because of this, ideal and precise locations and exposure times were not obtained. In particular, the data directly over the reactor

TABLE XIV

MISCELLANEOUS SAMPLES CHECKED SPECTRALLY FOR MAJOR FISSION PRODUCT IDENTIFICATION

Sample Description	Time of Analysis		Analyzed for	General Statement	Identification by	Remarks
	Date	Hour				
Smear from GCRE change room	1/4/61	2250	Gross fission products		Gamma spectra	Barium-lanthanum 140, zirconium-niobium 95, cesium 134, and uranium
Square of cloth from second body removed	1/4/61	2250	Gross fission products	Gross fission products found	Gamma spectra	
Air sample on MSA 2133 paper taken in control room of SL-1	1/5/61	2000	Gross fission products		Gamma spectra	Barium-lanthanum 140, zirconium-niobium 95, cesium 134, cesium 137, cerium 141, cerium 144, and iodine 131
450 ml of air from SL-1 half way up access stairway, collected by gas sampler	1/5/61	0530		Very low activity; no identification made		
Air sample taken outside entrance to SL-1 Administration Building on MSA 2133	1/6/61	1230	Iodine isotopes		Chemical separation and gamma spectra	Iodine 131 and 133 identified on separated iodine fraction. Unable to determine iodine 133 quantitatively

head has a large uncertainty due to the fact that the response is probably quite sensitive to the exact location of the detectors relative to the open nozzles. Even allowing for this streaming, however, the results are not completely consistent. It is believed that the recent results are more accurate than the earlier ones. Wherever possible, the data have been corrected for exposure in other fields i.e., during crane entry, etc. In several cases the correction was a significant fraction of the total and is not well known.

The data given below are representative of the measurements which have been made. The gamma ray results are believed to be accurate to somewhat better than one order of magnitude; as such they have been useful in the general planning of operations (e.g., radiation resistance required for photographic equipment). The neutron results indicated first that the reactor had shut itself down, that the Sb-Be source was still reasonably intact, and, finally, by the measurement of a low cadmium ratio, that the water level in the reactor was low and the core was possibly dry.

These results are also being used in the planning of an entry which is being specifically designed to make a reasonably accurate survey of the gamma ray and neutron levels both inside and outside the reactor vessel. The survey results, in turn, will be used to determine the location of the neutron monitoring instruments in the beam hole and above the core for the proposed poison solution filling operation. It will also give an indication, or, at least, a better basis for a calculation, of the reduction in gamma ray intensity to be expected within the building as a result of the poisoning operation.

Gamma ray measurements have been made with film badges, chemical dosimeters and high range ion chamber survey instruments.

Measurements on 1/9/61, during the recovery of the third body, gave gamma ray dose rates of 200 to 400 R/hr at a height of about 5 feet above the top of the vessel and distances of 14 feet and 6 feet respectively from the reactor centerline.

Measurements have been made on several occasions of the gamma ray dose at heights of 2 to 5 feet above the top of the reactor vessel and at unknown or averaged out radial positions and timing relative to the open nozzles. These are as follows:

Date	Distance Above Top of Head	Dose Rate - R/hr
1/25	3'	3000*
1/26	3'	1000
2/22	2'	410
3/16	4.5'	210
3/17	4.5'	350
3/29	5'	170

Measurements have also been made by placing film badges on the carriage in which the crane boom rides. These badges were spread over a 5 foot distance and positioned near the cargo door, about 18 feet from the reactor centerline. Results are as follows:

Date	Dose Rate - R/hr
2/22	67
2/28	75
3/16	65
3/17	29
3/29	30

* Results above and below a lead shield indicated that a significant part of the dose is probably coming from material on the fan floor.

Measurements have also been made by placing detectors on items lowered into the reactor vessel through nozzle #8. Results are as follows:

Date	Distance below Bottom of Head	Dose Rate - R/hr
2/22	2'	6500
3/16	4'	1100
3/17	4'	1200
3/29	8'	22,000

Successful neutron measurements were made with indium and gold foils inside the reactor vessel. Attempts with other materials and other locations gave negative results due to lack of sensitivity, low flux and/or contamination problems.

For both indium and gold, bare and cadmium covered foils gave activities which were not significantly different from each other - i.e., a cadmium ratio of unity. Because of this, with a poorly known neutron spectrum and a poorly known variation of activation cross section with neutron energy, it is impossible to convert the measured activity to a calculated neutron flux. It can be concluded, however, that the neutron source is not surrounded by a large amount of material which is thermalizing the neutrons. A measurement with an indium foil surrounded by 1/2" of polyethylene more than doubled the activation, giving further indication of the non-thermal nature of the neutron source.

b. Gamma Pinhole Camera

The gamma pinhole camera was used to locate "hot spots" in the reactor building. Three gamma pictures were taken of the SL-1 Reactor Building. The first picture was taken 70 feet from the Reactor Building in an easterly direction looking into the emergency personnel door. This picture was taken almost at ground level. The camera was placed in the trunk of a car. The light picture was exposed for eight seconds and the gamma film was exposed to a total dose of 6R. See Figure 65.

After the films were developed, the following major hot spots were located:

(1) Over the reactor head
(2) To the left center of the reactor head
(3) To the right center of the reactor head, on the fan floor (at least three spots)
(4) Above the reactor head on the fan floor
(5) To the right center of the reactor above the fan floor
(6) On the ground halfway between the Reactor Building and the pinhole camera.

The second picture was taken 70 feet from the Reactor Building in a northerly direction looking into the cargo door. This picture was taken at ground level in the same manner as the first picture. The light picture was exposed for eight seconds (shot into the sun makes definition difficult) and the gamma film was exposed to a total dose of 4R (Fig. 66). The same general hot spots were noted on this picture as were noted in Figure 65, only from a different angle.

The third picture was taken from 18 feet above the ground at the same location as the second shot. The light picture was exposed for ten seconds and the gamma film was exposed to a total dose of 1R (Fig. 67). From this picture the exact height of the hot spots was determined. There were three spots located above the reactor on the fan floor and one 7-1/2 feet above the fan floor in addition to the spot created by the flux coming out of the reactor vessel itself.

Triangulation of these (3) pictures locates the major hot spots (Fig. 68).

The X-ray film for the third shot was calibrated prior to the exposure. The relative densities of the three spots observed were:

Top	.08	(7-1/2 feet above fan floor)
Middle	.24	(on the fan floor)
Bottom	.03	(over the reactor head)

If one divides the middle density by three (the number of spots on the fan floor) it seems that the spot above the fan floor and the three spots on the fan floor may be similar in level of activity.

It is not known whether all of these spots at the fan floor level are radioactive objects or the result of scattering of gamma flux by the "I" beams directly over open ports in the reactor vessel. Any radioactive objects will contribute to the general gamma flux in and around the reactor building after the core is shielded.

2. Outside the Reactor Building

a. Radiation Surveys within the SL-1 Confines

From January 4 through January 14, 1961, the Idaho Operations Office, Health and Safety Emergency Team surveyed in and around the SL-1 buildings. On January 11, 1961, Combustion Engineering, Health Physicists measured the dose rates and smeared for contamination inside the buildings.

On January 13, 1961, Combustion, Health Physicists surveyed outside the buildings. A third radiation survey outside the buildings was conducted on January 18, 1961 prior to the resumption of recovery operations. Subsequent radiation surveys were made with each operational entry (Figs. 69 through 75). On March 6, 1961, isodose lines were drawn after a thorough survey was made in the SL-1 area (Fig. 76).

The gamma radiation surveys showed no day-to-day reduction in intensity; however, a gradual decrease was taking place. Comparing the radiation intensities obtained on March 6th with the measurements taken on January 13, 1961, a reduction of from 20 to 50 per cent can be noted, depending on the proximity to the Reactor Building. By extrapolation, the Idaho Operations Office, Health and Safety Group demonstrated that for the month of February the radiation half-life was 31 days. Using Combustion Engineering, Inc. data, a 36-day radiation half-life was calculated for the same period. This is reasonably good agreement considering instrument error.

The radiation surveys were most useful operationally in recovery planning and estimating personnel exposures. The data clearly established that the reactor was not behaving like a point source and that the gamma flux was coming from a rather large area.

The smear surveys indicated that there was general contamination throughout the Administration Building and Support Facility. The highest levels of contamination were found to be close to the Reactor Building and in areas where personnel traffic carried contamination from one location to another.

b. Soil Samples

Soil samples were collected during the radiation survey entry on January 13, 1961. These samples are designated as one through five on Figure 77. The gamma spectra for these samples is contained in Figure 64. Samples six through eighteen were collected February 17. It is obvious from these samples that fission products were discharged to the environs following the incident. As would be expected, the higher activity samples are found in close to the reactor building. Due to the tracking and redistribution of the fission products, it is difficult to say too much about the direction in which the major portion of fission products might have gone. The SL-1 building contained the fission products to a greater degree than might have been expected, although not designed specifically for this purpose.

c. Air Samples

Air samplers were located around the SL-1 fence (Fig. 78). Activity levels of 1×10^{-11} μc/cc have been measured by these samplers through the month of March. The site survey group of IDO Health and Safety positioned field air samplers at various distances from the SL-1 within a week of the incident. The high volume Staplex Sampler located at the SL-1 control point collected gross air dust activity levels of

2×10^{-10} μc/cc during the month of January. Subsequent samples collected during the month of March at the control point measured normal background activity in the order of 1×10^{-14} μc/cc. Much of the activity measured in January is believed to be due to redistribution of fission products. The Maximum Permissible Concentration for unidentified radionuclides is 1×10^{-11} μc/cc. This would indicate that at present at the perimeter of the SL-1 there is no significant hazard and that the gross fission products have pretty much settled out or decayed.

D. PERSONNEL EXPOSURE INFORMATION

At the time of the incident, the IDO Emergency Plan went into effect. From January 3 until January 15, 1961, the IDO Health and Safety Group, operating within the scope of their emergency plan, controlled the Health Physics Operation. Associated with this control, of course, was personnel monitoring. On January 15, 1961, Combustion Engineering, Inc. again assumed control of the Health and Safety aspects of the SL-1 recovery operations.

