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CEND-1009

SL-1 ANNUAL OPERATING REPORT,
FEBRUARY 1960-JANUARY 3, 1961

June 15, 1961

Nuclear Division
Combustion Engineering, Inc.
Idaho Falls, Idaho

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CEND-1009

REACTOR TECHNOLOGY

SL-1 ANNUAL OPERATING REPORT

FEBRUARY, 1960 - JANUARY 3, 1961

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

Approved

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ABSTRACT

The Stationary Low Power Reactor No. 1 (SL-1) was a boiling water reactor demonstration plant for remote military bases. The plant had been operated by Combustion Engineering, Inc. from February, 1959 until January 3, 1961 at which time a nuclear excursion rendered the plant inoperable. This report summarizes the operations of maintenance, test, health and safety, and administrative activities for the year February, 1960 until the incident. This report is written in compliance with U. S. Atomic Energy Commission Contract AT(10-1)-967.

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SUMMARY

The Stationary Low Power Reactor No. 1 (SL-1) was a boiling water reactor demonstration plant for application at remote military bases. The plant served as a test and training facility within the Army Reactors Development program. The plant had been in operation at the National Reactor Testing Station from August 1958 until January 3, 1961, at which time a nuclear excursion rendered the reactor inoperable.

This report covers the plant operation, test, and training effort at SL-1 during the period February 1, 1960 through January 3, 1961. During the report period the plant accumulated 5,848 hours at power which was 78% of available operational time. As of December 23, 1960, the plant had accumulated 9,819 hours at power since initial startup, representing an availability factor of 78%. To satisfy test and training requirements, the plant was started up and scrammed 196 times during 1960. Reliability of conventional plant equipment was considered satisfactory in view of the type of operational service to which it was subjected. A total of 16 malfunctions to non-reactor equipment were experienced, requiring plant downtime of 312 hours during the year.

During the report period the aluminum alloy clad core produced 568 MWD(t) of energy. The accumulated energy produced as of December 23, 1960 when the plant was shutdown was 931 MWD(t) or approximately 40% of design core life. An inspection of representative fuel assemblies made in August 1960 after 700 MWD indicated no apparent corrosion or radiation damage to the fuel plates. The inspection did reveal the deterioration of the boron aluminum strips which were attached to the fuel assemblies. The strips from a center positioned fuel assembly in the core had largely disintegrated and strips tack welded to peripheral assemblies were bowed in a wave fashion.

A greater than predicted gain in core reactivity was observed and attributed partially to the apparent loss of boron as well as normal burnup, thus the boron 10 was substantially depleted in the center region of the core. Cadmium shims were inserted in two Tee control rod positions in the core in November 1960, thus increasing the shutdown margin by approximately 1% reactivity.

A series of test runs was conducted at reactor power outputs up to 4.7 MW(t). These runs (reported in IDO 19023) indicated that the reactor, after addition of cadmium shim strips in two Tee rod positions, could only be operated stably at 4.7 MW(t) when the control rods were programmed with the center rod inserted farther in the core than the outer four rods. Routine operation at 3 MW(t) was satisfactory with control rods in a uniform banked position.

Performance of the SL-1 control rod drive mechanisms was erratic during the year with 53 cases of sticking occurring in 2,730 cases of control rod operations, i.e., scrams, rod drop tests etc. The instances of sticking increased significantly during the last two months of plant operation. An analysis of control rod and drive mechanism history indicates that the sticking was due to poor mechanical performance of the mechanisms. Maintenance of the mechanisms, which included periodic inspection and cleaning of the shaft seals and replacement of pinion gear bearings, resulted in elimination of the sticking when initially returned to operation. A replacement set of PL control rod drive mechanisms with substantial design improvement were scheduled for installation in May 1961.

A PL type condenser was designed, manufactured, and installed in a separate loop at the SL-1 during the year and performance testing was begun in October 1960. Results from performance testing of the condenser loop were satisfactory and full steam flow tests (13,000 lbs/hour of 300 psi saturated steam) showed that the condenser performance met the design rating.

A test program aimed at installation of SL-1 Core 2 during 1961, was in progress including inpile corrosion testing of Core 2 control rod materials and performance testing of two alternate shaft seals for replacement control rod drive mechanisms. Planning for installation of a Core 2 test fuel assembly and control rod drive was in progress.

Steam quality remained greater than 99% for reactor power levels up to 4.7 MW(t). For the same power range a decontamination factor between reactor water and steam (based upon Na^{24}) of 1×10^4 to 4×10^4 was observed. Radiation levels in the main steam system continued to be sufficiently low, although increasing in September 1960, by a factor of 3 when a fuel assembly with thermocouples penetrating the clad was installed. Full personnel access to this equipment for operation and maintenance with no over exposure was standard practice.

Quantities of low level liquid, gaseous, and solid wastes resulting from plant operation during 1960 increased significantly over the preceding year, primarily from installation of the instrumented fuel assembly. The increased quantities were easily handled and never precluded safe routine operation.

I OPERATIONS

The operations section of this report contains a summary of the SL-1 design characteristics augmented by the plant expansion data, a chronology of operations, operational and training summary and a list and brief description of all malfunctions.

A. SL-1 DESIGN CHARACTERISTICS

Plant System

Reactor heat output	3 MW (t)
Steam production	9020 lb/hr
Operating pressure	300 psig
Operating temperature	421 °F
Turbine generator output	300 KW (e)
Steam quality	Saturated
Space heating load	400 KW (t)
Ambient temperature	-60 to +60 °F
Air cooled condenser capacity	7.5×10^6 BTU/hr

Reactor Core

Cladding alloy for plates (Al - 1% Ni) ALCOA	X-8001
Length of active core	25.8 in.
Equivalent diameter of active core	31.4 in.
Number of fuel elements	40
Total thickness of fuel plates (0.050 in. meat, 0.035 in. clad)	0.120 in.
Number of plates per element	9
Average water channel gap	0.310 in.
Fuel in 40 elements, U ₂₃₅	14 Kg
Weight of U ₂₃₅ per element (approx)	350 gm
Total weight of B ₁₀ in B-Al strips attached to 40 fuel elements	22.6 gm

Control Rods

Number of cruciform rods	5
Size of cross	14-1/4 in.
Length of cadmium section	32 in.
Thickness - cadmium	0.060 in.
Thickness - Al-1% Ni alloy clad	0.080 in.

Scram Time	2 seconds
Withdrawal rate (approx. max.)	0.01% k/sec.

Nuclear Data

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

Heat Transfer

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

Plant Expansion

Condenser type	Grisscom Russell air cooled
Steam flow (10% moisture)	13,000 lb/hr
Condenser coolant heat transfer area	35,292 sq. ft.
Condenser design heat transfer	12,630,000 BTU/hr
Condenser coolant flow	543 lb/hr
Condenser coolant inlet temperature	60 °F
Number of coolant fans	4

B. SL-1 OPERATION CHRONOLOGY

<u>Date</u>	<u>Operation</u>
2/3/60	Begin seven day shift operation
2/11/60	Replaced #7 control rod drive motor
2/19/60	Shutdown for trainee examination
2/29/60	Renew seven day shift operation
2/29/60 - 3/29/60	(1) Hydrogen buildup test performed (2) Equilibrium xenon measured

(3) Decontamination factor test continued
 (4) Steam Quality Test
 (5) Water Decomposition Test

4/1/60 Rod drop performance test for design evaluation

4/1/60 Shutdown for plant maintenance

4/8/60 Control rod mechanism for rod No. 9 was disassembled, inspected, cleaned, and reassembled

4/9/60 Renew seven day shift operation

4/9/60 - 5/26/60 (1) Decontamination factor test continued
 (2) Fission break monitoring test
 (3) Water decomposition test

5/26/60 Shutdown for decay heat test and plant maintenance.

6/11/60 Commence seven day shift operation for Cadre training

6/25/60 - 6/26/60 Shutdown for NRTS open house

6/29/60 Hot criticals to determine best neutron detector location for startup

7/7/60 Replaced #3 control rod drive magnetic clutch

7/11/60 Shutdown for maintenance
 (1) Checked grid plate bolt tightness
 (2) Inserted unclad test coupons (Ag-In-Cd) and fluxwires
 (3) Inspected and cleaned all control rod drive mechanisms and replaced pinion bearings

7/16/70 Control rod drop tests for design evaluation

7/17/60 Resume power operation

7/31/60 Hot rod drop tests for design evaluation

7/31/60 Shutdown to remove test coupons (Ag-In-Cd) and two fluxwires

8/14/60 - 8/21/60 Shutdown
 (1) Inspect fuel elements, corroded boron side plates discovered and sections of plates from fuel 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

8/21/60 Control rod No. 1 stuck cold; was disassembled, inspected and reassembled, then found sticking above 20 in. Rod withdrawal was then limited to 20 in. for subsequent operation

8/24/60 Resume operation

8/24/60 - 8/25/60 Control rod worth experiments

9/11/60 Rod drop tests for design evaluation

9/11/60 Shutdown to remove fluxwires and test unclad coupons (Ag-In-Cd)

9/12/60 Inspected and cleaned control rod mechanisms and replaced pinion bearings

9/13/60 Cold critical and low power rod bank measurements

9/13/60 No. 5 rod calibrated

9/14/60 Rod bank vs. power measurements

9/16/60 Hot (zeropower and 2.5 MW) critical rod bank measured

9/23/60 Equilibrium xenon (2.5 MW) rod bank position measured

10/3/60 Begin hot checkout of PL loop

10/28/60 72 hour test of PL loop at power

11/1/60 Boiling noise study at 1, 2, and 3 MW

11/2/60 Increase power at 3.5 MW

11/3/60 Increase power to 4.1 MW

11/4/60 Increase power to 4.7 MW

11/5/60 Equilibrium xenon measured

11/6/60 Shutdown for maintenance

11/9/60 (1) Clad test coupons (Ag-In-Cd) installed
(2) Burr on shroud for rod No. 1 smoothed with special tool

11/10/60 Dummy rod No. 4 installed

11/14/60 Seal test on rod No. 4 performed

11/15/60 Inserted cadmium strips in tee slots of rod positions No. 2 and No. 6

11/16/60	Critical rod bank position measured at 180 °F
11/17/60	Resume power operation
11/21/60	3 MW stability test
11/30/60	Stability test continued at various power levels
12/4/60	Equilibrium xenon measurement
12/9/60	Transient test
12/15/60-12/20/60	Continuous operation at 4.7 MW
12/21/60	PL condenser performance test
12/23/60	Equilibrium xenon 2.56 MW critical rod bank position measured
12/23/60	Shutdown for maintenance

C. OPERATIONAL DATA SUMMARY

A summary of SL-1 operational data to January 3, 1961 is tabulated below:

	Annual Plant Totals	Cumulative Plant Totals
Hours plant available to operate	7,512	12,832
Hours plant operated (1)	6,585	10,986
Hours of power (2)	5,848	9,819
Reactor MWD (t) (3)	568	931
Turbine generator KW (e) (4)	1,131,100	1,937,200
Simulated heat load KWH (t) (5)	1,694,400	3,109,500
PL condenser steam flow 1,000 lbs.		2,985
Fuel burnup (grams) (6)	715	1,173
Percent design core life expended (7)		40
Percent hours plant operated	88	86
Percent hours plant at power	78	78

Notes:

- (1) Total time the plant was operated including checkout, startup and securing time.
- (2) Total time the reactor was critical.
- (3) Total of hourly reactor power calculations.
- (4) From KWH meter.