During the emergency period 25 individuals received over 3 Rads whole body dose and of these, 12 received over 10 Rads. Exclusive of these individuals, the average gamma whole body exposure received by the remaining personnel involved in the recovery through January 14, 1961, was .422 Rads.

From January 15 through March 24, 1961, for comparable periods of time, the following average whole body gamma exposure was received:

Date	Rad Average
January 15 - 25	0.100
January 26 - February 7	.360
February 8 - February 19	.112
February 20 - March 2	.135
March 3 - March 13	.106
March 14 - March 24	.112

The exposure of personnel to ionizing radiation has been closely watched and kept within the framework of the Federal Register, 10 CFR, Part 20. The maximum accumulated whole body exposure which can be received in any quarter is 2.5 Rads. The only over exposures on this basis occurred prior to January 15, 1961 during the emergency.

Analysis of the records show that pocket dosimeters read generally higher than the film badges with a gross error of ± 40%.

A urinalysis program was established for all individuals involved in the operation. Seventy-three persons received a significant exposure to the airborne fission products, e.g., 1000 d/m/sample of urine. These exposures were due to equipment (respiratory protection) failure and/or to the fact that assault masks are, at best, 95% efficient, and the airborne radioactivity levels were extremely high in the support facility and Reactor Building within the first week after the incident. There were no internal exposures to personnel after January 15, 1961 when everyone was restricted to working outside the Reactor Building.

E. CHARTS FROM RECORDING INSTRUMENTS AT SL-1 AND OTHER LOCATIONS

1. SL-1 Charts

Reactor control room instrument charts divulged little information. Most of the instruments had been turned off during the shutdown period which began on December 23, 1960. The Log Power, Linear Power, and period channels of the neutron detection equipment were on and operating at the time of the incident, the associated recorders were off, therefore, no data is available.

Only two of the recorders are meaningful to the analysis of the incident; the reactor pressure recorder and the reactor water level recorder. The reactor pressure recorder indicates a pressure which may have gone as high as 270 psi with all indications higher being inconclusive. It appeared as if the pen had been dragged across the chart when the chart was removed from the recorder since this line goes counter to the timewise rotation of the chart. The reactor water level indicator shows a change in level agreeing with the log entries of filling the reactor followed by pumping down to "on scale". At 73°F this represents a water level approximately 2 feet, 5 inches below the bottom of the reactor head. The water level in the waste storage tank has been checked and agrees reasonably well with this quantity of water pumped out of the reactor prior to the incident. Shortly thereafter the incident occurred, and the indicator shows an appreciable surge with the final indicated water level steadying out at approximately the normal operating level. If the water level instrument has not been completely destroyed, it might be interpreted that about 5 feet of water was blown out of the reactor.

The chart from the stack monitor in the SL-1 building was recovered after the incident. Apparently this instrument ceased to function shortly after the incident, and it appears to give no information other than that there was a large increase in the radiation background.

2. AREA Charts

The charts of three different hot cell instruments were obtained from the hot cell facility just south of the SL-1 plant, as follows:

Constant Air Monitor (CAM) - NMC Model BM-2

Six-Unit Area Monitor - Tracerlab Model RM-103

Stack Gas Monitor - Tracerlab Model MAP-1/MGP-1

The output of the six detecting units of the Model RM-103 and the Stack Gas Monitor are fed to a Brown 12-point recorder. This recorder has a time cycle of three minutes. With 12 points, the recorder will then print a point each fifteen seconds. The time indicated on the Brown Recorder chart was in error, plus 126.5 minutes, at the time of removal. With no known power interruption, the chart should have been in error plus 126.5 minutes at the time of the incident which was indicated between 2310 and 2312 on January 3. Corrected for the time error this places the time of the incident between 2104 and 2106 or 9:04 and 9:06 PM, January 3, 1961.

The radiation levels indicated before the incident are as follows:

Point	Location of Detector	Radiation Level
1	West wall of operating area	1.2 mr/hr
2	South hot cell filter	1
3	North hot cell filter	1.3 mr/hr
4	Decontamination room	1
5	Chem Lab west wall	1
6	Service area	1
7	Not used	
8	Not used	
9	Stack monitor	1.8 mr/hr
10	Not used	
11	Not used	
12	Not used	

The indicated radiation on points #1, #2 and #3 is probably not real but drift in the zero point.

At the time of the incident, all registering points showed a sharp rise. Point #3 is the highest with a peak at 135 mr/hr. The decay was so rapid that the actual peak cannot be distinguished.

After 45 minutes all points were essentially at an equilibrium level as follows:

Point	Level
1	2.5 mr/hr
2	3 mr/hr
3	8 mr/hr
4	2.5 mr/hr
5	1.5 mr/hr
6	2 mr/hr
9	1.8 mr/hr

There is a variation shown by the points which must be attributed to fluctuation either due to an unstable field, or the instrument. Since these instruments have not been completely checked and calibrated, it is probably the instrument.

Until the chart was removed, all points stayed fairly constant with the exception of point #9 which comes from the stack monitor. Since the connections to the stack had not been made, this instrument can be considered as a moving filter CAM. It takes air from the fan loft and not from the stack. The chart showed air activity which lasted about 1-1/2 hours starting at 1445 and ending at 1615 on January 4, 1961. The peak level was 500 c/m and approximately 3.5 times the normal background.

The Brown Recorder chart is not easily read. The data was removed from the chart as accurately as possible and plotted for reference in Figures 79 through 85. The chart shows the time from one reading to the next for each monitoring point; however, the times indicated in Figures 79 through 85 are not exact from one monitoring point to another, therefore, nothing relative from one set of points to the next can be inferred. The chart itself has not been reproduced since the mass of points indicated on the chart serves to confuse rather than clarify.

The Stack Monitor also records on its own instrument. Figure 86 is a selected portion of this chart for January 3, 1961. A full-scale deflection was also recorded at 1111 hours and small deflections were recorded at 0930 and 1337 hours. These were also registered on point #9 on the Brown recorder. The radiation at 0930 was explained by work being done in the AREA hot cell, but the traces at 1111 and 1337 are not explained.

Figure 86 shows the incident trace on the Stack Monitor. The trace shows a very rapid rise at about 2058 and a very rapid decline. It was at equilibrium again in about 20 minutes. Airborne radioactivity was shown at various times during the following days.

The time constant switch on the Stack Monitor was set at 40 seconds and the range was set at 100x during January 3. The airflow through the instrument was approximately 7.5 CFM.

The constant air monitor at the AREA hot cell facilities (not to be confused with the Stack Monitor) is a Nuclear Measurement Corporation model BM-2 with a linear count rate meter. The air at 5 CFM is drawn through a filter monitored by a GM tube. The output of the count rate meter is recorded. The chart from this recorder was removed at 1559 on January 9, 1961, and reproduction of portions of this chart are shown in Figures 87 and 88.

In Figure 87 the time is approximately 7 minutes slow. A sudden rise attributed to the SL-1 incident would then be placed at 9:04 PM on January 3, 1961. The instrument was off scale for three minutes and was at equilibrium again within twenty minutes.

The equilibrium point after the abrupt rise was 40 c/m higher than before. The maximum chart level was 2000 c/m and at the peak the radiation level was far in excess of this point.

A fluctuation also occurred twelve hours before the incident. This irregularity was due to the moving of a contaminated cut-off machine past the CAM. This CAM indication is supported by the other charts. On the area monitoring Brown Recorder chart, point #2 indicated a high reading of 5.2 mr/hr at 0936 which would be the time of the deflection on the CAM chart. Another deflection of 6.8 mr/hr was shown by this same point #2 at 0951.

Figure 88 shows the accumulation of airborne radioactivity on the CAM filter which put the instrument off scale. Evidence indicates the CAM did not change from the 2x, or 2000 c/m, range. This buildup started at 1345 January 4, 1961 and was off scale by 1515. The trace came back on scale at 1720 showing radioactive decay until midnight. The trace stayed more or less level at 1600 c/m until 0500, January 5 when it started to rise again. There was a variation from 1600 c/m to 1840 c/m until about 2200, when the instrument went off scale and stayed there.

V. EVALUATION OF ACCIDENT

A. EVIDENCE FOR A NUCLEAR POWER EXCURSION

The conclusion that a nuclear excursion occurred in the SL-1 reactor is a rather obvious one. Nevertheless, it is worth while to examine critically the evidence for such an excursion. There are numerous mechanical indications that an explosion of some kind occurred within the reactor vessel. Further, the contamination that has been observed on the clothing of the men removed from the building and on other objects from the building shows definitely the presence of fission products and enriched uranium. Although an explosion that would blow some fraction of the fuel out of the reactor and would reduce portions of the fuel plates to small fragments presumably could result from either a nuclear or a chemical energy release, the occurrence of a nuclear excursion of substantial energy release is established by the following additional evidence, recorded in section III.

1. The activation of the gold and copper samples recovered from the reactor building. This could be caused only by neutrons.

2. The Sr-91 measured in the fuel sample from the clothing of one of the men. This measurement shows far too much of the short-lived (9.7 hour) Sr-91 to be accounted for by the steady reactor operation prior to December 23.

3. The sudden, quickly decaying, burst of radiation recorded by the monitors in the AREA hot cell building. This rather obviously is gamma radiation from the fuel (probably that ejected from the reactor), and shows a very large percentage of short-lived fission products.

The above evidence constitutes what appears to be unmistakable proof of a nuclear excursion in which the maximum reactor power exceeded the normal steady operating power by several orders of magnitude.

B. ENERGY RELEASE REQUIRED TO PRODUCE OBSERVED RESULTS

The second consideration of importance is whether a nuclear excursion could account for all the observed damage within the reactor building, and for any other observable evidence. The main non-nuclear evidence to be accounted for is the following:

(1) The ejection of shield plugs, and the damage done by the plugs in their trajectories; the blowing-off of a cover plate that is thought to have been bolted down on the vessel head; and the ejection of iron punchings from the reactor shield.

(2) The existence of reactor fuel outside the reactor tank, at least some of it in finely divided form.

(3) Burns suffered by one of the men (No. 3).

Items 1 and 2 above would be accounted for rather obviously if the excursion had been of sufficient energy to melt some of the reactor fuel.

Some analysis has been made of the items mentioned in 1 to determine what magnitude of pressure surge within the reactor vessel would account for them.

An analysis was made for the shield plug which lifted the number 3 crew member and penetrated the fan floor and remained stuck until removed on January 8, 1961. The assumptions were made that the shield plug penetrated the fan floor and was constrained by the largest diameter flange, and that the fan floor was constructed of a single sheet of metal equal in thickness to the sum of the thicknesses of the two sheets in the actual floor. The calculated average accelerating pressure, acting on the shield plug as a piston over the length of its engagement with the nozzle in which it normally sat, was 300 psi. It seems reasonable to assume that the maximum pressure might have been about twice this value or 600 psi. Another shield plug, after ejection, struck and bent one of the fan floor I-beams. In order to evaluate the forces involved a test was performed by dropping a 210-pound steel weight onto a similar I-beam. By determining the energy required to deform the I-beam to a comparable bent configuration, an estimate of 203 psi was obtained for the average pressure accelerating the shield plug.

These estimates indicate that the average pressures required to account for the observed effects of shield plug ejection lie in the 200 to 300 psi range, far below the pressure that would cause a failure of any component of either the reactor vessel or the vessel head. The head is the weakest member of this complex. Calculation indicates that a static pressure of 1570 psi would be required to cause failure. However, it is not obvious that the average pressure acting on the shield plugs is a reliable indication of the maximum pressure exerted on the reactor vessel.

The ejection of iron punchings from the top vessel shield could be explained by impact on the bottom of the vessel head which probably occurred during the excursion. The probable cause is the impact of water on the underside of the head. It must be remembered that the water in the vessel was cold, and that the probable nuclear energy release was too small to raise the average temperature of all the water in the reactor vessel to the boiling point. Thus, one visualizes the generation of a high pressure by the local vaporization of water in the core region, and the acceleration of water by the expansion of this local steam volume. The initial effect will be a net downward acceleration of the pressure vessel. As the water level in the vessel rises because of the steam expansion, the air in the space above the water will be compressed, and may initiate the ejection of the shield plugs before the rising column of water reaches the vessel head. When the water strikes the head, the upward acceleration of the head and the vessel may be rather large, and sufficient to eject the iron shield punchings. For example, if the water exerts momentarily a pressure of 600 psi on the head, the total upward force on the head would be over the one million pounds, some forty times the weight of the pressure vessel.

In the light of the processes visualized above, it must again be said that the pressure exerted on the reactor vessel head -- or on the shield plugs -- is not related in a simple way to the pressure exerted on the lower portions of the reactor vessel during the period of acceleration of the reactor water. If the minimum period of the nuclear excursion was in the 5 to 10 millisecond range (as appears to be the case), and if the duration of the main accelerating pressure pulse was roughly the same as the minimum period (as has been observed at longer periods in the BORAX and SPERT tests), then maximum vessel pressures in the range 1000 to 2000 psi, or even higher, would not be incompatible with the observations on the shield plugs.

Past experience with experimental reactor excursions, particularly with the final BORAX-I excursion, provides ample evidence that a nuclear excursion in an SL-1 type reactor could produce results of the magnitude discussed above or larger. The only uncertainty is how large a nuclear energy release would be required.

The mechanical consequences of a nuclear release are not related in a simple way to the magnitude of the nuclear energy generation. This fact has been demonstrated many times experimentally and is obvious from simple theoretical considerations. For example, the energy required to raise all the water in the SL-1 core from its initial temperature (about 100°F) to the boiling point would amount to about 57-Mw-sec. Thus, the water, if it absorbed all the energy of the nuclear release, could act as a sink for as much as 57-Mw-sec of energy without any appreciable pressure increase; yet we know from the experiments with BORAX and SPERT that an energy release of this magnitude could result in substantial mechanical effects. Even though the magnitude of a nuclear energy release cannot easily be related to the resulting mechanical effects, one is nevertheless greatly interested in the total energy because it is the only characteristic of the excursion that can be related directly to those other results, such as activation levels and fission product concentration, that are unequivocally nuclear in origin.

In attempting to relate nuclear energy release to mechanical results, it should first be noted that experimental nuclear excursions (of the BORAX-SPERT type) have not been observed to generate very high pressures when the energy release has been insufficient to cause melting of the fuel plates. The highest reported pressure for a non-melting excursion is about 70 psi, which was observed in a 5 m sec. excursion of the BORAX-I reactor, made with the reactor coolant initially at saturation temperature.(28) The highest reported pressure from the SPERT experiments is approximately 50 psi, observed in a 5.5 m sec. transient made with initially cold reactor water.(29) In the latter experiment, the fuel plate surface temperature reached a measured maximum value of 590°C. If that measurement is correct, the fuel plate came very close to melting. It is worth noting also that the peak pressure observed in the SPERT experiments, for an excursion of given period, is lower for cores with wide fuel plate spacing (like SL-1) than for cores with narrow spacings. It is true that the pressures measured in BORAX and SPERT were probably not as high as the pressures actually existing at, say, the center of the core, because pressure measurements at such locations are extremely difficult. Nevertheless, the observed pressures were probably comparable to, or higher than, those exerted on the reactor vessel. In the final BORAX experiment, in which a substantial fraction of the fuel was melted, the pressure was very much higher -- at least as great as 6000 psi.[30]

Attempts to calculate the pressures produced by nuclear excursions are subject to very large uncertainties because the process of pressure generation involves the local formation of steam in water whose temperature is below that of saturation. Thus, the generation of a transient dynamic pressure depends upon the difference between the rate of steam generation and a rate of steam condensation, both of which depend upon a complex transient heat transfer situation. When melting occurs, the uncertainty of the situation increases greatly because the surface area of the fuel increases to an unknown extent. In view of these uncertainties, it appears that the best assumption one can make is that the pressure generated by a nuclear excursion in a reactor like SL-1 would cause only a relatively small pressure rise (less than 100 psi) unless the energy production were sufficient to cause fuel plate melting. If melting of a substantial fraction of the core occurs, rather high pressures are probably to be expected. Inasmuch as the peak pressure in the SL-1 excursion appears to have been a good deal less than that in the BORAX excursion, it may well be that the extent of melting in SL-1 -- or at least the extent of melting relative to the amount of water present -- was less than that of the BORAX case; indeed it seems reasonable that a relatively small amount of melting could have produced pressures in the 600 psi range. It is doubtful that any amount of theoretical analysis alone can give any closer estimate than this of the extent of the nuclear excursion necessary to produce the observed mechanical effects, although later observations may throw some light on the situation.

The energy required to raise the temperature of all the SL-1 fuel plates* to the melting point is 97 Mw-sec. The melting process would absorb another 55 Mw-sec in the latent heat of fusion if all plates melted. Since the spatial distribution of power in the reactor was far from flat, and since the centers of the fuel plates would get considerably hotter than the surfaces in a short period power transient, the actual energy produced by the reactor before all of the fuel plates melted would be several times larger than the figures above. Moreover, the experimental indication is that in BORAX-type transients, only a fraction of the energy -- roughly half -- appears as temperature of the fuel plates, the remainder being transferred to the water. On the other hand, since we do not postulate complete melting of the fuel, it is conceivable that the total energy release could be considerably smaller even than the 152 Mw-sec mentioned above. A more careful consideration of these points is given in a following section, but for the moment it seems reasonable to say that an energy release in the 100 to 200 Mw-sec range would be consistent with the observed mechanical results of the SL-1 excursion.

C. NUCLEAR INDICATIONS OF MAGNITUDE OF ENERGY RELEASE

The precision of the nuclear indications of the energy release is rather poor. These indications consist of the Sr-91 determination, the indications of the AREA monitors, and the activations of gold and copper samples.

The nominal value of the nuclear energy release indicated by the Sr-91 is 50 Mw-sec, but values from 21 to 64 Mw-sec would lie within the range of uncertainty of the radiochemical determination. Furthermore, the Sr-91 content of fuel fragments determines only a lower limit to the energy release, inasmuch as the initial fission product, which produces Sr-91 as a daughter, is Kr-91 (half-life 9.8 sec); it may have escaped partially from the fuel sample if the sample was ever in the molten state.

---

* Active portions only: i.e. uranium-aluminum "meat" plus immediately adjacent cladding.

The AREA monitors indicate a considerably larger energy release, but again the uncertainties are large. These monitors, at the time of the excursion, showed a burst of high radiation intensity which decayed rather rapidly, superimposed on a radiation level which decayed only very slowly. The simplest assumption is that the radiation is from fuel expelled from the reactor, that the initial high indications were from gamma rays from short-lived fission products produced during the power excursion, and that the later indications were from gamma rays emitted by long-lived fission products formed during the period of operation prior to December 23. The basic principle of the determination is to compare the radiation indications of the monitors during the short time interval immediately after the excursion with their later indications. Thus on the assumptions that there was no important preferential escape of fission products, and that the distribution of the fuel "seen" by the monitors did not change after about the end of the first minute following the excursion, one gets a comparison of the relative numbers of fission products formed in the excursion with the number of fission products remaining from the operation prior to December 23. The analysis involves calculation of the relative attenuations of the short-lived gammas (predominately of higher energies) and the long-lived gammas by the air between the SL-1 and the AREA locations (approximately 500 feet) and by any other shielding materials that intervene. This involves some uncertainties connected with air scattering, but there are much more important uncertainties in the total effective amount of shielding, and in the time of occurrence of the excursion.