- (5) Total of hourly space heat output calculations.
- (6) For U²³⁵ only.
- (7) Based on 2300 MWD design lifetime.

D. SL-1 TRAINING SUMMARY

The SL-1 plant is used as a training facility by military personnel. Trainees are assigned to each operating crew for on-the-job experience in reactor operation and maintenance prior to qualifying as operators or chief operators.

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

Training activities are performed through the Idaho Nuclear Power Field Office as part of the Army Nuclear Power Program.

Totals in SL-1 Operator Training

	1960	Cumulative
Army	45	70
Navy	7	12
Air Force	5	12
Maritime	8	26
Combustion Engineering, Inc.	<u>2</u>	<u>5</u>
Total	67	125

Qualified SL-1 Operations Personnel

	1960	Cumulative
Chief reactor operators	8	19
Reactor operators	20	39

Special ratings of trainees

Electrical	3	10
Instrument	6	15
Mechanical	9	21
Health Physics	2	5
Other	43	67

E. MALFUNCTIONS

A total of 38 malfunctions occurred during the period from February, 1959 to December 23, 1960, at SL-1 resulting in a total reactor down time of 885 hours. Of the 38 malfunctions 16 occurred during this report period resulting in 312 hours of reactor down time. The following is a summary of malfunctions and the actual hours of resultant reactor down time:

<u>Date</u>	<u>Malfunction Number</u>	<u>Type</u>	<u>Resultant Reactor Downtime</u>
3/12/60	23	Incorrect test instrument installation	55 min
4/20/60	24	Failure of Reactor Water Purification Pump	25 hrs
5/1/60	25	Station Auxiliary Breaker tripped	2 hrs 17 min
5/8/60	26	Station Auxiliary Breaker tripped	1 hr 5 min
6/3/60	27	Punctured screen in Purification System mixed bed resin column	162 hrs 42 min*
6/20/60	28	Loss of power to scram channel caused by improper training exercise	10 min
7/16/60	29	Steam leaks in primary piping following maintenance period	18 hrs 1 min
7/16/60	30	Tube failure in power supply Channel I	33 min
7/24/60	31	Steam leaks in primary piping following maintenance period	23 hrs 40 min
7/28/60	32	Failure of Reactor Water Level Transducer caused by high ambient temperature	39 min
8/22/60	33	Steam leak in main steam piping original welds	33 hrs 15 min
8/28/60	34	Accidental addition of resin to primary system	39 hrs 13 min

<u>Date</u>	<u>Malfunction Number</u>	<u>Type</u>	<u>Resultant Reactor Downtime</u>
10/25/60	35	Spurious trip of Motor Control Center Supply circuit breaker	1 hr
12/3/60	36	Slippage of motor drive shaft attached to main condenser exhaust dampers	5 min
12/7/60	37	Condensate circulating pump failure	2 hrs 23 min
12/16/60	38	Failure of motor control center supply circuit breaker caused by improper mounting and loose connections	2 min
TOTALS			311 hrs 0 min

* Eighty hours of actual maintenance were required for the correction of this malfunction; the remainder was scheduled downtime.

A brief description of each malfunction and corrective action during the report period follows:

No. 23 - The reactor scrambled during normal operation from a false high water level signal. Investigation showed that a helipot used to provide an electrical indication of water level to a test instrument had erroneously been installed on a shaft with an eccentric movement. Corrective action was removal of the helipot.

No. 24 - The purification pump is a canned rotor pump employing modified automotive type spark plugs to provide a pressure seal for motor leads. The plugs were removed during an overhaul and were improperly reinstalled providing an inadequate electrical connection which, after limited operation, caused the pump motor failure. Failure of this pump required securing the plant to prevent water quality degradation. Corrective action included repiping purification system to permit water flow without pumping during reactor operation and replacement of the spark plug type of pump motor seals with a new type.

No. 25 - Tripping of the station auxiliary breaker and failure of reset caused a reactor scram by high condenser pressure because of loss of coolant fan. The breaker is rated to trip at 400 amps and 25 °C. The temperature of the breaker at the time of tripping was 47 °C with a load

of 260 amps. Initial action was to provide better ventilation of breaker by removing cover. Additional overload heaters were ordered to trip at 50 °C at the rated current.

No. 26 - Tripping of the station auxiliary breaker and failure of reset caused a reactor scram by high condenser pressure because of loss of coolant fan. The breaker is rated to trip at 400 amps and 25 °C. The temperature of the breaker at the time of tripping was 47 °C with a load of 260 amps. Initial action was to provide better ventilation of breaker by removing cover. Additional overload heaters were ordered to trip at 50 °C at the rated current.

No. 27 - The lower screen in the mixed bed resin container ruptured. It was discovered when resin from the mixed bed column plugged the feedwater filter. Corrective action consisted of shortening the tool used to facilitate resin addition, instruction of operators in the use of the revised tool and replacing the punctured screen.

No. 28 - During a military training period, the steam supply was reduced to the turbine and caused a decreased power output from the generator. Fuses blew in a voltage regulator and power supply to Nuclear Channels I and IV scrambling reactor. Corrective action consisted in revising method of scrambling reactor for training purposes, evaluating protective devices and replacing fuses.

No. 29 - Steam was visually observed blowing onto the operating floor from under the reactor top shielding during a reactor startup. The reactor was manually scrammed. Leaks were found on control rod inlet and outlet cooling water fittings. Corrective action included tightening or replacement of all fittings and revision of reactor startup procedures to include inspection of water fittings prior to startup and inspection of the reactor top area for water and steam leaks when the reactor reaches operating temperature and pressure.

No. 30 - During startup, the reactor was scrammed on five separate occasions from abnormal operation of Nuclear Channel I. Corrective action included installing Channel IV in the scram circuit replacing Channel I. A vacuum rectifier tube in the Channel I power supply was found to be defective.

No. 31 - Steam was observed blowing onto the operating floor from under the reactor top shielding during a startup. The reactor was manually scrammed. Steam leaks were found to originate in No. 3, 7, and 9 control rod drive outboard shaft seals. Replacement of the neoprene shaft seals which had hardened was required.

No. 32 - A reactor scram caused by high water level occurred. A transformer ambient coil in the reactor water level controller was found to have failed due to excessive temperature. The coil was replaced.

No. 33 - Radioactive steam was released to the operating floor raising the air activity to several times the normal tolerance. The reactor was manually scrammed. The steam leaks were found to originate from three original welds in the main steam system. Two water leaks were found in screwed fittings on the purification shutdown cooler. The welds and screwed fittings were repaired by using heliarc welding methods.

No. 34 - A large quantity of used resin was added to the hotwell by an undetermined method after an extended shutdown. The reactor was manually scrammed. The problem was first detected when the feedwater pumps did not show proper discharge pressure. Corrective action included removing all pump screens, draining and flushing the hotwell several times and all filters in the reactor system were changed or cleaned.

No. 35 - The reactor was scrammed by main condenser overpressure. Condenser overpressure was caused by loss of power to the main condenser cooling air fan. Loss of power occurred when the fan motor main breaker tripped from overload. No reason for the overload trip was discovered. Action taken consisted of continuous monitoring of amperage of the fan motor to enable evaluation of the malfunction.

No. 36 - The automatically controlled main air condenser mixer and exhaust dampers slipped on the motor drive shaft and shut causing a reactor scram from condenser overpressure. The dampers were modified to provide a positive lock to the motor shaft by pinning.

No. 37 - The condensate circulating pump motor was found to be overheated and shorted out before corrective action could be taken. The plant was manually scrammed. The damaged pump (canned rotor type) graphite bearings were found to be excessively worn causing increased friction and the motor burnout. Manufacturer's recommended preventive maintenance for these bearings is each 1500 hours; however, this period had been extended to 3000 hours locally, based on conditions found during previous inspections. This pump failure at 2900 hours indicated a reschedule to every 2000 hours would be satisfactory. To provide back up, a second pump was installed in parallel in the condensate system.

No. 38 - The utility bus breaker supplying power to the motor control center tripped, causing loss of the SL-1 condenser fan. Before steam flow could be reduced, the SL-1 condenser became pressurized and the reactor scrammed. The equipment and utility breakers were found to have been improperly mounted and their bus connections loose, discolored and corroded. The breakers were bench tested with no indications of malfunction. The breakers were thoroughly cleaned and re-installed to allow plant operation.

II SL-1 MAINTENANCE

During 1960 an effort was made to extend plant operational periods by performing general preventive maintenance without shutting down the plant. As previously reported in IDO 19012 maintenance of some SL-1 components which had no backup units required the plant to be shutdown periodically. Corrective action taken during 1960, i.e., addition of backup units, design modification, or change in procedure which permitted longer periods without maintenance or permitted maintenance during routine plant operation are discussed in the next section.

A. CORRECTIVE ACTION TO REDUCE MAINTENANCE DOWNTIME

1. 1500 Hour Maintenance of Canned Rotor Pumps.

The manufacturer's recommendations for maintenance of four canned rotor pumps installed in the SL-1 plant was for inspection and replacement of bearings after each 1500 hours of operations. This maintenance job required approximately 40 man-hours with the plant shut down. To eliminate this shutdown requirement, the following steps were taken:

a. A backup condensate circulating pump was purchased and permanently installed parallel with the first pump, thereby allowing switch over and maintenance of one pump each 1500 hours while the other pump allowed continuous plant operation.

b. Although the purification system circulating pump could be shut down for short periods it could not be removed from the system during plant operation because of the requirement for reactor water purification. To eliminate this maintenance problem a spare pump was purchased. The purification system was revalved to allow the purification pump to be removed during operation and replaced with a spare.

The third and fourth canned rotor pumps installed in the plant without back-up were used for high and low pressure drain return to the hotwell. A system study revealed that these pumps could be valved out and removed from the system for maintenance during plant operation while drain water was returned via float type traps to the plant system.

2. Provision of Additional Nuclear Scram Channel

Since only two nuclear power scram channels were provided on SL-1, failure of either during operation resulted in plant shutdown. To eliminate the shutdown required for maintenance of either nuclear read out instrument and still provide two power scram channels, a scram contact was installed on the

channel IV linear level recorder and captive key actuated bypass switches were installed on channel I and II. It was, therefore, possible to maintain either of the power scram channels during operation while channel IV served as the second scram channel during the short outage.

3. Turbine Lube Oil Cleaning or Replacement

No provision was made in the SL-1 plant design for periodic or continuous cleaning of the turbine generator lubricating oil. To reduce the requirement for securing the plant to replace the oil, a procedure was adopted for taking samples of the oil for periodic testing for water and dirt. As long as the impurity limits were not exceeded no shutdown for oil replacement was made.

4. Selection of High Grade Turbine Governor Oil

Oil available through normal supply channels was found to require replacement in the Woodward turbine governor following 500 to 1000 hours operation. This maintenance operation required securing the turbine-generator for up to four hours for replacement. To reduce this maintenance problem a survey of the oil suppliers and users was made with the resulting recommendation that Texaco Regal Type RO oil be used. The oil was obtained, installed, and satisfactory operation of the governor in excess of 2500 hours without oil change was experienced.