Those monitors sensitive enough to give a reliable reading on the long-lived fission products were recorded by a recorder that printed out for any given monitor at 3-minute intervals. Since there were several monitors connected to the print-out recorder, all of which observed the effect of the excursion, it is possible to establish the time of the excursion (on the time scale of the recorder chart) to within an uncertainty of one minute. Any closer specification of the time must be inferred from the shapes of the decay curves recorded by the monitors. One possibility is to try to fit the (normalized) decay curves of the print-out monitors to the curves of the two monitors -- the stack monitor and the constant air monitor -- that were recorded continuously, and that were too insensitive to be used themselves for an energy determination. The results are shown in Figures 89 and 90. In each figure the rapidly-decaying component of each monitor record is plotted along with the continuous curve from the stack monitor. Each record is normalized to the stack-monitor curve at its second print-out point. The second point, rather than the first, was chosen for normalization because the stack-monitor record appears to have gone off scale, and its early readings may be distorted. In Figure 89 it is assumed that the excursion occurred one minute before the first print-out (monitor #1); in Figure 90 this time interval is reduced to 1/2 minute. It is evident that this approach does not allow one to increase the precision of the time estimate. It is also evident that the print-out records decay less rapidly than the records of the continuously-recorded instruments. This difference has not yet been explained.

A second approach is to try to fit the records of the print-out monitors by a simplified theoretical calculation. As the first step in this analysis, the readings of all the monitors were plotted, all normalized to the same value at the time 23 minutes after the assumed time of occurrence of the excursion -- after the short-lived components had decayed below the observable level. Figure 90 shows this curve for the case of a one-minute time interval between the burst and the first reading. The assumption was made that the readings beyond 23 minutes were due to fission products from long-term reactor operation at 2.25 Mw up to December 23 (the approximate average power over the preceding few months), and an attempt was made to estimate what burst of fission energy would produce enough short-lived fission products in the same fuel to give the additional activity observed in the time interval up to 23 minutes.

The estimate depends rather strongly on the amount of shielding between the fission products and the monitor, for the softer gamma rays from the long-lived fission products are attenuated more strongly. The minimum assumption is about 500 feet of air plus 2 inches of pumice concrete, known to be present. Estimates were made for this case, and for the case of an additional 12 inches of ordinary concrete, assumed to be effective in the SL-1 building. The effects of this shielding are tabulated below, where the ratio,

$$\frac{\text{Gamma level due to fission products from 1 Mw of steady power, after 11 days}}{\text{Gamma level due to fission products from 1 Mw-sec burst, after 1 minute}}$$

is given. In computing air attenuation only the first scattering was taken into account.

Shielding	Ratio: $\frac{\text{Gamma level from 1 Mw steady power F.P.}}{\text{Gamma level from 1 Mw-sec burst F.P.}}$
None	17.7
500 ft air	15.9
500 ft air + 2 in. pumice concrete	14.4
500 ft air + 2 in. pumice + 12 in. ordinary concrete	6.0

In Figure 91 the theoretical curves for the two different shielding cases are plotted, for an excursion energy of 400 Mw-sec. The more strongly shielded source gives a reasonable fit, while the less strongly shielded source gives a curve that is definitely too low. The latter case could be made to fit reasonably well if the assumed energy of the excursion were approximately doubled. It goes without saying that the amounts of fuel constituting the effective radiation sources are quite different for the upper and lower curves of Figure 90. For the upper curve the source amounts to approximately 2.8 per cent of the SL-1 fuel, while for the lower curve it is only 0.055 percent.

Figure 92 shows the results of a similar analysis for the case in which the time interval between excursion and first reading is assumed to be 1/2 minute. The excursion energy for both curves is 200 Mw-sec. Both curves decay too rapidly.

Evidently the analysis will show lower burst energies as the postulated amount of shielding is increased. It is questionable, however, whether a very large shielding increase above the maximum assumed here is reasonable, for the fraction of the fuel that must be assumed to be "visible" becomes too large. If the basic approach used in the analysis of the data is applicable, it appears that the excursion energy cannot be much less than 300 or 400 Mw-sec. If short-lived fission products were expelled from the reactor in disproportionate quantities -- as is perhaps not unreasonable, since they include many of the gaseous and volatile products -- then the AREA monitor records can give little information on the energy release. If the radiation "seen" by the monitors was primarily scattered radiation from inside the reactor vessel, then the analysis must be modified, but may still yield information. It does not appear worth while to analyze the scatter radiation case until further operations on SL-1, such as filling the vessel with poison solution, have given additional information.

The activations of gold and copper samples, equivalent to some 10^{10} thermal nvt near the top (outside) of the reactor vessel, indicate, if anything, a higher energy release than either of the preceding considerations. Indeed it is difficult to understand how enough prompt neutrons could have escaped if the initial water level was at the point estimated, and if only a single nuclear excursion occurred. If these activations were produced by prompt neutrons from a burst of, say, 100 Mw-sec energy content, then the shielding between the reactor core and the activated samples could have been no more than the equivalent of 1 or 2 feet of water. Just before the excursion, the reactor vessel was almost full of water. It is well known, from the BORAX and SPERT experiments, that in single transients the nuclear reaction is over before any large motion of the water takes place. Consequently, it is hard to see how so many prompt neutrons could have escaped from the initial excursion, even though much of the water may have been expelled from the tank as a consequence of the excursion. Calculation indicates that the expulsion of half the water from the reactor tank would take about half a second even if the pressure in the tank remained as high as 500 psi over the entire period of expulsion.

An alternate possibility is that the activations were caused by delayed neutrons from fuel that was expelled from the reactor vessel, or was plastered inside the top of the vessel during the excursion, or was exposed shortly after the excursion by the expulsion of water. A point source of fresh fission products, produced from an instantaneous burst of fissions equal to 1 Mw-sec of energy production, could supply a total neutron dose (current) of about 4.6×10^8 neutrons/cm^2 at a surface ten feet away. Thus, if 10 per cent of the fuel from a 100 Mw-sec excursion acted at an effective distance of ten feet, a supply of about 4.6×10^9 neutrons/cm^2 would be furnished to the receptor.

These neutrons would have to be slowed down to become effective for activation, and after moderation the major fraction of them would be lost by absorption in the moderating material before diffusing to the receptor. However, the average energy of the delayed neutrons is only about 400 Kev, and they would be moderated rather quickly by any hydrogenous moderator that was present. Very roughly, the fraction of neutrons lost to the moderator would be proportional to the ratio $\frac{\sqrt{\tau}}{\sqrt{\tau}+L}$, where τ is the slowing down area and L^2 is the thermal diffusion area. For 400 Kev neutrons in water, for example, $\sqrt{\tau}$ is not much greater than L. Somewhat more precisely, a "two group" diffusion-theory calculation indicates that if a uniform current density J of 400 Kev neutrons enters the plane surface of a semi-infinite body of H_2O, a current of thermal neutrons equal to about J/4 will emerge from the same surface. Thus it appears that the supply of thermalized delayed neutrons would be lower than the supply of unmoderated delayed neutrons by at least a factor of four.

Combining the rather rough considerations described above, it appears that the delayed neutrons from fuel that had generated 10 Mw-sec of excursion energy (10 per cent of the fuel from a 100 Mw-sec excursion) would fail to account for the activation levels by something like a factor of 10. This factor could be wiped out if the fuel were localized sufficiently near the receptors. Such a localization is conceivable, but does not seem probable.

Another possible explanation of the high activation levels is the occurrence of multiple power excursions, or continued operation of the reactor after the initial excursion. These possibilities are discussed in a later section.

D. EXCESS REACTIVITY REQUIRED FOR NUCLEAR EXCURSION

Having decided that melting of part of the fuel is a reasonable approximate criterion for the production of a nuclear excursion consistent with the observed results, one next asks how much excess reactivity would be required to produce such an excursion. For a reasonably quick estimate, it was decided that the most suitable procedure was an almost entirely empirical extrapolation of the SPERT results. The BORAX results are also applicable, and past work has indicated that they correlate reasonably well with the SPERT results; but it has been found that a knowledge of the pressure attained during the excursion is necessary for a good correlation, and the pressure data in the BORAX experiments are fragmentary.

The basic correlation depends upon relating the surface temperature of the hottest fuel plates in the reactor to the period of the excursion. Once this has been done, relatively simple considerations lead to reactivity values and estimates of total energy.

If it is assumed that the important shutdown mechanism in short period excursions is the formation of steam voids in the core, it seems reasonable that the rate of steam formation should be related to the fuel plate surface temperature through a simple relation of the heat-transfer type, and the most straightforward assumption is that the rate of steam production will be a simple function of the difference between the plate surface temperature and the saturation temperature of the water adjacent to the plate.

Further, it is known that if one plots for a BORAX-type excursion (Ref. 29, Fig. 42) the ratio P/P_{max} as a function of the ratio t/Υ, one gets a generalized power-excursion curve which varies remarkably little from excursion to excursion over a wide range of periods. In this plot, P is the instantaneous value of the power, P_{max} is the maximum power, at the peak of the excursion, t is the time from an arbitrary zero, and Υ is the steady exponential period of power rise prior to the termination of the excursion. Excursions are frequently characterized by the reciprocal of this period, designated by α. As a result of this relatively constant generalized shape of the power curves, a reasonable assumption is that the amount of steam required to remove the excess prompt reactivity of the reactor is formed in a time interval proportional to, or nearly proportional to, the period Υ.

The amount of steam required is just the ratio of the prompt excess reactivity, ℓ/τ (or $\alpha\ell$) to the steam void coefficients* of reactivity, C_v'.

These considerations suggest that one plot the difference between the maximum fuel plate temperature and the transient saturation temperature of the water ($T_{sat} = T_{max} - T_{sat}$) against the quantity

$$\frac{\ell}{\tau} \cdot \frac{1}{C_v} \cdot \frac{1}{\tau} \cdot \frac{1}{A} = \frac{\alpha^2 \ell}{C_v' A}$$

where ℓ is the effective prompt neutron lifetime, C_v' is the void coefficient of reactivity, in k_{eff} per cm^3 of steam, and A is the fuel plate area. The ratio should be roughly proportional to the average volumetric rate of steam production per unit area of fuel plates.** If the correlation is made for fuel plates of constant area, A may be replaced by N, the number of plates in the core; if the void coefficient is expressed in terms of dollars of reactivity per cm^3 of void (C_v) a factor of β (the delayed neutron fraction) enters, and the expression becomes:

$$\frac{\alpha^2 \ell}{C_v N \beta} \quad .$$

Since the temperature difference to be plotted is that between the plate surface temperature and the transient saturation temperature of the water, one needs to know the transient pressure generated in the water. It was found from the measurements on three different SPERT Reactors, that the measured peak pressures for a given reactor were approximately proportional to α^3. At a given value of α , the

* In this simplified approach the average coefficient (over core volume) is used. This is obviously a gross approximation, but improvements on it appear to involve extensive complication.

** Logically, one should use the mass rate of steam formation. The correlation apparently works because the transient pressure (and hence steam density) is not affected strongly by any variable except α.

pressure decreased, from reactor to reactor, as the plate spacing became larger. Although the variation with plate spacing was not defined precisely by the experimental results, it appeared that a reasonable correlation was given by the assumption:

$$P_{max} = f\left(\frac{\alpha}{\sqrt[3]{\mu}}\right)$$

In Figure 93 the measured maximum pressures for excursions in the three different SPERT reactors are plotted as a function of α , and as a function of $\alpha / \sqrt[3]{x}$. Also plotted, in Figure 94, are the same type curves for the first SPERT core tested (17/28), which does not correlate with the subsequent three cores. The reason for this lack of correlation is not obvious, but the construction of the first core is known to be different from the construction of the later cores. In predicting the SL-1 pressure, the curve of Figure 96, the one applying to three different cores, was used. The result is Figure 95, which shows the predicted pressure in the fuel element end box of the SL-1 reactor as a function of period of the power excursion. This curve is obviously subject to a good deal of uncertainty; but the uncertainty is not an important one, since the pressure curve is used for what amounts to a relatively small correction on the fuel plate surface temperature calculation.

In Figure 96 the values of ΔT_{sat} are plotted against the ratio

$$\frac{\alpha^2 \ell}{C_v N \beta}$$

for all the reported SPERT excursions (four different cores) for which both fuel plate temperatures and transient pressures were measured. The range of core characteristics covered is wide, as may be verified from Table XV, in which the characteristics of the four cores are listed. Although there is a rather wide scatter of the points, the scatter for a single reactor (17/28) is responsible for almost all of it, and the correlation of the results from different cores is good, although it may be fortuitous. It is this curve (Fig. 96) that is used with Figure 95 to predict the maximum fuel plate temperature, as a function of α ,

TABLE XV

STATIC CHARACTERISTICS OF SPERT I CORES
(from Reference 31)

Core	B-12/64	B-16/40	A-17/28	B-24/32
Clad Material	Al	Al	Al	Al
Critical Mass, Kg U^{235}	4.3	3.6	3.9	4.3
Total U^{235} Loaded, Kg	5.4	4.5	4.7	5.4
H/U Ratio	760	540	320	270
M/W Ratio	0.46	0.63	0.79	1.14
Available Excess Reactivity, $	4.3	5.6	5.2	6.6
Temperature Defect, 20^{o}-95^{o}C,$	1.44	1.67	1.47	1.73
Temperature Coefficient, 20^{o}C,$/^{o}C	-1.8×10^{-2}	-1.7×10^{-2}	-0.67×10^{-2}	-1.1×10^{-2}
Temperature Coefficient, 95^{o}C,$/^{o}C	-2.0×10^{-2}	-3.4×10^{-2}	-2.7×10^{-2}	-3.4×10^{-2}
Central Void Coefficient, $/$cm^3$	$+0.8 \times 10^{-4}$	-4.7×10^{-4}	-9.3×10^{-4}	-17×10^{-4}
Average Void Coefficient, C_v $/$cm^3$	-0.93×10^{-4}	-2.9×10^{-4}	-4.6×10^{-4}	-7.3×10^{-4}
ℓ/β (sec)	11×10^{-3}	10×10^{-3}	7×10^{-3}	7×10^{-3}
$\dfrac{C_v'/\beta}{\ell/\beta}$ ($\Delta k/cm^3$-sec)	-0.009	-.03	-.07	-0.10

for the SL-1 reactor.

Figure 97 gives the results of the SL-1 predictions. The lower line shows the maximum fuel plate surface temperature consistent with the SL-1 reactor characteristics tabulated in Table XVI.

TABLE XVI

CHARACTERISTICS OF SL-1 USED IN TRANSIENT ANALYSIS

Total Number of Fuel Plates	360
Equivalent Number of SPERT Plates (on basis of equal total surface area)	451
Fuel Plate Meat Thickness, in.	0.050
Fuel Plate Clad Thickness, in.	0.035
Fuel Plate Total Thickness, in.	0.120
Coolant Channel Thickness, in.	0.310
Initial Water Temperature, $^{\circ}F$	100
Effective Prompt Neutron Lifetime	5.6×10^{-5}
Effective Void Coefficient of Reactivity, % k_{eff}/% void	-0.20
Maximum/Average Power Ratio	3.0

Two obvious objections to the method of prediction may be raised: the effect of the high water head above the SL-1 core has been neglected, as has also the effect of the larger fuel plate thickness (relative to the BORAX and SPERT reactors).

The neglect of the head effect is based on reference (29), which reports a series of SPERT tests comparing the transient behavior of a reactor with a 9-foot head with that of the same reactor when the head was reduced to two feet. Tests were run over a range of periods from two seconds to 10 m sec, with cold water and with saturated water. For the cold water case no head dependence was observed over the entire range of periods. With saturated water the peak excursion power was higher when the 9-foot head was used. The percentage increase in peak power caused by the higher head decreased with decreasing period, amounting to only 25 per cent at a period of 10 m sec.

The fuel plates of the SL-1 reactor, although similar in composition to the SPERT and BORAX plates, differ considerably in thickness. In the correlations, it has been assumed that this difference in thickness has no effect, and this assumption is expected to be a relatively good one, since it is only the surface temperature of the plates that is being correlated. Because of the plate thickness there is, however, a very substantial temperature drop from the center of the fuel plate to the surface. This temperature difference has been computed on the assumption of solid conduction in the water adjacent to the fuel plates. This assumption gives a lower limit to the ratio of fuel plate center temperature to surface temperature. The results are plotted in Figure 98. By combining the result of this calculation with the predicted plate surface temperatures, the upper curve of Figure 97 results, giving the center temperature of the hottest fuel plate as a function of the reciprocal period of the excursion. These curves predict that the center temperature of the hottest fuel plate would reach the melting point for an excursion of reciprocal period 80 sec^{-1}, and that the surface temperature would reach the melting point for a reciprocal period of 190 sec^{-1}. These values of α correspond to periods of 12.5 m sec and 5.3 m sec, respectively; the corresponding excess reactivities are 1.15 per cent and 1.76 per cent.

Figure 97 shows that the temperature drop in the fuel plate is quite important in determining the central temperature of the plate during a short-period excursion, even though the plate is relatively thin and has a high thermal conductivity. This effect is shown more directly in Figure 99. The figure applies to the case of a fuel plate in stagnant water, heated by a power generation that is increasing exponentially with period τ. The initial temperature of the plate and the water is $38^{\circ}C$ ($100^{\circ}F$), and the quantity plotted is the computed central temperature of the plate at the time the surface temperature reaches $121^{\circ}C$ ($250^{\circ}F$). The latter temperature was chosen as a representative surface temperature at which boiling might begin for periods of a few milliseconds. It is evident from the figure that when the period is less than 3 milliseconds the central temperature reaches the melting point of aluminum before the surface temperature reaches the boiling point of the water. Also plotted on the figure are three calculated points for the BORAX-I reactor. The difference in behavior due to the difference in plate thickness (120 mils for SL-1 versus 60 mils for BORAX) is striking.

With the aid of Figures 97 and 98 estimates can be made of the maximum amount of heat stored in the SL-1 fuel plates as indicated by plate temperatures during excursions of various periods. Experience indicates that the total heat production in any given excursion would be greater than the maximum heat stored in the plates, by a factor not greater than two.

To arrive at the total maximum heat stored, one must take account of the temperature distribution within the individual fuel plates -- from center to surface -- as well as the gross temperature distribution over the reactor core, corresponding to the neutron flux distribution. Treating these separately: if the gross power (or neutron flux) distribution in the core were perfectly flat, then a power excursion of sufficient magnitude to heat the centers of the fuel plates to the melting point would store a maximum of about 60 Mw-sec of heat in the plates; an excursion of sufficient magnitude (and short enough period -- Figure 97) to heat the surfaces of the plates to the melting point would store a maximum of something like 240 Mw-sec. If the gross over-all ratio of the maximum-to-average power density in the reactor is 3.0, then the energy stored in all the fuel plates, at the time the hottest segment of the hottest plate reaches the center-melting condition, is about 20 Mw-sec, just a factor of 3 lower than the estimate for the flat-flux case. Similarly, the energy storage at the time the hottest segment reaches the surface-melting condition is about 80 Mw-sec.

E. POSSIBLE MEANS OF REACTIVITY ADDITION

The preceding estimates indicate that the minimum amount of excess reactivity required to cause the "melting" nuclear excursion is in the range of 1.2 per cent to 1.8 per cent k_{eff}. A further consideration is whether this amount of reactivity could be added to the reactor sufficiently rapidly by pulling the central control rod manually.

A spare SL-1 control rod actuator assembly was used for mock-up on which the speed of manual rod withdrawal was measured for several subjects. The equipment is the same as that on SL-1 except for the control rod, which is simulated by a weight to give a total movable load of 84 lb., the net weight of the SL-1 movable assembly in water. This arrangement is clearly shown in an early mock-up (Fig. 100). The test was conducted by instructing the subject to lift the rod as rapidly as possible, while an electric timer, measured the elapsed time from beginning of rod motion to some predetermined distance of withdrawal. Distances up to 30* inches were measured. Figure 101 shows a man in position to start lifting the rod and also after lifting 30 inches. (This mock-up is identical to that of Figure 100, but is set in a pit because the stand was not available).

Inasmuch as a single timer was used, it was necessary for each subject to lift the rod a number of times to obtain a complete curve. The results are plotted in Figure 102. The lifting tool was an 18-inch length of straight pipe, 1-1/2 inches in diameter with a hex nut welded to the lower end to engage the threaded portion of the mechanism rack. The lifting test was done by three different men. Results from an earlier test are not shown as they did not go beyond 15 inches. The results of both tests are consistent and both have the wide scatter shown on Figure 102. The earlier tests included use of a Tee bar lifting tool with no significant difference in lifting time.