B. MAINTENANCE OF REACTOR AND PLANT EQUIPMENT THAT REQUIRED PLANT SHUTDOWN

Maintenance of various reactor and electric components still required the SL-1 plant to be secured, the major items in this category being the reactor water level indicators and control rod drive mechanisms. The maintenance and inspection of control rod drives and reactor water level indicating devices were generally scheduled to coincide with reactor shutdown for operational reasons, training, and installation of test equipment. The 1960 Maintenance History for the control rods and drives is presented in Table 1. Inspection, maintenance, and calibration of the water level indicating devices was accomplished in January and July of 1960 during shutdown periods. This method of scheduling maintenance, while generally satisfactory, resulted in some conflicts between testing, training, and maintenance requirements.

C. ROUTINE MAINTENANCE OF PLANT EQUIPMENT

The routine maintenance, i.e., replacing electron tubes, lubricating equipment, maintaining recorder ink and charts, repacking pumps and valves, and operational inspection of equipment was carried out in 1960 during plant operation. The maintenance scheduling system outlined in the Appendix was used effectively for routine maintenance scheduling, restocking of parts used, and recording of work performed.

TABLE 1

1960 MAINTENANCE HISTORY ON SL-1 CONTROL ROD DRIVE MECHANISMS

<u>Date</u>	<u>Rod #1</u>	<u>Rod #3</u>	<u>Rod #5</u>	<u>Rod #7</u>	<u>Rod #9</u>	<u>Source</u>
Jan., 60	AB	AB	AB	AB	AB	Maintenance Log
Feb. 11, 60				C		Maintenance Log
Apr. 4, 60					A	Maintenance Log
July 5, 60					D	Maintenance Log
July 7, 60		E				Maintenance Log
July 13, 60	AB	AB	AB	AB	AB	Maintenance Log
Aug. 15, 60		A				Maintenance Log
Aug. 21, 60	A					Power History Log
Aug. 26, 60					A	Maintenance Log
Sept. 13, 60	AB	AB	AB	AB	AB	Maintenance Log
Nov. 11, 60	F					Maintenance Log

LEGEND OF MAINTENANCE FUNCTIONS

- A. Inspected and cleaned mechanism.
- B. Installed new set of bearings adjacent to pinion gear.
- C. Replaced drive motor.
- D. Replaced shear pin.
- E. Replaced magnetic clutch.
- F. Removed burr from top edge of control rod shroud.

D. OPERATIONAL TESTING OF PLANT SAFETY DEVICES

During 1960 provisions were made for testing plant safety devices in conjunction with plant operations.

Plant safety devices so tested included relief valves, turbine lube oil pressure safety devices, turbine overspeed trips, etc. Control rod scram devices were adequately tested during plant startup or shutdown. During extended periods of operation, the plant safety devices were checked by use of bypass key switches for each device. A definite schedule and conditions for testing these devices were established and followed.

E. PLANT IMPROVEMENTS

Certain modifications to plant equipment were carried out to provide more reliable operation. A list of the modifications to SL-1 plant equipment accomplished during 1960 follows:

1. Nuclear instrument cable insulation was changed from low temperature (polyethylene) to high temperature (teflon) material to eliminate local hot spots causing cable failure.
2. Turbine vacuum-recorder transducer was relocated to eliminate a signal error included by a water log.
3. Turbine horizontal casing seal was regasketed with Johns Mansville asbestos rope packing and Permatex.
4. An additional sample tap was added to the main steam line approximately 20 feet downstream from the reactor to allow simultaneous sampling for steam quality and dissolved gases.
5. The turbine generator governor linkage was adjusted to allow adequate steam flow at full power operation.
6. The station auxiliary circuit breaker was equipped with a 50° C temperature sensitive trip mechanism to replace a trip mechanism rated at 25° C.
7. The BF₃ chambers in nuclear channels V and VI were lowered six inches for improved neutron detection because of critical rod bank position changes.
8. Additional ventilation and cooling was provided for the water level transmitter.

9. An alarm system, consisting of a bell and light wired across the purification pump power supply, was installed.

10. A steam sample cooler was installed.

11. To aid operation of the plant expansion loop the following changes in the original SL-1 piping were made:

a. Installation of two check valves in the gland seal water piping to the feedwater pump glands to prevent backflow.

b. Tie-in of plant expansion non-condensable vent line to SL-1 vent line to permit monitoring of radioactive non-condensibles released from the reactor.

13. A communications system between the plant expansion condenser building and control room was installed.

14. A 15 HP feedwater pump motor was replaced with a 25 HP motor to handle higher feedwater rates at increased operating power levels.

15. Flexible piping was installed in place of rigid control rod seal cooling tubing to allow convenient quick and reliable attachment of these lines during frequent assembly and disassembly.

16. A temporary liquid waste disposal system was installed utilizing the plant expansion drain pump. A permanent system had been designed and funds approved.

17. The main condenser exhaust louvers were permanently pinned to the motor drive shaft.

18. An identical second condensate pump was installed in parallel with the primary condensate pump.

19. A harmonic drive (sealless-zero leakage) test valve was installed in an air ejector steam line.

20. The non-condensable vent line was electrically traced and insulated to prevent freeze-up during cold weather.

21. A thermocouple was installed in the main feedwater line.

III SL-1 CORE I PERFORMANCE

The SL-1 Core I was inspected during this report period. Physics measurements were made periodically. Additional poison in the form of cadmium strips was added to several tee slot positions to increase reactor shutdown margin lost by corrosion of the burnable boron side plates. Hot cell examinations of sections of corroded boron strips and an irradiated fuel element provided preliminary information on Core I metallurgy and mechanical stability. Insertion of an instrumented fuel assembly gave temperature data to aid in evaluating core metallurgy. A core neutron flux map was made in support of core physics.

A. PHYSICS

1. 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 I. 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 °F. Water temperature was measured by thermocouples in an instrumented fuel assembly. As the No. 5 rod 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 1.

The other calibration of No. 5 rod quoted in CEND 1005 was carried out on August 25, 1960 with control rod No. 7 fully inserted. A water temperature of 155 °F was recorded at the inlet to the purification system. The differential worth curve for No. 5 rod obtained in this calibration showed an unusually high peak and a consequently high worth of 2.5%. The high worth obtained here for rod No. 5 is probably due primarily to the complete insertion of No. 7 rod. 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 reported in CEND 1005 (August 25, 1960) was carried out only for the withdrawal range of 0 to 12 inches of No. 9 with the 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

INTEGRAL ROD WORTH CURVES

FIGURE 1

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 fourtimes 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 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.

2. Core Fuel Loading History

All but three of the fuel elements which were in the reactor at the start of Combustion Engineering, Inc. operating have been in the positions shown for the entire 932 MWD. Figure 2 shows the arrangement of fuel assemblies as of February 3, 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 Combustion Engineering, Inc. 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.

a. September 23, 1959 (213 MWD)

(1) Fuel element No. 6 was moved from position 45 to position 87, which was previously uncoupled.

(2) Instrumented fuel element No. 63 was placed in position 45.

b. October 7, 1959 (229 MWD)

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

LEGEND-

- FULL STRIP OF BORON
- HALF STRIP OF BORON

1. SMALL NUMBERS DENOTE BORON PLATE NUMBERS
2. NUMBERS LOCATED IN UPPER RIGHT HAND CORNER DENOTE GRID POSITION
3. NUMBERS IN CENTER DENOTE FUEL ELEMENT NUMBER.

SL-I CORE I FUEL LOADING
FIGURE 2

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

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

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

(5) Fuel element No. 62 was placed in position 66. This was a new element with:

- (a) Full boron strip No. 73 containing 0.41 gm B^{10}
- (b) Half boron strip No. 75A containing 0.19 gm B^{10}

c. August 21, 1960 (680 MWD)

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

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

(3) Fuel elements Nos. 19 and 59 were removed from the position 85 and 75 respectively, examined, and returned to their position.

Thus, in Figure 2, elements 38 and 42 have been replaced by elements No. 1 and 62 respectively. These latter two elements have been in the reactor for 253 and 703 MWD of operation respectively.

3. 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 only 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 were banked, and to insure that the reactor was at the desired power level and xenon condition. These measurements were made periodically during the SL-1 operation (See Figure 3). The second category includes rod positions taken on a routine basis (once each shift) by the operating crew, and recorded in the operations log. Figure 4 shows some of these points corrected to a bank, and corrected for core vapor content but not for xenon history. It will be noted that the general trend indicated by the few points in Figure 3 is borne out by the data in Figure 4.

4. Flux Maps of Core

Flux maps were prepared by irradiation of cobalt-aluminum alloy wires at a known power-time history. Radial and axial plots are presented in Figure 5 and 6 respectively.

CRITICAL ROD BANK POSITION WITH EQUILIBRIUM XENON CONCENTRATION AT 2.56 MW (τ)

FIGURE 3

HOT OPERATING (8000 lbs/hr STEAM FLOW AND EQUILIBRIUM XENON) CRITICAL ROD BANK POSITIONS IN SL-1 VS. MEGAWATT DAYS OF OPERATION

Fig. 4

CURVE NO. 2
CWL/SKH
DEC. 14, 1960
CKD *CKD*

FIGURE 5

5. Instrumented Fuel Assembly

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, 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 Cambridge, England 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 drop. No additional ΔT was measured which could be attributed to the buildup of a corrosion film.

The first instrumented (thermocouples) fuel assembly test 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 No. 38 were placed in the SL-1 storage well during the October 3, 1959 shutdown.

A second fuel assembly was instrumented with thermocouples. 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 and thoroughly cleaned with acetone, rinsed with distilled water and dried.

Magnesium-oxide insulated chromel-alumel thermocouples, 25 feet long were used for the instrumentation. They were clad with .066 inch O.D. Type 347 stainless steel tubing with 0.008 inch 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 600 °F and 50 °F intervals.

Eight of the above thermocouples were inserted in the fuel plates. 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 such 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 was removed and the hole brazed shut when an unreparable sheath crack occurred during fabrication.

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 the rigidity and ease of routing in the reactor. A photograph of the test assembly is shown in Figure 7.

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 Nos. 6, 7, and 8 were not usable. Fuel plate temperature data are plotted in Figures 8 and 9.

B. METALLURGY

1. Fuel Elements

The SL-1 fuel plates were fabricated by standard roll bonding techniques modified to include silicon bonding. This process was developed by Argonne National Laboratories 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 10 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 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 1000 °F. Fuel plates were hot rolled in air at 980 °F to three 25 percent 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.

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 for visual inspection:

INSTRUMENTED FUEL ASSEMBLY

FIGURE 7

INSTRUMENTED FUEL ASSEMBLY PLATE TEMPERATURES
REACTOR AT 3MW

Fig. 8

THE EFFECT OF REACTOR POWER
ON SL-1 MAXIMUM FUEL PLATE
TEMPERATURE

Fig . 9

<u>Fuel Element No.</u>	<u>Position No.</u>
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. 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 by placing them so that they could be viewed through control rod ports. Because of the severity of the aluminum-boron strip deterioration visual inspection was terminated after three fuel assemblies had been viewed. The fuel plates in the assemblies observed (Nos. 19, 59, and 42), did not show any sign of warpage, swelling, distortion or corrosion. It was concluded that corrosion of film buildup had not changed noticeably since the last inspection. These visual observations indicating no corrosion film buildup corroborate maximum 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, was removed to the AREA hot cell facility. Prior to the incident, visual examinations, 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. 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.