It is a simple matter to combine the results of Figure 102 with the calibration curve of control rod No. 9 (the center rod) to obtain a curve of possible reactivity increase as a function of time for the manual withdrawal of that rod. The results of that operation are shown in Figure 103. Two curves are shown, one (Fig. 27) corresponding to the control rod calibration from CEND-1005, the other (Fig. 25) corresponding to the calibration curve measured by Argonne National Laboratory during the early operation of the reactor. The two different rod-worth curves lead

* Since the rod, in its disconnected position, is about four inches below the zero of the rod position indicator, these withdrawals correspond to "indicated positions" plus 4 inches i.e. a withdrawal of 20 inches from the disconnected position represents a withdrawal of 16" from the "indicated zero" position.

to two different estimates of the shutdown reactivity of the reactor, with the result that the estimated required rod withdrawal (from indicated zero) to reach an excess reactivity of 1.8 per cent is nearly the same in both cases - 23 inches on the basis of Figure 25 and 24 inches on the basis of Figure 27.

From Figure 103 it can be shown that the manual rate of reactivity addition is quite adequate to achieve an exponential power rise of period as short as 5.3 m sec. According to the figure, the possible rate of reactivity addition over the important range is at least \$20 per second. With this rate of addition, an estimate by the method of Hurwitz (34) indicates that the power increases from source level by about a factor of 2×10^6 by the time a period of 5.3 m sec is reached. The maximum power reached in a self-terminated BORAX-type excursion has been shown to be, (28) roughly, about half the ratio of the total energy release to the minimum period. Thus the SL-1 excursion, if it reached a minimum period of 5.3 milliseconds and generated a total energy of 100 Mw-sec, would have reached a maximum power of:

$$P_{max} = \frac{100}{2 \times 0.0053} = 10^4 \text{ Mw or } 10^7 \text{ Kw}$$

Thus the rod withdrawal could have proceeded far enough to shorten the period to 5.3 milliseconds before the excursion terminated itself provided only that the source level was no higher than about:

$$S < \frac{10^7 \text{ Kw}}{2 \times 10^6} = 5 \text{ Kw}$$

The above reasoning indicates that the required rate of rod withdrawal to produce a period as short as 5.3 milliseconds was well within the limits of human capability. It does not attempt to explain why so large a withdrawal of the rod -- corresponding to nearly the full length -- might have been made.

Although manual withdrawal of the central control rod is a possible explanation, the question remains as to other possible sources of the reactivity addition. The only possible ones appear to be releases of chemical energy within the reactor vessel by some means. It hardly seems reasonable to invoke the possibility of a metal-water reaction in the cold system when other reasonable possibilities for reactivity injection are evident. The possibility of an explosion of hydrogen-oxygen mixture, or hydrogen-air mixture above the reactor water requires some consideration. A preliminary analysis of this possibility indicated that it is a remote one.

The pressure rise due to recombination of radiolytic hydrogen and oxygen has been evaluated. In this connection it is pertinent to note that just prior to the incident the water level was lowered from the top of the control rod drive nozzle to approximately 2-1/2 feet below the bottom of the head. This introduces about 30 ft^3 of air. Making allowance for hydrogen and oxygen which could possibly be trapped in the No. 5 control rod upper housing and the two water level control housings, the time required to build up a concentration of hydrogen such that the heat release on ignition would result in a maximum pressure of 50 psi is approximately 90 hours at the gas release rate expected with vigorous gas stripping. Under the conditions prevailing in the SL-1 vessel under which there is no appreciable gas stripping, the time required to obtain this concentration is about 300 hours. The hydrogen generation rates used are those reported in Radiation Decomposition of Water Under Static and Bubbling Conditions, by Sheffield Gordon and Edwin J. Hart in "Peaceful Uses of Atomic Energy - Second United Nations International Conference on the Peaceful Uses of Atomic Energy," Geneva 1958, Volume 29.

F. POSSIBLE ADDITIONAL ENERGY RELEASES

The remaining question is whether any other mechanism came into play after the initial nuclear excursion. In this case, the possibility of some chemical reaction between aluminum and water can probably not be eliminated. What can be said is that there does not appear to be any necessity for postulating an additional source of energy other than that of the nuclear excursion, and that if some chemical energy release occurred, it was not large compared to that to be expected from the nuclear excursion. On the other hand, the conditions in the reactor after fuel had melted may well have been favorable for a metal-water reaction, and the possibility that some reaction occurred cannot be ruled out.

The real problem, however, appears to be not the discovery of an additional source of energy release, but the rationalization of the quite large nuclear effects indicated by the AREA monitors and by the activated gold and copper samples. A possible explanation of these effects would be the continuation of nuclear energy generation after the initial excursion. Three different types of additional releases can be imagined:

1. The ejection, or partial ejection, of control rods by the pressure buildup in the reactor vessel may have led to a second excursion following very closely after the initial one. The energy release of the second pulse may have been considerably larger than that of the first, producing enough melting or deformation of the reactor core to render it permanently subcritical.

2. The reactor may have experienced a series of lesser "chugs" after the first excursion, these chugs continuing until the reactor was rendered subcritical through loss of water or through core deformation. This postulate seems less likely than the one above, for a rather special set of circumstances must be postulated to support it.

3. The reactor may have operated at some relatively low, fairly stable power after the excursion, until it became subcritical by the boiling away (or leakage) of water. This postulate seems least probable of all.

Any of the above possibilities could account for the observed high activation levels. Any excursion, subsequent to the initial one, which ejected fuel from the reactor, would tend to rationalize the high energy release indications of the AREA monitors with the relatively mild mechanical effects. To remove the apparent discrepancy between the indications of the monitors and the indication of the Sr-91 analysis, one would have to postulate that the sample analyzed for Sr-91 was ejected by the initial excursion, whereas subsequent excursions ejected additional fuel which contributed to the indications of the monitors.

G. SUMMARY AND DISCUSSION

The evidence available for evaluating the probable course of the accident is meager, and any analysis must involve a good deal of supposition. In this treatment it has not been considered worth while, however, to deal at length with remote possibilities that can only be subjects of speculation. Rather, the more obvious probabilities have been examined to determine whether they can account for the evidence gathered to date, and to discover what discrepancies exist. It is almost certain that as further evidence becomes available the quantitative results of the analysis will need modification, and it is possible that evidence may be discovered which will invalidate the basic suppositions. At present, however, the conclusions discussed below appear to be the most reasonable that can be drawn.

The following conclusions can be stated with conviction:

1. A nuclear energy release occurred, characterized by a maximum power level which was higher, by orders of magnitude, than the normal operating power of the reactor.

2. The nuclear energy release was sufficient to account for all the mechanical and thermal effects observed to date.

3. The addition of reactivity by manual withdrawal of the central control rod, in sufficient quantity and sufficiently rapidly to cause the nuclear energy release, was well within the limits of human capability. The estimated amount of rod withdrawal required to cause the excursion is large, corresponding to nearly the entire length of the rod, and evidence to establish a reason for such a hypothetical withdrawal is lacking.

The only available route to a consolidation of the mechanical and nuclear evidence with theoretical considerations, to form a quantitatively consistent picture, is via considerations of the magnitude of the energy release. At the present time these considerations show discrepancies if the simplest and most straightforward assumption -- the assumption of a single nuclear excursion which caused permanent shutdown of the reactor -- is adopted. Quite possibly the discrepancies lie within the limits of error of the observations and their interpretation. These limits themselves are highly uncertain. In any case, it appears that all the evidence could be explained by some modification of the "simplest" assumption quoted above. At the present time it can be said that none of the evidence indicates any phenomenon that cannot be explained in a straightforward way, but at the same time it must be said that the sum of the evidence does not define an unambiguous chain of nuclear events.

The foregoing general statement is amplified in the following discussion.

The Sr-91 determination indicates a nuclear energy release between 21 and 64 Mw-sec. Besides the uncertainty in the radiochemical determination, indicated by the spread in the values, the estimate involves the questions of whether the sample analyzed was typical, and whether some of the Kr-91 precursor of the Sr-91 escaped from the fuel.

The AREA monitors indicate a minimum energy release of about 400 Mw-sec. This minimum estimate could perhaps be reduced to 300 Mw-sec by further attempts at curve fitting. The estimate suffers from uncertainties in establishing the time of occurrence of the excursion (relative to the chart scales) and from uncertainties in the effective shielding of the fission products responsible for the monitor indications. Further, if gaseous and volatible fission products escaped from the fuel, errors would be introduced, probably in the direction of giving too high an energy indication, since the escaped products (predominantly short-lived) would probably occupy positions less shielded from the monitors than would the fuel fragments containing the predominantly longer-lived products. Finally, if future operations show that the monitor indications were due primarily to scattered radiation from inside the reactor vessel (i.e. the core was uncovered), then the records must be re-evaluated.

The gold and copper activations may be interpreted either as indications of high energy production in a single nuclear excursion, or as indications that the initial excursion did not shut the reactor down permanently. On the former supposition, it appears that the activations must be considered to be caused by delayed neutrons. The estimated activation by delayed neutrons depends on the assumption of the amount of fuel expelled from the reactor, and on the assumption of its location relative to the receptors. If it is arbitrarily assumed that 10 per cent of the fuel was expelled and was located at an effective distance of 10 feet from the receptor, with no shielding, and that the expelled fuel was typical, then the indicated energy of the excursion is at least 1000 Mw-sec.

If the supposition of reactor operation after the first excursion is adopted, then the relationship of the activation levels to other energy indications is not defined.