2. 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 B¹⁰, or 0.4 w/o B¹⁰ (for full and half strips, respectively) were tack welded to the sides of fuel assemblies (93% enriched B¹⁰ 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 by extruding, at 850 °F, a billet containing a powder mixture of X-8001 aluminum and B¹⁰ powder, clad in X-8001 sheet. The extruded sections were cold rolled to final

thickness, annealed, cut and then tack welded to the side plates of the fuel assemblies. Although it was assumed that these strips would still be clad with X-8001 after extrusion, metallographic examination of sections of an unused strip, as well as sections of recovered aluminum-boron strips, do not have complete cladding. It is certain that the edges and ends were unclad.

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 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. Figure 10 visual inspection was stopped after removal of assembly 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 fragments recovered from the bottom of the core are shown in Figures 11 and 12.

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 (Figure 2). 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:

a. Thickness traverses are being made of poison strip remnants to obtain data on possible swelling.

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

c. Metallographic samples are being taken from high and low burnup regions in search of indications, such as gas generation around borides and local corrosion.

SL-1 FUEL ASSEMBLY No. 19 - SIDE VIEW - AUG. 1960
Fig. 10

BORON STRIPS AND DEBRIS RECOVERED FROM
REACTOR VESSEL WITH SPECIAL TOOLS

FIGURE II

OUTER SURFACE OF PLATES NO.42 AND NO.35

FIGURE 12

d. 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.

e. B^{10} burnups are being established by isotopic analysis for areas adjacent to samples in c. and d. 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.

3. Cadmium Shims

The inferred loss of boron poison was to be compensated 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.

A modified picture frame technique was used in which three cadmium strips 0.02 in. x 4-14/16 in. x 29 in., were placed adjacent to 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 1/16 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.

4. 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 in a dummy fuel element which was inserted into position No. 27 in July, 1961. Three of these specimens were in the reactor at the time of the incident. The samples and their holders are shown in Figure 13.

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 lifting studs 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 (Figure 14).

AG, IN, CD, CLAD SPECIMENS
FIGURE 13

AG-1N CD UNCLAD SPECIMENS
FIGURE 14

IV CONTROL ROD AND DRIVE MECHANISM

A. DESIGN DESCRIPTION

The core structure of the SL-1 was designed to accommodate fifty-nine fuel assemblies and nine control rods. During initial testing it was found that only forty fuel assemblies and thus only five control rods were required. Only five have been utilized since Fall of 1958.

The five control rods are composed of cadmium sheets with aluminum-nickel cladding (Figure 11). In November, 1960 a sixth control rod channel was used for testing a rod drive mechanism which used an all aluminum rod having no shutdown value.

The cadmium sections of the control rods extend 3-1/8 inches below the nominal upper fuel dimension when the control rods are in their zero position. Stainless steel ball-joint end fittings are riveted to the upper ends of the control rods. These are used to connect the control rod to the rod drive mechanism by means of a ball-joint gripper located at the lower end of the rod drive extension shaft (Figure 11). 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 (Figure 15).

Vertical linear motion is imparted to the rod by the rack and pinion gearing. 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 (Figure 16).

A rotary shaft pressure seal is used where the pinion drive shaft penetrates the mechanism housing. The pressure seal is of the positive clearance, breakdown type seal, which has controlled leakage (Figure 16). Water provides cooling for the seals and 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 require approximately .01 gpm of water bled continuously from each control rod drive housing.

The control rod drive motor and position indicator assembly (Figures 15 and 17) 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 magnetic clutch (Figure 13).

ROD DRIVE HOUSING

CONTROL ROD DRIVE

FIGURE 15

CONTROL ROD DRIVE

FIGURE 16

CONTROL ROD DRIVE CLUTCH ASSEMBLY
FIGURE 18

Failure of the clutch current automatically results in insertion of the rods rapidly 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 over-riding clutch which free-wheels on a rod withdrawal but engages when the rod is moving in. In the event of a 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 limit switches. This synchro arrangement assures the operator of positive rod position indication at all times during operation. The limit switches (Figure 17) are used to operate the upper and lower limit switches, control panel indicating lights, and electric motor interlocks.

The control rod drive mechanism and the pressure breakdown seals were designed and developed by Argonne National Laboratory and Alco Products, Inc. Tests on a lead mechanism were conducted by Alco and ANL to successfully demonstrate design performance. Over 8000 cycles and 250 scrams were made after which time visual inspection indicated satisfactory performance.

1. Disassembly and Assembly Procedures

The Nuclear Power Plant Operator's 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 from this document is given below with only figure numbers being 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.

"Disassembly and Assembly of Components

1. Seal Disassembly (Reference Figure 16)

- 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 18)

1. Remove motor from base.

"Removal of Motor and Clutch Assembly (Reference Figure 18)

1. Disconnect electrical connection (#2) to isolate unit electrically.
2. Loosen two set screws (#2) and slide coupling off spline.
3. Remove four holddown 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 16)

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 (Figure 19) on top of rack (9) and raise rod not more than four 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 fuel element 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 (15). Lift off spring housing and place on clean blotter paper.
14. Place special tool CRT #2 (Figures 20 and 21) 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.

"Installation of Control Rod Drive

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 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.

SPECIAL TOOL CRT NO. 1

FIGURE 19

- "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 dis-assembly. 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 18)

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 #t1 and replace socket head cap screws (53) - (54) .

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

B. OPERATING HISTORY OF SL-1 CONTROL ROD DRIVES

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 system resulting in failure to meet the drop time requirements (one second for 10 inch drop; two seconds for a 30 inch drop) and which did not require a power assist from the drive assembly.

Type II

Sticking of a control rod system 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 system 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 - ALPR, dated December, 1959 (during ANL operation of the reactor) and forwarded to CEND in June, 1960, in which difficulty in meeting the scram time requirements was 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 SL-1 rod sticking incidents ^{1/} that took place from February 5, 1959 to December 23, 1960, is shown in Table II. It includes all rod sticking incidents experienced in rod withdrawals, rod drop tests, rod exercising and rod scrams. Figure 22 presents the instances of all control rod operations with and without sticking for the same period.

^{1/} Taken from the SL-1 Operating and History Logs.

LEGEND

- - INSPECTED AND CLEANED SHAFT SEALS
- ⊕ - INSPECTED AND CLEANED SHAFT SEALS ON ROD 9 ONLY
- - REPLACED PINION BEARINGS
- ▼ - REPLACED MECHANISM ON ROD 7
- - No. OF ROD MOVEMENTS WITH STICKING
- - No. OF ROD MOVEMENTS WITHOUT STICKING

NUMERICAL SUMMARY OF SL-1 REACTOR CONTROL ROD MOVEMENTS FOR PERIOD
MARCH, 1959 THROUGH DECEMBER, 1960

Figure 2 2

TABLE II

SL-1 Control Rod Operational History

Date	Reactor Condition	Rod Operationn	Rod No. 1			Type of Sticking	Action Taken
			Drop Time (sec)	Drop Height (in)	Hang up Location (in)		
7/26/59	162°F Cold	Rod Withdrawal				I	Retested O.K., 30" drop in .9 secs.
9/4/59	Hot-300 psi	RDT*	2.05	30		I	Retested O.K., 30" drop 1.05 sec. Sept. 9
10/9/59	Hot-300 psi	RDT		20	20	II	Retested O.K. 10" drop in .45 sec.
10/31/69	Hot-300 psi	RDT	6.6	30	17	I	Removed negator spring, retested O.K. 30" drop 1.45 sec.
7/5/60	Hot-300 psi	Scram		18.6	17.8	II	Retested; stuck at 29 from 29.9 drop; removed negator spring Retested; stuck at 1" Retested; stuck at 30". Replaced negator spring and retested O.K. 2 times.
7/11/60	Hot-300 psi	Scram			17	II	Retested O.K. from 10" drop - July 25
8/21/60	112°F Cold	Rod Withdrawal			14.3	III	Retested; stuck at 15" Control rod dis-assembled. Sticking in shroud. Retested again and drops freely from 20"
8/23/60	Cold	RDT Withdrawal			22.4	II	Retested O.K. 20" drop in .85 sec.
9/3/60	Hot-300 psi	1" Rod Exercise		18.2		II	Exercise O.K. at 17.8" 4 Sept; limited to 20" withdrawal Sept. 7

SL-1 Control Rod Operational History

Rod No. 1 (Cont)						
Date	Reactor Condition	Rod Operation	Drop Time (Sec)	Drop Height (in)	Hang up Location (in)	Type of Sticking Action Taken
9/11/60	Hot-300 psi	*RDT		10	10	III Retested O.K. 10" drop in .4 sec. Sept. 13
10/17/60	Hot-300 psi	Rod Exercise		18	16	II Driven back to 18" dropped clean on Oct. 19; burr removed from shroud Nov. 11 to permit full rod withdrawal.
11/20/60	Hot-300 psi	Scram		22		I Retested O.K. dropped clear from 19.6 Nov. 21
11/27/60	Hot-300 psi	Scram		20	20	II Retested O.K. at 10"
11/28/60	Hot-300 psi	Scram		17	15	II Retested O.K.
11/28/60	Hot-300 psi	RDT		30	30	II Retested O.K. from 17" 29 Nov.
12/19/60	Hot-300 psi	Rod Withdrawal		20	20-28	III Used pipe wrench to assist clutch
12/19/60	Hot-300 psi	Individual Rod Scram		25	25	II Retest: Failed to drop from 16" finally dropped in 19 sec. Failed second retest from 30"; retested O.K. from 30" in 1.13 sec.
12/20/60	Hot-300 psi	Rod Exercise (Withdrawal)		25.2	25.2	II Retested O.K. from 30" Dec. 21
12/22/60	Hot-300 psi	Rod Exercise (Full)		19.6	23	II Failed to drop from 19.4" 12/23/60
12/23/60	Hot-300 psi	Scram		19.4	19.4	II Plant shut down

*RDT = Rod Drop Test

Type I - Required no power assist
 Type II - Required normal power assist
 Type III - Failed to operate with normal power assist

SL-1 Control Rod Operational History

Date	Reactor Condition	Rod Operation	Rod No. 3		Hang up Location (in)	Type of Sticking	Action Taken
			Drop Time (Sec)	Drop Height (in)			
8/10/59	Hot-300 psi	Scram	1	20	1	II	Retested O.K. from 10" 11 August
9/4/59	Hot-300 psi	RDT	2.75	30		I	Retested O.K. from 30" in .96 sec. 9 Sept.
9/12/59	Hot-300 psi	RDT	6.55	27		I	Retested O.K. from 30" in .59 sec. Sept. 14
9/14/59	Hot-300 psi	RDT	5.6	30		I	Removed 1 negator spring; retested O.K. from both 25 & 30" Sept. 15
10/3/59	Hot-300 psi	Scram	3.0	24		I	Retested O.K. 10" drop in .35 sec. Oct. 9
10/30/59	Hot-300 psi	RDT	6.7	30		I	Removed negator spring; retested O.K. from 30" drop in .71 sec.
7/5/60	Hot-300 psi	Scram		18.6	17.8	II	Retested O.K. Dropped from 30" in .71 sec.
7/11/60	Hot-300 psi	Scram		17.6	17	II	Removed negator spring; retested O.K. from 10" drop July 15
8/14/60	Hot-300 psi	Scram	2.8	17.4		I	Retested O.K. 20" drop in .59 sec. 21 Aug.
12/19/60	Hot-300 psi	Individual Rod Scram		25	25	II	Retested, hung up at 3". Retested O.K. dropped from 16" in 1.22 sec.
12/23/60	Hot-300 psi	Individual Rod Scram		19.4	19	II	Plant shut down