The estimates of the nuclear energy release required to account for the mechanical evidence are based on the judgment that some melting of fuel is required in order to generate transient steam pressures in excess of about 100 psi. The assumption of fuel melting is supported by the presence of fuel fragments outside the reactor vessel. The higher pressures generated by molten fuel are evidently due to the greater subdivision of the fuel and the resulting increase in heat transfer area. It therefore seems reasonable that the melting criterion that should be applied is that the surface temperature of at least some fuel plates must have approached the melting point. The estimates based on correlations of SPERT data indicate that the hottest points on the surfaces of the hottest fuel plates would reach the melting point in an excursion of 5.3 m sec period, caused by an excess reactivity of 1.76 per cent. If one assumes a value of 3 for the gross maximum/average power density ratio over the core, then the estimated total energy stored in all fuel plates at the time the hottest surfaces reached the melting point is 80 Mw-sec. On the basis of SPERT and BORAX experience, the total energy release should not be more than a factor of 2 above the maximum heat storage in the plates, and for the case of relatively massive plates, such as those in SL-1, it seems probable that the factor would be considerably less than 2. Further, the actual gross maximum/average ratio may have been a good deal higher than 3 if the reactor was made critical by the withdrawal of the central rod alone. A higher maximum/average ratio would yield a lower total power estimate. In view of these considerations, the figure of 80 Mw-sec is probably not far from a minimum limit for the energy of the excursion, although a somewhat lower value would not be inconceivable.

A rough maximum limit for the energy production in a single excursion can be set by the following consideration. Quite evidently the SL-1 excursion was not as violent as the final BORAX-I excursion, which released about 135 Mw-sec of energy. It seems very probable that if the fuel plates of the SL-1 had reached temperatures as high as those in the final BORAX excursion, the consequences would have been at least as severe. The total amount of aluminum in fuel plates in the SL-1 core was almost exactly twice as much as that in BORAX-I, and the effective heat capacity must therefore have been about twice as great. It therefore appears that an energy release equal to twice the BORAX release, or 270 Mw-sec, represents an absolute maximum limit, and it seems very probable that the release was actually considerably less than this.

The estimates of energy releases discussed above are summarized in Table XVII.

TABLE XVII

GROSS ESTIMATES OF TOTAL ENERGY RELEASE

(These estimates do not treat maximum/average power ratios consistently; see following discussion)

Basis of Estimate	Energy Release (Mw-sec)
$\frac{\text{Sr-91 Content}}{\text{U-235 Content}}$ of Sample of Debris	21-64
AREA Monitors	300 to 400 minimum
Au and Cu Activations	1000 minimum*
Theoretical Limits Based on Mechanical Evidence	80-270

On the basis of the gross values of energy release recorded in the table, it would appear that the minimum theoretical estimate nearly overlaps the maximum estimate from the Sr-91 determination, and in view of the large uncertainties involved, the difference could hardly be considered a discrepancy. It is necessary however to examine the meaning of the Sr-91 determination more carefully. Presumably the fuel that was ejected and analyzed came from the hottest portion of the core, and since the Sr-91 determination was converted to an absolute figure by comparison with the uranium content of the sample (rather than by comparison with a long-lived fission product), the determination actually gives an estimate of what the total energy release would have been if all regions of the core had produced an energy density equal to that of the hottest region. To

* On the assumption of a single excursion.

agree with the theoretical estimate, the Sr-91 value would therefore have to be higher by a factor of 3, the maximum/average ratio assumed in the theoretical estimate.

On the other hand, estimates based on the AREA monitors result from a comparison of the short-lived fission product activity with the long-lived fission product activity, and they should be independent of the maximum/average power ratio provided only that the power distribution in the core during the excursion was the same as the distribution during the preceding steady operation. This condition is certainly not satisfied precisely, and it is perhaps conceivable that the determination might be high by a factor of as much as 2 because of differences in the two power distributions. The estimates based on activations by delayed neutrons should also be reduced in the ratio (average power density/ (maximum power density), on the assumption that the ejected fuel is from the hottest core region. This would reduce the energy estimate to a minimum of about 300 Mw-sec. It can therefore be said that the activation estimates and the AREA monitor estimates are not in obvious disagreement, but they both appear to indicate energies somewhat higher than seems reasonable for a single excursion.

If multiple excursions occurred the upper limit of the total energy release set by mechanical damage considerations does not apply. In order for ejected fuel to register the nuclear effects of the multiple excursions it would, however, have to be ejected by some excursion subsequent to the first one. Under most circumstances, one would be inclined to assume that subsequent excursions would be less violent than the initial excursion, and therefore that the major portion of the ejected fuel would be ejected by the first excursion. In the SL-1 case, it is conceivable that a second, more violent excursion followed very shortly after the first one -- perhaps as a "tail" to the first one -- because of the ejection of the center control rod by the first pressure pulse. If the second excursion occurred while some steam from the first was still present in the reactor core, it is conceivable that the mechanical effects of the second may have been somewhat reduced by the cushioning effect of that steam.

The high activation levels of the gold and copper foils could be explained most easily by multiple excursions or by steady operation of the reactor after the initial excursion. Under such conditions, one can visualize a gradual decrease of water level in the reactor vessel, by either expulsion, leakage or evaporation allowing more and more neutrons to leak out as time goes on. If such operation involves power levels as high as about 1 Mw, one would expect to see evidence of the prompt fission gammas on the AREA monitors after the water level had fallen to the vicinity of the top of the core. Hence it is concluded that if operation continued after the first excursion, and was finally terminated by loss of water then the operation either involved maximum power levels less than about 1 Mw, or the operation was terminated within a few seconds after the initial excursion, too soon for its radiation to be resolved from that due to the first excursion by the AREA stack monitor. The possibility of leakage of the water out of the vessel cannot be discounted at this time, for even though the average pressures indicated by the ejection of shield plugs are no higher than 200 or 300 psi, the pressure in the vessel at core level may have been much higher, and could conceivably have caused a failure.

VI. FURTHER INVESTIGATION

A. OBSERVATIONS TO BE MADE AT SL-1

The most important effort in further investigation consists of gleaning all of the evidence available from the SL-1 reactor. As this information is obtained, the evaluation presented in this report will be reassessed. The results of analysis of new evidence collected may narrow the range of uncertainty in the evaluation presented in this report, thus, resulting in confirmation of the present conclusions and surmises, and/or in the development of new conclusions. To provide a basis for collection of such information a document was prepared setting forth the observations to be made during the recovery and decontamination of the SL-1 facility.(35) This is re-printed below:

"Observations to be Made During Recovery of SL-1 Facility for Evaluation of the Incident

"As operations leading to the shutdown and cleanup of the reactor and reactor building progress, a complete description should be compiled of the state of the reactor, the reactor building, the location of equipment and tools, the location of debris, etc. Whenever possible, this description should be illustrated by photographs which show the spatial disposition of all the objects found in and around the building subsequent to the accident. It should be borne in mind that at the time of observation it is frequently not clear what value should be attached to the material found; the only alternative to running the risk of losing information is to assemble as complete a record as possible. Care should be taken to record any rearrangements of the contents of the reactor building which result from penetrations made during core shutdown procedure. Any tools or equipment left in or about the building during operations subsequent to the accident should be recorded.

I. Information Obtained Outside the Reactor Vessel

A. Physical Arrangements

"As soon as is possible after the shutdown of the reactor an attempt should be made to obtain a complete photographic survey of the interior of the reactor building including both the operating floor and the fan floor. From these photographs, it should be possible to identify the tools and equipment which were available or in use at the time of the accident. A continuing search should be maintained for mechanical evidence of the force of the explosion. The trajectories and effects of missiles should be recorded in more detail as access permits. Any broken or overstressed parts should be examined as carefully as the radiation level permits, should be photographed completely, and should be preserved for future examination. Whenever work can be done on the head, its mechanical condition should be examined in detail, and any possible observations relative to the overstressing of the head, the bolts and all other parts should be made before the head or the bolts are otherwise disturbed. Particularly, all parts of the control rod mechanisms need to be located and sufficient evidence obtained to determine whether they had been on the reactor and ejected (along with evidence of trajectory), or whether they were lying in readiness for re-assembly.

"'It is pertinent to determine the actual configuration of all of the electrical units prior to the incident, therefore a check should be made to determine this. For example, a check on motor controllers to see if they are on, off, or reset; examine front and back of control panel for switch positions and jumpers, etc.'"[36]

B. Radiation Survey

"A complete map of the radiation intensity within the SL-1 reactor building should be made. The pin-hole camera or other techniques should be employed to locate and measure the main sources of radioactivity. The amount and extent of fuel expelled from the reactor core should be estimated from this map. Exact locations and conditions of each piece of fuel or other "hot" material should be logged and a permanent record kept of each item.

"Any entrance into the fan room should be made with great care in order not to disturb material that may be present in this area before photographs and radiation measurements can be made.

C. Evidence of Neutrons and Material Ejected from the Reactor

"Of primary importance in connection with the collection of evidence in the reactor operating room is a careful inventory of all items, their location, physical appearance and disposition.

"The following items may furnish evidence as to the nature or extent of the incident:

1. An inventory of fuel and fuel elements found outside the vessel should be made. The fuel may subsequently be analyzed for:
   (a) extent and location of melting
   (b) evidence of multiple melting and re-solidification
   (c) evidence of aluminum-water reaction
   (d) amount of U-235 burnup

   This last item will be helpful in correlating the final position of a given fuel sample with its position before the incident.
2. Flexitallic gaskets which were newly installed should be analyzed for chromium-51 (27 day half-life) and cobalt-58 (71 day half-life). Other new stainless steel items would yield the same information. It should be pointed out that these activities will be very low and it may be too late to obtain useful information from these items even now.
3. Number 4 control rod had a "stellite" bushing installed during shutdown. "Stellite" has a high cobalt content which would provide accurate information for flux calculations. It is very probable that this bushing was used once before about two years ago. If so, the residual cobalt-60 activity would render any analysis useless.
4. Several light bulbs with Tungsten filaments are readily available. The Tungsten activity (W-185 half-life = 74 days) will be very low and again it may be too late to measure this activity.
5. A sample of the material shown in the movie on the floor adjacent to the reactor is of interest. This may be blotting paper or perhaps pieces of aluminum which were ejected from the core in molten form and solidified on the floor. In addition, the nature of the white matter on the roof above the reactor should be determined.
6. Evidence of lifting the pressure relief valves should be collected either by examination of the valve or radioactive contamination on the downstream side of the valve. This may

furnish an indication of the pressure buildup during the incident.