SL-1 Control Rod Operational History

Date	Reactor Condition	Rod Operation	Rod No. 5		Hang up Location (in)	Type of Sticking	Action Taken
			Drop Time (sec)	Drop Height (in)			
4/24/59	Hot-300 psi	RDT		17.3	1.8	II	Retest from 10", drop time .8 sec.
8/10/59	Hot-300 psi	Scram	1.2	20	1.4	II	Retested O.K. from 10" 11 August.
9/4/59	Hot-300 psi	RDT	2.06	30		I	Retested O.K. 30" drop in 1.35 sec. Sept. 9
10/3/59	Hot-300 psi	Scram	3.0	24		I	Retested O.K. 10" drop in .38 sec. Oct. 9
10/30/59	Hot-300 psi	RDT	6.7	30		I	Removed negator spring; retested O.K. 30" drop in .71 sec.
7/11/60	Hot-300 psi	Scram		17.6	17	II	Retested O.K. from 10" drop - July 15
7/16/60	Cold	Rod Withdrawal		0	0	III	Replaced aluminum shear key; rod operated O.K.
8/22/60	Cold	Rod Withdrawal		8	8	III	Replaced aluminum shear key. Rod operated O.K.
9/11/60	Hot-300 psi	RDT		10	10	II	Cooling flow was 180 gph; dropped freely when cooling was secured; retested O.K. drop from 10" in .45 sec. Sept. 13
11/15/60	Hot-300 psi	Rod Withdrawal		16.5	16.5	III	Drove out freely 3 hours later
11/19/60	Hot-300 psi	Rod Exercise (Withdrawal)		19.4		II	Rod operated O.K. in latter rod banking operation.

SL-1 Control Rod Operational History

Date	Reactor Condition	Rod Operation	Rod No. 5 (cont)			Type of Sticking	Action Taken
			Drop Time (Sec)	Drop Height (in)	Hang up Location (in)		
10/20/60	Hot-300 psi	Scram		22		I	Retested O.K. dropped clean from 19.6 Nov. 21
12/3/60	Hot-300 psi	Rod Withdrawal		5.5	5.5	III	Clutch assisted by hand O.K. at rod exercise at 19.6
12/17/60	Hot-300 psi	Rod Exercise (Withdrawal)		20	19	III	Clutch assisted by hand. Retested O.K. 30" drop in 1.71 sec. Dec. 20
12/19/60	Hot-300 psi	Rod Withdrawal		20	20-28	III	Clutch assisted using a pipe wrench.
12/19/60	Hot-300 psi	Individual Rod Scram		25	25	II	Retested O.K. dropped from 16" in .52 sec.
12/22/60	Hot-300 psi	Rod Exercise (Full)		30	26.7	II	Dropped clean from approx. 19.5" in 82 sec. on 12/23/60

SL-1 Control Rod Operational History

Rod No. 7							
Date	Reactor Condition	Rod Operation	Drop Time (sec)	Drop Height (in)	Hang up Location (in)	Type of Sticking	Action Taken
4/24/59	Hot-300 psi	RDT		17.3	2.5	II	Retest from 10", drop time .8 sec.
4/25/59	Hot-300 psi	Scram		19.4	5	II	Retested O.K. April 27
5/1/59	Hot-300 psi	Scram		21.4	21.4	II	Retested. Failed to drop past 25" from 30" drop. Retested O.K. 10" drop in .8 sec May 4
5/8/59	Hot-300 psi	Scram		21.3	20	II	Retested O.K. 30" drop in 1.5 sec. May 9
5/14/59	Hot-300 psi	Scram		19	14.8	II	Retested O.K. May 15
5/20/59	Hot-300 psi	RDT		30		II	Hung up on all 5 tests Retested 5 times O.K. from 10" .6 sec. drop time - May 21
6/1/59	Hot-300 psi	RDT (Withdrawal)			25.2	III	Could not reach 30", O.K. on full travel June 10.
9/4/59	Hot-300 psi	RDT	2.93	30		I	Retested O.K. 30" drop in 1.29 sec. Sept. 9
9/12/59	Hot-300 psi	RDT	2.21	30		I	Retested O.K. from 10" Sept. 14
2/11/60	Hot-300 psi	Scram		16.2	10.6	III	Refer to Malfunction Report No. 18. Replaced mechanism. Retested O.K. 30" drop in .73 sec.
2/13/60	Hot-300 psi	Scram		17	13.0	II	Brake on motor stuck; released brake; retested O.K. from 20.4"

SL-1 Control Rod Operational History

Rod No. 7 (Cont)							
Date	Reactor Condition	Rod Operation	Drop Time (sec)	Drop Height (in)	Hang up Location (in)	Type of Sticking	Action Taken
7/5/60	Hot-300 psi	Scram		18.6	17.8	II	Retested O.K. dropped from 30" in 1.04 sec.
7/11/60	Hot-300 psi	Scram		17.6	17	II	Removed negator spring; retested O.K. from 10" drop July 15
11/12/60	Cold	Rod Withdrawal		2	2	II	Retested O.K. dropped from 11.3 in .67 sec.
11/18/60	Hot-300 psi	Scram		19	2	II	Operated O.K. on return to bank position.
11/19/60	Hot-300 psi	Rod Exercise (Withdrawal)		19.4		II	Operated O.K. in a rod re-banking operation.
11/20/60	Hot-300 psi	Scram	9	22		I	Retested O.K. dropped clean from 19.6 Nov.21
11/25/60	Hot-300 psi	Scram		19.2	2	II	Retested O.K. dropped clean from 19.4.
11/27/60	Hot-300 psi	Scram		20	20	II	Retested O.K. dropped clean from 10"
11/27/60	Hot-300 psi	Scram		19.3	3	II	Retested O.K. dropped clean from 10"
11/28/60	Hot-300 psi	Scram		17	15	II	Retested O.K.
12/3/60	Hot-300 psi	Scram		15.4	2.8	II	Scrammed O.K. on Dec.6 from 19.5"
12/7/60	Hot-300 psi	Scram	3.9	19.1		I	Retested O.K. dropped clean from 19.4
12/7/60	Hot-300 psi	Scram		19.4	1	II	Retested O.K. dropped clean from 17"
12/12/60	Hot-300 psi	Scram		19.2	3	II	Retested O.K. dropped clean from 18.8 on Dec.13.

SL-1 Control Rod Operational History

Rod No. 7 (Cont)							
Date	Reactor Condition	Rod Operation	Drop Time (Sec)	Drop Height (in)	Hang up Location (in)	Type of Sticking	Action Taken
12/19/60	Hot-300 psi	Individual Rod Scram		25	25	II	Retested O.K. dropped from 16" in .51 sec. Dropped from 30" in 1.71 sec.
12/22/60	Hot-300 psi	Rod Exercise (Full)		19.6	<3	II	Failed to drop from 19.4", 12/23/60
12/23/60	Hot-300 psi	Individual Rod Scram		19.4	19.4	II	Plant Shut down

SL-1 Control Rod Operational History

Rod No. 9							
Date	Reactor Condition	Rod Operation	Drop Time (Sec.)	Drop Height (in)	Hang up Location (in)	Type of Sticking	Action Taken
8/11/59	Hot-300 psi	Scram		19.8	14	II	Retested O.K. from 10"
9/4/69	Hot-300 psi	RDT	2.11	20		I	Retested O.K. 10" drop 9 Sept.
9/9/59	Hot-300 psi	RDT		20	16	II	Retested. Hung up at 10". Removed 1 negator spring, tested O.K. from 30" 1.22 sec.
9/12/59	Hot-300 psi	Individual Scram		20.7	17	II	Retested O.K. 20" drop in .57 sec.
2/17/60	Hot-300 psi	Scram		22	22	II	Retested O.K. 21.6" drop in .68 sec.
6/21/60	Hot-300 psi	Scram		18.6	18.6	III	Failed aluminum shear key in drive gear. Retested O.K. from drop of 18.4"
7/5/60	Hot-300 psi	Rod Withdrawal		0	0	III	Failed aluminum shear key in drive gear; replaced shear key - operated O.K.
11/27/60	Hot-300 psi	Scram		18.2	18.2	II	Retested O.K. on Scram, 18.5" drop in 1.28 sec.

Of sixty-two rod sticking incidents during either a drop test ($>10''$) or a five-rod scram, forty of these incidents took place over a twenty-two month period. The percentage of rod sticking incidents considering the number of scrams and number of rod drop tests (does not include 10 inch 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 (not including 10 inch rod drop tests) increased markedly to approximately 13.0%. See Table III.

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 had been operated more frequently than any of the other rods. Prior to the last shutdown period, No. 9 had performed without any type of sticking incident for a period of six months.

In instances of 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. However, 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 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 that at the first convenient plant shutdown period to check rod No. 1 to determine the cause of sticking. On November 9, 1960 (refer to ABWR-OPR-1777) a burr was removed from the upper inside edge of the shroud of control rod No. 1 to allow free rod movement above 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 inches for a 30 inch drop test (June 1, 1959) although it was fully withdrawn in the next test.

C. 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.

TABLE III

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

Type of Sticking	Rod Numbers					Total
	1	3	5	7	9	
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

Type of Sticking	Rod Numbers					Total
	1	3	5	7	9	
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

Rod Operation	Type of Sticking			Total Sticking Incidents
	I	II	III	
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.

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.

a. Tests conducted on various clutches have indicated the following:

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

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

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

It is believed from the above tests that the torque delivering capacity of the two used clutches is representative of other SL-1 clutches.

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:

<u>Maximum Torque Using a Single Hand</u>	<u>Maximum Torque Using Two Hands</u>
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 No. 1 and No. 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 may be that the center mechanism had fewer problems than any other mechanism simply due to less misalignment in its rod mechanism system because of its central location.

The limit switches are mounted to the top of the drive assembly. The object of the limit switches is for such things as rod-in and rod-out indication and rod low indication ^{1/}. 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 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.

^{1/} Indicates rods below three inches. Following a reactor shutdown or scram, all control rods must be below three inches before any control rod can be raised

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 hotwell. 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 five mechanisms) a sufficient high frictional force could be obtained to prevent scrambling 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 in low leakage rate and in low resistance to scrambling.

In summary, it is believed that the seals caused additional frictional retarding forces to be supplied to the control rod system. However, 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,

1959) 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 product 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 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 No. 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 Shrouds

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 Argonne National Laboratories 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.

In reviewing the control rod and shroud design for unrestricted movement of a control rod blade, (refer to ANL drawings RE-7-21235E and RE-18537-F), 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 time. 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, Inc., 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 (Table II) shows sticking incidents are not reproducible and sticking generally occurred above fifteen inches. If inward distortion were a prime cause for sticking, it would be expected to occur more frequently in the region below thirteen 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 shutdown, No. 9 rod was successfully scrambled 130 times ¹/with only one instance of sticking. This instance occurred November 28, 1960, when it hesitated momentarily at its drop height of 18.5 inches 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.

1/ Data taken from Power History Log

V PLANT EXPANSION

The SL-1 plant was expanded for higher power operation. To provide the additional capacity, a heat dump loop designed around a PL condenser was constructed. The purpose of the loop was to demonstrate operation and accomplish condenser testing in addition to higher power operation of SL-1.

A. PL CONDENSER DESCRIPTION

The PL condenser is a two pass heat exchanger consisting of a condenser section (one pass) containing six rows of 35 finned tubes each and a subcooling section (one pass) containing one row of 35 finned tubes designed to condense 13,011 lbs/hr of wet steam (10% moisture, 11 lbs/hr non-condensibles) at 15 inches Hg abs and subcool the condensate to 100 °F. Four fans blow the cooling air up through the tube bundles during operation. Figures 23 and 24 are photographs of the completed installation.

The condenser is to be used in climates where the outside ambient temperatures will normally be considerably below 32 °F. In order to prevent condenser freeze-up, the inlet air to the condenser is tempered by mixing hot air from the condenser exhaust with the outside air. To accomplish this, the exhaust louvers and the recirculation louvers are automatically controlled to maintain condenser inlet air temperature. As the inlet air temperature decreases due to colder conditions outside the exhaust louvers close and the recirculation louvers linked with the exhaust louvers open, recirculating more hot air with the inlet air thereby maintaining a constant inlet air temperature. The inlet air louvers are operated manually. Figure 25 is a photograph of the louver installation.

B. CONSTRUCTION

Figure 26 is a flow diagram of the expansion loop as built. Steam from the reactor (300 psig, saturated) is throttled across a valve and expanded to the PL condenser where it is de-superheated, condensed and the condensate subcooled. The condensate drains to a hotwell and is returned to the reactor via the feedwater system.

The loop, constructed of stainless steel, employed heliarc welding. Dye penetrant tests were used for all welded joints on stainless steel piping. After completion, the entire system was satisfactorily hydrostatically tested and cold flushed several times. Functional testing of condenser fans, pumps, and valve motors was accepted following inspection as outlined in the construction contract. Figures 27, 28, 29 are photographs of the plant expansion loop. Figure 30 shows the condenser being unloaded. The PL condenser, the largest unit used in the expansion loop was handled without difficulty.

PL CONDENSER BUILDING

FIGURE 23

PL CONDENSER LOOP FLOW DIAGRAM

FIGURE 26

HOTWELL AND GAGE BOARD FOR PL CONDENSER

FIGURE 27

C. TEST OPERATION

1. Acceptance Test

Both cold and hot loop acceptance testing of the loop was performed. Cold checkout consisted of filling and circulating water using the booster pump through the system. Electrical, instrument and mechanical equipment was checked for proper functioning. Before starting hot checkout, the system was steam cleaned to flush any organic or foreign material from the system. Approval to operate the expansion loop using reactor steam (reactor power not to exceed 3 MW (t)) was obtained. The system was operated for three hours with 4600 lbs/hr steam flow while maintaining the condenser pressure below 8 psig. All the steam during the three hour run was condensed and drained to the radioactive waste system. Samples of condensed steam were taken from both the hotwell and condenser and analyzed for oil and undissolved solids. Although these samples indicated a clean condition, visual observation of stainless steel machinery chips in the condenser header indicated an unacceptable condition. Following three hours of additional steam cleaning, the loop was found acceptable for operation.

Hot checkout was accomplished in two parts; operational checkout of the feedwater system, and low steam rate operation of the system. During the hot checkout, instrumentation was checked and calibrated. No major difficulties during initial startup and operation were encountered.

2. Operator Training

Following the acceptance test, the loop was operated at various steam rates for operational training of personnel. Drafts of operating instructions were prepared and reviewed prior to inclusion in the plant operating manual.

3. Performance Test

Upon completion of the acceptance testing and operator training, a continuous 72 hour test run at 7800 lbs/hr reactor steam flow (300 psig, saturated) to the PL condenser from the reactor was performed. Reactor power was maintained at 3 MW (t) for this test. Minor shake-down malfunctions, such as leaking valves, developed during the run, were repaired, and did not require securing the loop.

After receipt of AEC approval for reactor operation at 4.7 MW (t) - equivalent to about 14,000 lbs/hr steam flow - the plant was operated to obtain the condenser test data shown in Table I.

TABLE IV

EXPERIMENTAL TEST DATA

Run	1	2	3	4	5	6	7*	8*
Steam to Condenser and Ejectors	12,100	12,000	12,000	8,000	12,100	12,200	12,100	12,200
Condenser Pressure In. Hg. Abs	21.2	16.2	19.3	7.2	14.4	11.2	12.2	11.2
Inlet Steam Temp. - °F	315	318	318	309	317	317	318	315
Temp. in Rear Header - °F	181	174	179	114	170	159	165	160
Condensate Temp. - °F	95	91	95	80	85	80	151	150
Inlet Air Temp. - °F	63	51	63	63	41 ⁺	32 ⁺	40 ⁺	40 ⁺
Outside Ambient Temp. - °F	41	26	26	23	41	32	40	40
Recircu. Louver Position Right/Left % open	25/25	34/34	40/40	50/42**	0/0	0/0	0/0	0/0
Atmos. Pressure in Hg.Abs.	25.22	25.22	25.22	25.22	25.22	25.22	25.22	25.22
Ejector Steam Flow #/Hr	189	185	185	185	187	365	187	365
Steam Generation Rate #/Hr	13,500	13,200	13,200	9,150	13,550	13,700	13,550	13,800
Reactor Power Mw(t)	4.54	4.46	4.45	3.11	4.53	4.49	4.43	4.40
Air Flow #/Hr x 10 ⁻³	540	540	540	580	540	540	540	540

* No Subcooler Flow

+ Assumed Equal to Outside Ambient

** Indicator Calibration Error

ACCURACY OF MEASUREMENTS

Steam Flow	± 2%
Air Temperature	± 1°F
Air Flow	± 5%
Steam & Condensate Temperature	± 2°F
Condenser Vacuum	± 1/2 In. Hg.
Louver Position	± 1%

The cold weather conditions existing during the condenser operation required that much of the testing be performed while using the air recirculation system to maintain inlet air above freezing. Tests were performed to determine the performance of the condenser and the louver control system.

The air mixing characteristics of the louver system was determined using a hot wire anemometer probe. Traverse readings of air flow and temperature were taken across the entire inlet and outlet faces of the condenser. The inlet air temperature recorded in Table I is a numerical average of all points taken in the traverse readings.

It should be noted that test runs number 5 thru 8 were performed with the recirculation louvers completely shut thus allowing outside ambient air to pass thru the condenser without preheating with recirculation air. It is assumed therefore that for these runs the condenser inlet air temperature is equal to outside ambient air temperature.

Runs number 7 and 8 were performed by bypassing the subcooler section of the condenser. This was done by securing the normal condenser outlet valve and removing the condensate directly from the reversing header which is at the inlet to the subcooler.

The circulation louver position recorded in run number 4 was spurious due to a calibration error of the indicator during this run. It should be noted that although the two position indicators do not agree, the actual measured inlet air temperature is the desired controlled temperature.

An evaluation report, CEND 139, PL Condenser Evaluation, June 26, 1961 presents a full study of the test results.

VI REACTOR CHEMISTRY

Routine analyses of the SL-1 plant water and steam were conducted for quality control and the results are discussed below:

A. REACTOR WATER

1. pH

pH was measured 6 times daily during normal plant operation, and at least once daily during shutdown periods, throughout the year. Water quality was maintained within specs. (pH 6.5 - 7.5) during the year with the following exceptions:

a. On June 10, a resin bed retaining screen was punctured during a resin change, (malfunction #27) causing resin to enter the reactor vessel. This event was undetected by the crew on shift, and during the subsequent startup the reactor heat decomposed the resin, forming some decomposition acid, and causing the pH to decrease to approximately 5.0.

Efforts to remove the decomposed resin from the reactor system and restore water quality included flushing the reactor, operating the purification system, and changing of the purification system and feedwater filters. On June 21, satisfactory pH limits were regained and routine plant operation was resumed.

On November 29, the pH increased to 7.75. When adjustments to the flow rates through the use of the ion exchange columns did not improve the condition, the NA form cation exchange resin, in use at the time, was replaced with an H⁺ form resin. The high pH was then brought within specification.

b. Resistivity

A resistivity specification for water was set at 0.5×10^6 ohm cm. Based upon operational checks 6 times daily during reactor operation and once daily during shutdown water resistivity was maintained with minor exceptions in a range of 0.5×10^6 ohms cm to 2.6×10^6 ohm cm during the year. A specific exception occurred on June 3, 1960 when malfunction number 27, as discussed above, occurred. A non-operation period of 10 days was required to clean up the reactor and restore water resistivity to specification.

2. Oil

A by-weekly oil analysis of reactor water indicated an oil concentration in the range of 0.2 - 0.4 ppm throughout the year. As compared with a 1 ppm

spec. earlier in reactor history, oil was found in the reactor water in excessive amount, due to a faulty deep well pump. This was corrected by modification of the pump and installation of a diatomaceous earth filter in the raw feed water line.

3. Oxygen

An oxygen analysis was performed weekly. Oxygen concentration never exceeded 0.5 ppm throughout the year with results ranging from 0.3 to 0.5 ppm as compared to the specification of 1 ppm.

4. Radioactivity

Radiochemical analysis of reactor water was conducted on a weekly basis. Gross activity ranged from $1.0 - 1.5 \times 10^6$ d/m/ml, with over 99% of this from NA^{24} . Other significant nuclides found, and the ranges of their activities include:

- a. $\text{Mn}^{56} - 3.5 - 4.0 \times 10^4$ d/m/ml
- b. $\text{I}^{131} - 1.5 - 2.4 \times 10^3$ d/m/ml
- c. $\text{A}^{51} - 1.1 - 1.2 \times 10^3$ d/m/ml

Gross alpha-beta and gamma activity averaged 0.3 ppm throughout the year.

5. Chlorine

A bi-weekly analysis for chlorine indicated concentrations ranging from 0.3 - 0.8 ppm as compared to a specification of 1 ppm.

6. Total Solids

The results of a bi-weekly analysis for total solids in reactor water ranged from 0.3 to 0.8 ppm as compared to a 1 ppm specification.

7. H_2O Decomposition

The results of analysis of water for radiolytic decomposition radicals at reactor power levels from 1 - 4.7 MWT are plotted in Figure 31. The ratio of H to O was the expected 2.1.

EFFECT OF REACTOR POWER ON RADIOLYTIC DECOMPOSITION OF SL-1 REACTOR WATER
Fig. 31

B. REACTOR STEAM

1. Quality

Calorimetric measurements of steam quality were made on a monthly basis, except during power operational tests, when they were performed daily. At no time was steam quality less than 99%, and it did not change significantly with reactor power out-put up to 4.7 MWT.

2. Radioactivity

Bi-weekly measurements of NA^{24} in reactor water and condensed steam were made to determine the decontamination factor (D.F.) ($\frac{\text{NA}^{24} \text{ water}}{\text{NA}^{24} \text{ steam}} = \text{D. F.}$)

Values of the DF ranged between 1.0×10^4 and 4.0×10^4 for power levels of 1 to 4.7 MWT.

C. REACTOR MAKEUP WATER

1. Reactor makeup water was prepared by pumping raw well water containing 135-150 ppm hardness and 0.5 - 2.5 ppm oil, through a diatomaceous earth filter and mixed bed ion exchange bed.