7. Smears should be taken on the shield plug removed with the third body and checked for aluminum oxide, nickel, and enriched uranium. This information would give additional evidence as to aluminum-water reaction and the extent of core meltdown.
8. The filter inserts and ion-exchange resins in the coolant system and by-pass purification system should be retained for analysis of total and dissolved solids (especially boron), and fission products. If the coolant system was in operation during the incident, products of any chemical reaction in the core would be present. All water in the system between the reactor and the ion-exchange columns should be collected, if possible, to provide a sample of water from the reactor following the incident.
9. The volume of water in the contaminated water tank should be measured. This should indicate the volume of water in the vessel at the time of the excursion. A sample of this water should be analyzed for total and dissolved solids, pH, conductivity, and fission products.
10. Water samples should be obtained at as many points in the plant as possible; i.e., filters, ion-exchange columns, contaminated water tank and steam line. These samples should be analyzed for total and dissolved solids (especially boron and aluminum oxide) pH, conductivity and fission products. If the water circulation system was in operation, it may be possible to characterize any chemical reactions that may have initiated the nuclear excursion.

II. Evidence from Inside the Pressure Vessel

A. Physical Observation

"Visual (photographic) observations of the core should be made to the greatest possible extent. This should be begun before any objects inside the vessel are disturbed and should continue during the core disassembly. In this process the following should be looked for:

(1) the position of core components following the incident in an attempt to relate these with positions before the incident,

(2) the position and extent of melted fuel,

(3) the position of added cadmium strips in Tee-rod positions 2 and 6 with respect to the core,

(4) the number of boron strips still in the core,

(5) the control rod positions relative to the core,

(6) damage to control rod extension shafts.

During the disassembly of the core, a record of the positions and quantities of fuel, boron strips, cadmium strips, control rods and cobalt flux wires should be compiled. The exact location of each item prior to its removal from the vessel is of utmost importance. It will, no doubt, be found, as fuel samples are recovered that there are several characteristic types of samples - melted, partially melted, unmelted, etc. These should be examined for the following:

(1) size distribution of melted or nearly melted fuel particles,

(2) evidence of centerline melting in apparently unmelted plates,

(3) evidence of multiple melting and re-solidification,

(4) evidence of aluminum-water reaction.

In addition, the control rods and the core structure should be examined for evidence of melting during the incident.

B. Analysis of Core Components

"Once the core is disassembled, the following detailed analyses of the various items are recommended:

(1) Metallographic and chemical analysis of the fuel should be carried out to determine the pattern of melting of the plates (multiple melting, melting at the center line, etc.) and to search for oxides of aluminum and uranium. Analysis for the amount of U-235 burnup will be useful in correlating the position and extent of melting of a sample with its location before the incident (assuming that this is not obvious from the final reactor configuration).

(2) Recovery and analysis of some of the cobalt flux wires is of great importance. However, identification of the location, both radially and axially; of the wire in the core is of almost equal importance, since there will be an uncertainty of a factor of 4 or 5 in interpreting the wire activation if its location is completely unknown.

(3) A careful search should be made for corroded flakes of the boron-aluminum plates, or pieces of these plates lying at the bottom of the vessel. Both the boron-aluminum plates still in the core and any samples collected from the bottom of the vessel should be analyzed for boron burnup. An unirradiated boron aluminum plate should be analyzed for boron content to provide a base point.

"In regard to samples from the core, two AEC laboratories have expressed interest in obtaining fuel samples; ANL for aluminum water reaction analysis and ORNL for fission product distribution analysis. Requests from other sources are anticipated."

B. ESTIMATION OF THE INTEGRATED RADIATION DOSE IN OTHER BUILDINGS AND OFFICES OF THE FACILITY

In considering the effect of the incident, it is of interest to determine the radiation doses which would have been experienced by the occupants of the surrounding buildings of the facility if the incident had occurred during normal working hours. This question is not only of interest to the SL-1 but also has significance in connection with the evaluation of other reactor facilities. An attempt should be made to determine whether or not any of the personnel normally at work in the facility, other than those on the reactor floor, would have been subjected to lethal or near lethal doses of radiation from the fuel expelled from the reactor vessel into the reactor building. It is worth noting that the type of accident which actually occurred was not discussed in the Hazards Report and the consequences of a partial core meltdown and expulsion of fuel from the reactor vessel to the personnel normally in the facility was not evaluated. There does not appear to be, however, any good reason why this accident should have occurred preferentially during off-shift operation of the reactor rather than at any other time. In conjunction with these calculations, a determination should be made of the gamma ray shielding required on the exterior of the reactor building to reduce the level of radiation from an excursion of this type to a tolerable dose in the offices of the facility. An examination should be made of the evacuation plan in order to make an estimate of the dose that people would have obtained between the time the first alarm sounded, indicating that the reactor had undergone an excursion, and the time that they had left the facility.

C. FURTHER PHYSICS ANALYSIS

Further analysis can be made to refine the estimates of critical rod heights, boron burnup, etc., however these analyses are extensive due to the complexities of the SL-1 core. There is a reasonable likelihood that the further investigation under A above would be more meaningful if more precise numbers were available and on this basis, the following program should be considered.

1. Change of the Differential Reactivity Worth of the Central Control Rod With Lifetime

a. A complete analysis of the change of the differential worth of the central control rod in the SL-1 reactor during its life would be prohibitively costly both in manpower and computer requirements. However, a schematic study of the effect of uranium and boron depletion on the worth curve of the central control rod would be very desirable rather than attempt to provide a detailed analysis of the SL-1 reactor. This can be done by considering a slab reactor with slab control sheets, the outer control sheets being held at a constant withdrawal position throughout core life and the central control sheet being held at the same level during burnup. At the end of each time step, the central control sheet would be set at several different positions of insertion and the reactivity calculated for these positions.

If it is assumed that the boron poison strip corrosion was irradiation dependent, some account of this effect could be taken by carrying through the depletion calculation with one central control rod calibration at the end of the reactor life (essentially 931 MWD).

b. Some work has already been started and should be completed on a detailed evaluation by theoretical methods of the differential reactivity worth curve of the central control rod in the SL-1 reactor at the beginning of life in four conditions. These conditions correspond to the core with and without burnable poison, with and without Cd Tee rods. This work is connected with the question to what extent the presence of the boron or Tee rods affect the worth of the central rod.

2. Dependence of Control Rod Bank Position on Lifetime

The deviations of the observed bank positions from the roughly estimated prediction have been taken as an indication of the mechanical loss of boron poison from the core. In view of the various approximations made in the calculations it is felt by CEI that this type of evidence of boron loss is weak. It is, therefore, important to determine whether or not there was, if fact, any anomalous behavior of the control rod bank position during life. Because of the extraordinary complexity of the SL-1 reactor core, it is very difficult to obtain a precise theoretical prediction free from a large number of purely analytical uncertainties. There are two steps in this part of the study. The first step, which has been essentially completed and reported here, is to obtain some idea of the uncertainties resulting from a spectrum of disadvantage effects for the boron.

The second step that remains to be undertaken is a detailed axial synthesis of two-dimensional burnout calculations with discrete boron poison. A study of this sort for the SL-1 reactor entails an extensive analysis effort.

3. Study of Rod Bank Calibration Method

Estimates of the reactivity shutdown of the SL-1 reactor during its operating lifetime have been obtained from experimental rod calibrations made at a number of times during the reactor life. In general, the assumption has been made that the worth of the total rod bank is the sum of the worth of the individual rods. This assumption, while convenient, is always open to question.

Rod calibrations made by ANL personnel prior to power operation gave control rod worth values which for the most part were larger than those obtained by CEI. Many of these ANL measurements employed boric acid dissolved in the water to bring the reactor critical in various configurations. This procedure is objectionable on the basis that the boric acid solution changes the control rod worth. Whether or not this is a significant effect on their measurements can be evaluated theoretically.

Three analytical approaches are proposed to study some of the uncertainties involved in the rod calibration measurements.

a. Schematic Study of Control Rod Worth Additivity

The simplest method of studying the interaction effect between control rods in the SL-1 reactor is probably by means of a two-dimensional

X-Y slab reactor with control sheets similar to that discussed in 1-a. The worth of the control rod bank would be computed at a number of points between the all-in and all-out position. Then one control rod would be taken out of the banked configuration and its differential worth computed at a variety of bank positions, the positions being chosen in such a way that the reactor is at all times near critical. The final series of calculations would probably be made with one control rod fully inserted, one control rod in motion for criticality adjustment and the rest of the control rods in a banked configuration. This would correspond to one of the sequence of control rod calibrations. It should be emphasized that this entire study is schematic and is aimed at investigating the validity of reactivity additivity of the control rods rather than evaluating the worth of specific configurations of control elements in the SL-1 reactor.

b. Control Rod Worth Additivity in SL-1 Reactor

A more limited study would appear desirable of the control rod worth additivity in the SL-1 reactor by carrying out a few three-dimensional calculations in which the five control rods of the SL-1 are represented by smeared homogeneous poison in five cells. These calculations would be of necessity carried out with an extremely coarse three-dimensional mesh and would aim at representing the specific measurements made on the SL-1.

c. Effect of Dissolved Boron Poisons on the Rod Worth

This brief study would probably be made in one-dimensional geometry to determine the influence of boron dissolved in the reactor water on the in-to-out reactivity worth of a control rod. The boron concentrations considered should cover the range encountered in the ANL control rod calibration measurements. The results should indicate the correction to be applied to the ANL measurements.

4. Reactivity Effect of Changes in the Fuel Inventory

During the operational history of the SL-1 reactor, a number of changes were made in the fuel inventory. For the most part, the core was operated with 40 fuel elements although some of these fuel elements were changed from time to time. For a brief period the core operated with 41 fuel elements, one of these being an instrumented element. During the fuel inspection of August 1960, some of the boron strips were lost from several of the fuel elements. It would be desirable to make an approximate evaluation of the reactivity implications of these core rearrangements to demonstrate that the shutdown of the reactor was not seriously influenced by any of these alterations.