Weekly analytical results from reactor makeup water samples are summarized below:

- a. pH 6.5 - 7.5
- b. Resistivity 2×10^6 - 7×10^6 ohms cm
- c. Oil content ≤ 0.5 ppm
- d. Total solids ≤ 1 ppm
- e. Chlorides ≤ 1 ppm

VII HEALTH PHYSICS AND SAFETY EVALUATION

A review of the Health Physics and Safety aspects at the SL-1 plant for 1960 are presented in this report. This report covers plant radiation levels, air activity levels, disposal of contaminated wastes and personnel exposures.

A. PLANT RADIATION LEVELS

Since the reactor operating floor was designated as an accessible area during reactor operation, a continuing survey of radiation levels was performed to insure safe personnel exposure limits. Table V presents radiation data at specific operator stations or equipment locations on the reactor operating floor.

The apparent increase in general levels in the November 1960 data was first observed after installation of an instrumented fuel assembly in the core in September 1960. Since several thermocouples penetrated the cladding of the instrumented assembly fuel plates, it was to be expected that some fission products, i.e., principally inert fission gases, would be released into the main steam system. The increase in radiation level caused no operational difficulty.

B. PERSONNEL EXPOSURE SUMMARY

A summary of the radiation exposure for all personnel assigned to the SL-1 reactor is shown in Figure 32. The maximum accumulated whole body exposure for this year was 1300 milliroentgens (including 100 mr estimated for December) as compared to the prescribed maximum allowable exposure for one year of five roentgens.

C. AIR ACTIVITY

Minor spills of radioactive materials and leakage of contaminated gases resulted in above MPC levels in some instances. Respiratory protection in all instances was provided. The most significant of these are listed below:

1. A leak around the turbine gland seal resulted in gross beta-gamma air concentrations of 8.0×10^{-6} $\mu\text{c/cc}$.

TABLE V

SL-1 RADIATION LEVELS DURING 3 MW POWER OPERATION

Location	mr/hour					
	Dec. '58	May '59	July '59	Nov. '59	May '60	Nov. '60
Lower Landing of Stairway to Operating Floor	0.1	0.6	0.7	0.7	1	
Between Lower & Middle Landing	0.1	0.1	0.1	0.1	.2	
Doorway, Operating Floor Level	1	1	1	1	1	3
General Operating Floor	-	3	3	4	5	8
Middle, Turbine Control Panel	5	6	6	7	7	6
Purification Panel Near Valve No. 8	-	17	24	31	30	60
Floor Over Purification Vault After Addition of Lead	-	-	-	20	35	35
Operating Floor Between Purification Panel & Shield Block	-	10	12	15	16	23
Feedwater Pump	1	1	1	1	2	5
Main Steam Line Near Reactor	30	20	30	30	30	60
Turbine Casing	4	6	6	11	16	16
Main Steam Line Near Turbine	5	8	8	8	10	16
Condenser	-	-	-	45	50	-
Hotwell	-	3	3	8	8	10
Simulated Heat Load Heat Exchanger Trap	-	28	30	40	29	27

PERSONNEL EXPOSURES FOR ALL SL-1 PERSONNEL DURING 1960

FIGURE 32

2. A steam leak around the head of the turbine throttle valve resulted in a gross beta-gamma activity of 2.4×10^{-4} $\mu\text{c}/\text{cc}$. Specific radionuclides were identified as I^{131} , I^{133} , Kr^{88} , Xe^{133} , and Xe^{135} .

3. A steam leak in the AE side of the PL-2 system resulted in a gross beta-gamma activity of 1.0×10^{-4} $\mu\text{c}/\text{cc}$.

D. WASTE DISPOSAL

1. SL-1 Stack Effluent Release

Efforts to continuously monitor gross activity and identify the major radionuclides released from the plant air ejectors were continued. Table VI shows a monthly summary of gaseous waste discharged from the SL-1 stack in 1960. The conclusions from these monitoring efforts are as follows:

a. A significant increase in off-gas release rate occurred in September. An instrumented fuel assembly which had exposed fuel, was placed in the reactor and the reactor was operated.

b. Spectral analysis of off-gas identified Xe^{138} , Xe^{135} , Xe^{133} , Kr^{88} , Kr^{85} , as the major radionuclides.

c. A total release to the environment for the year 1960 has been computed at 1460 curies.

2. Liquid Waste

Liquid waste resulting from SL-1 operations was collected in the retention tanks for sampling, measurement and subsequent discharge to the leaching bed. Table VII shows the monthly liquid waste discharge for 1960. A total volume of 57,500 gallons of liquid waste containing a total gross activity of 3.2×10^5 μc was discharged to the leaching bed. The increase in specific activities in June and November is attributed to decontamination work subsequent to maintenance operations on the turbine generator, the purification system and PL-2 loop and pump pit.

3. Solid Waste

The contaminated solid wastes generated at the SL-1 reactor site during this year consisted of paper and rags. A total of 1220 cubic feet of waste containing 8.2×10^5 μx (principally Na^{24} with traces of fission products) and weighing 2130 pounds was shipped to NRTS burial ground.

TABLE VI

SL-1 STACK EFFLUENT RELEASE - 1960

Month	Form	Total Activity curies	Radionuclides Identified
January	Gaseous	0	Kr ^{85M} , Kr ⁸⁸ , Xe ¹³³⁻¹³⁵⁻¹³⁸
February	Gaseous	8	Kr ^{85M} , Kr ⁸⁸ , Xe ¹³³⁻¹³⁵⁻¹³⁸
March	Gaseous	33	Kr ^{85M} , Kr ⁸⁸ , Xe ¹³³⁻¹³⁵⁻¹³⁸
April	Gaseous	22	Kr ^{85M} , Kr ⁸⁸ , Xe ¹³³⁻¹³⁵⁻¹³⁸
May	Gaseous	28	Kr ^{85M} , Kr ⁸⁸ , Xe ¹³³⁻¹³⁵⁻¹³⁸
June	Gaseous	9	Kr ^{85M} , Kr ⁸⁸ , Xe ¹³³⁻¹³⁵⁻¹³⁸
July	Gaseous	18	Kr ^{85M} , Kr ⁸⁸ , Xe ¹³³⁻¹³⁵⁻¹³⁸
August	Gaseous	11	Kr ^{85M} , Kr ⁸⁸ , Xe ¹³³⁻¹³⁵⁻¹³⁸
September	Gaseous & Particulate	350	Kr ^{85M} , Kr ⁸⁸ , Xe ¹³³⁻¹³⁵⁻¹³⁸ Cs ¹³⁸
October	Gaseous & Particulate	350	Kr ^{85M} , Kr ⁸⁸ , Xe ¹³³⁻¹³⁵⁻¹³⁸ Cs ¹³⁸
November	Gaseous & Particulate	300	Kr ^{85M} , Kr ⁸⁸ , Xe ¹³³⁻¹³⁵⁻¹³⁸ Cs ¹³⁸
December	Gaseous & Particulate	330	Kr ^{85M} , Kr ⁸⁸ , Xe ¹³³⁻¹³⁵⁻¹³⁸ Cs ¹³⁸
TOTAL		1460	

TABLE VII

SL-1 RADIOACTIVE LIQUID WASTE SUMMARY - 1960

Month	Volume (gallons)	Total Activity uc	Radionuclides
January	2,600	1.17×10^2	Na ²⁴ (principal)
February	3,100	1.86×10^3	Na ²⁴ , I ¹³¹ , I ¹³³
March	3,600	2.10×10^2	Na ²⁴ , Cd ¹¹⁵ , I ¹³¹ , In ¹¹⁵ , Sr ⁹⁰
April	2,700	2.0×10^3	Na ²⁴ , Cd ¹¹⁵ , I ¹³¹ , In ¹¹⁵ , Sr ⁹⁰
May	2,700	2.2×10^3	Na ²⁴ , I ¹³¹ , I ¹³³
June	8,500	1.0×10^5	Na ²⁴ , Cd ¹¹⁵ , I ¹³¹ , I ¹³³
July	6,600	3.7×10^3	Na ²⁴ , I ¹³¹ , I ¹³³ , Cd ¹¹⁵
August	5,200	9.7×10^4	Na ²⁴ , I ¹³¹ , I ¹³³ , Cd ¹¹⁵
September	7,300	2.0×10^4	Na ²⁴ (principal)
October	8,700	1.0×10^3	Na ²⁴ (principal)
November	6,500	1.04×10^5	Na ²⁴ , I ¹³¹ , I ¹³³
December	Records Not Available		
TOTAL	57,500	3.2×10^5	

VIII SL-1 ADMINISTRATION

A. SECURITY

On August 29, 1960 a system of Administrative Control of security was instituted at SL-1. Under this system ingress and egress into and out of SL-1 was handled by the activation of robot doors and gates from the receptionist's office in the SL-1 Administration Building. The security receptionist handled badge exchange and security documentation so that IDO was not required to man Post 903. The responsibility for security on the back shifts and on weekends was delegated to the Chief Operator. Periodic patrols were made within the SL-1 perimeter on a prearranged schedule established jointly by the SL-1 Security Officer and the NRTS Patrol Headquarters.

B. SS MATERIAL

A procedure manual covering the handling and control of SS Material was developed and submitted in draft form to the Commission. The procedure was revised to reflect certain desired change, but was never resubmitted. During the year periodic inventory checks were made to assure proper accountability of SS Material. Arrangements were made to store fuel elements in CPP pending a future decision as to disposition of the elements. In all transactions involving SS Material the prescribed regulations for handling and accountability were followed.

C. PROCUREMENT

Combustion Engineering, Inc. used the warehouse and purchasing sections of Phillips Petroleum Co. as the sources of supplies and equipment. A procurement procedure was developed and approved by the Commission which met both the requirements of Combustion and the Commission. Because of the lack of storage space at SL-1 inventories were held to a minimum commensurate with the required lead times for critical items. This resulted in the issuance of many more requisitions than the dollar volume would normally require.

D. PROPERTY CARE AND CONTROL

Surfacing the areas within the perimeter was completed during the year and minor repairs to buildings were carried out as periodic inspections indicated the need. All capital items were inventoried and tagged by mid-year. As additional items were acquired, property records were established and the equipment tagged.

IX APPENDIX

A. MASTER TEST PLAN

A master test plan was developed for Fiscal Year 1961. The plan served as a framework for the first quarter and succeeding test schedules.

A plan was divided into test programs which defined major evaluation areas. Each program contained specific tests related to the accomplishment of the program. Flexibility was designed into the plan so that new tests could be added to a program as data and results indicated.

The plan consisted of six major test programs as follows:

1. Power Extrapolation Test Program (T-1)

- a. Instrumented fuel assembly test to measure fuel temperature at design reactor power levels and at increased reactor power levels.
- b. The stability and transient behavior of the reactor as power is increased will be investigated.
- c. PL type condenser module performance test will evaluate condenser operation under various operating conditions.
- d. Plant operations will be evaluated at increased power operation.
- e. Accelerated burnup of the SL-1 Core-I will be initiated.

2. SL-1 Core-I Test Program (T-2)

The nuclear, mechanical, and metallurgical behavior of the SL-1 Core-I will be evaluated at existing and higher power levels.

3. PL Components Test Program (T-3)

An open ended test program for obtaining design information for PL plant components will be conducted. Initial experiments under this program are:

- a. Reactor Transient Behavior Tests
- b. Ag-In-Cd Corrosion Test
- c. Control Rod Scram Tests

4. SL-1 Core-2 Component Test Program (T-4)

- a. A PL type fuel element will be performance tested.
- b. A PL type control rod and drive will be performance tested.
- c. The power production in UO_2 type fuel elements will be determined.

5. SL-1 Core-2 Performance Test Program (T-5)

- a. Critical experiments on SL-1 Core-2 will be performed prior to operation.
- b. The SL-1 Core-2 will be evaluated during core life.

6. Depleted SL-1 Core-I Post Irradiation Program (T-6)

The destructive and non-destructive evaluation of the SL-1 Core-I will be performed.

B. STATUS OF TEST EFFORT ON DECEMBER 23, 1960

1. T-1 Power Extrapolation Test Program

- (a) Instrumented fuel assembly test to measure fuel temperature at normal and increase reactor power levels was successfully accomplished.
- (b) The stability and transient behavior of the reactor at increased power levels was investigated.
- (c) PL type condenser modul performance test evaluating condenser operation under various operating conditions was in progress.
- (d) Plant operations were evaluated at increased power levels.
- (e) Accelerated burnup of the SL-1 Core-I was initiated by several higher power runs.

2. SL-1 Core Test Program (T-2)

- (a) Visual inspection of the core was performed.
- (b) Non-destructive testing of an SL-1 fuel element was performed in the AREA hot cell.
- (c) Core lifetime measurements were continued.
- (d) A core thermal flux map was prepared.

(e) Plant chemistry program was revised and tests performed within the new program. Additional equipment such as fume hood and single channel gamma spectrophotometer had been procured.

3. PL Components Test Program (T-3)

(a) Reactor transient tests were performed.

(b) Ag-In-Cd specimens clad and unclad were irradiated in the SL-1 to determine corrosion rates.

(c) Control rod tests (hot and cold conditions) were performed. A test control rod (non-poison) and mechanism was installed to test various mechanism components.

Preliminary work had been started in preparation for installation of the lead control rod mechanism and Core II fuel element. Evaluation was in progress to prepare the SL-1 facility for Core I removal and Core II installation.

C. PREVENTIVE MAINTENANCE SCHEDULE

Following installation and initial operational testing of all mechanical, electrical, and electronic components and systems in 1958, a preventive maintenance program was initiated at SL-1 to insure continued reliable operation. The program was based on conventional methods for maintenance and the schedule based on manufacturers' recommendations, standard operational schedule and good maintenance practice.

" 1. Equipment History File

a. Applicability

The provisions of this publication apply to the following sections:

- (1) Electrical
- (2) Electronic
- (3) Mechanical

b. Responsibility

It was the responsibility of the Section Supervisor to initiate and maintain the Equipment Data Cards (ENG Form 1851) for all items of equipment for which his Section has maintenance responsibility.

c. Use

(1) This file will be located in the Plant Superintendent's office.

(2) This file will provide a ready reference for nomenclature on any item of equipment and will also show any modifications or repairs, thus providing information for future evaluation (replacements or new plants).

d. Entries

The following information will be recorded in the spaces listed below:

(1) Equipment: Name of the item (example: feedwater pump No. 1).

(2) Location: Place of installation (example: operating floor of reactor building).

"(3) Furnished on: Enter the date received. If original plant equipment enter 1 June 1958.

(4) Contractor: List purchaser of item (example: Argonne, etc.).

(5) MFR: Manufacturer as listed on equipment name plate.

(6) Model, serial number and other pertinent information will be listed in appropriate spaces,

(7) Remarks: Describe very briefly all modifications and repairs. This entry will be preceded by the date written in the following manner: 2 Oct 58. The person making the entry will initial the card following the brief description. All entries for modifications will include the authority (example: manufacturer's instruction, verbal instruction by Plant Superintendent, etc.).

2. Equipment Maintenance Log

a. Applicability

The provisions of this publication apply to the following Sections:

- (1) Electrical
- (2) Electronic
- (3) Mechanical

b. Responsibility

The Section Supervisors are responsible for maintaining their Section Equipment Maintenance Log in accordance with the provisions of this publication.

c. Use

The Equipment Maintenance Log (EML) is a detailed record of all pertinent information that has occurred or work that was accomplished on equipment maintained by the individual Sections.

d. Entries

(1) All entries in the EML will be clearly printed in ink. Date entries will be made in the following order: day, month (spelled out or abbreviated), and the last two digits of the calendar year (example: 9 Aug 58). All entries should be detailed and as clear and complete as possible.

(2) Remarks: This column will contain information of a detailed nature and include any checks, abnormal conditions, equipment status changes, maintenance or calibration performed on plant equipment.

"Entries need not be on individual items that are part of a section check sheet (example: instrument manual standardization, diesel generator oil level, etc.); however, an entry is required to indicate that section check sheets have been accomplished and whether equipment was functioning properly or not (example: daily instrument check, o.k.; diesel generator check, o.k.). Entries will be made on all defective items and items that are replaced as a result of any maintenance, checks or inspections, whether items are a minor component or an integral part of a major component. Entries on integral parts are to include a description with the defect noted, and what corrective action was taken, if any, but need not include the serial number. This entry is used to establish and maintain an adequate stock supply system. Entries on major components are to include a description, serial number, and defect of item being replaced. A separate entry will indicate description and serial number of replacing components. This information will be reflected on the Scheduled Maintenance Card and Equipment Data Card maintained on each major component.

(3) Initials: This column will contain the initials of the person responsible for remarks entered.

(4) Erroneous entries: If an erroneous entry is made in the EML, line through the entry and initial this line at the edge of the page.

3. Scheduled Maintenance File - Mechanical Section

a. Applicability

The provisions of this publication apply to the Mechanical Section:

b. Responsibility

The Section Supervisors are responsible for maintaining the Scheduled Maintenance File (Form ALPR-1) in accordance with the provisions of this publication.

c. Use

(1) The Scheduled Maintenance Cards are used to insure that all ALPR equipment is properly serviced and checked on a periodic schedule.

(2) Scheduled Maintenance Cards will be filed by the Section maintaining the equipment.

(3) Upon permanent transfer of equipment, the card will be placed in a dead file and destroyed after a period of three months.

(4) Scheduled Maintenance Cards are required on all equipment which requires an inspection and/or scheduled maintenance.

" d. Entries

(1) All entries on the Scheduled Maintenance Card will be typed or printed in ink.

(2) Work to be performed: This section of the card will contain all the work to be performed on the piece of equipment to be maintained (example: lubricate, inspect, clean, etc.).

(3) Inspection period: This section is divided into twelve periods and will be used for twelve inspections regardless of frequency of the inspection periods. The individual performing the maintenance will enter the date that the work is performed in this section.

(4) Work performed by the individual performing the maintenance will place his initials in the space under the inspection period.

(5) When the twelve inspection periods are completed, a card will be attached to the completed card for additional inspections.

4. Scheduled Maintenance File - Electrical Section

a. Applicability

The provision of this publication applies to the Electrical Section:

b. Responsibility

The Section Supervisor is responsible for maintaining the Scheduled Maintenance Cards (SMC) (Form ALPR-2) in accordance with the provisions of this publication.

c. Use

(1) The SMC is a record of modification, calibration, lubrication, and maintenance of individual items of ALPR equipment. The SMC is not designed to replace Equipment Data Cards, but to be used to supplement information contained on the Data Cards.

(2) SMC's will be filed by the Electronic Section.

(3) Upon permanent transfer of equipment the SMC will be transferred to a dead file and destroyed after a period of three months.

(4) SMC's are required on all equipment, which requires an inspection, scheduled maintenance or calibration. SMC's are not required on cables, adapters, connectors, relays, hand tools, and similar items.

"d. Entries

(1) All entries in the SMC will be typed or clearly printed in ink. All entries should be brief, clear, and complete as possible.

(2) Maintenance Record

(a) This section of the SMC will be used to indicate performance of scheduled maintenance, lubrication, and calibration. It will also be used to record modification status.

(b) Publication: This column (A) will be used to indicate the authority for modification and/or calibration if directed by an organization other than the using organization.

(c) Remarks: This column (B) will be used to indicate any unusual conditions or difficulties encountered during maintenance or calibration. Calibration entries will indicate whether or not the calibration was satisfactory (example: instrument indicates within design tolerance). The entry will also indicate the component out of tolerance, if appropriate.

(d) Type: This column (C) will be used to indicate the type of operation performed (example: scheduled maintenance, calibration, modification, etc.).

(e) By: This column (D) will be initialed by the technician recording the information.

(f) Date: This column (E) will be completed in the following manner: day, month, (spelled out or abbreviated), and the last two digits of the calendar year (example: 9 Aug 58).

(3) Repair Record

(a) This section of the SMC will be used to record all repairs accomplished on the item of equipment.

(b) Remarks: This column (F) will be used to briefly describe the extent and type of defect or repair (example: replaced V-101 (6AH4) tube). When plant personnel cannot accomplish the required repair or cannot correct the malfunctions encountered, a concise analysis of the trouble will be entered prior to shipment of the equipment for repair.

(c) By: This column (G) will be initialed by the technician recording the information.

(d) Date: This column (H) will be completed in accordance with paragraph 2. c. (2) (f).

"(4) Equipment status

(a) This section is used to record the present status of the item of equipment.

(b) Remarks: This column (I) will be used to record the status of the equipment (example: in use, in storage, under repair, etc.).

(c) Date: This column (J) will be accomplished in accordance with paragraph 2. c. (2) (f).

(5) Erroneous Entries

If an erroneous entry is made on an SMC, line out the entry and initial this line at the edge of the card.

(6) When a section of an SMC is filled, close out the remaining unfilled sections of the SMC with diagonal ink lines. A new SMC will be added to a filled card to provide space for additional entries.

5. Scheduled Maintenance Suspense File

a. Applicability

The provisions of this publication apply to the following Sections:

- (1) Electrical
- (2) Electronic
- (3) Mechanical

b. Responsibility

This file will be initiated and maintained by the Plant Superintendent and will be located in the Superintendent's office.

c. Use

(1) This file is maintained as a backup for Scheduled Maintenance Files located in each Section. It gives the Plant Superintendent's office assurance that scheduled maintenance has been accomplished.

(2) The Plant Superintendent will pull the cards (From ALPR-3) from the suspense file when due for maintenance and give them to the appropriate Section Supervisor keeping an informal record of cards outstanding.

(3) Section Supervisors will make the appropriate entries on the cards after all the maintenance has been accomplished and return the cards to the Superintendent's office.

"(a) Item: This will describe equipment to be maintained.

(b) Period: This will indicate frequency of maintenance (example: Weekly, monthly, bimonthly, etc.).

(c) Serial No.: Serial number of item if applicable.

(d) Date maintained: This will indicate when the work was actually accomplished.

(e) Section Supervisors will initial card to indicate that maintenance has been completed.

(f) Date Due: This column will indicate when item is next due for maintenance.

(4) Daily Maintenance Check-Off Sheet"

In addition to the scheduled maintenance program, Daily Check-Off Sheets are maintained by the Electrical, Electronic, and Mechanical Section Supervisors. These sheets cover equipment requiring daily attention and not covered by Section 2.