PROCEEDINGS SERIES

REACTOR SAFETY
AND
HAZARDS EVALUATION TECHNIQUES

I

PROCEEDINGS OF THE SYMPOSIUM
ON REACTOR SAFETY AND HAZARDS EVALUATION TECHNIQUES
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INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA 1962

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SOME ASPECTS OF THE WTR AND SL-1 ACCIDENTS

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Abstract — Résumé — Аннотация — Resumen

SOME ASPECTS OF THE WTR AND SL-1 ACCIDENTS. The Stationary Low Power Reactor No. 1(SL-1), a three megawatt prototype reactor, underwent a nuclear excursion at the National Reactor Testing Station (NRTS), Idaho, on 3 January, 1961. Three military operators received fatal injuries and the core was severely damaged. Large amounts of radioactivity were released inside the reactor building; however, release of radioactivity from the building to the atmosphere was slight. This was the first fatal reactor accident in the history of reactor operation in the United States. Prior to the accident, the reactor had operated for 931 megawatt days, approximately 40% of its core life.

Primary efforts subsequent to the incident consisted of removal of the victims from the reactor building, determination of the nuclear status of the reactor, and investigations as to the cause of the accident.

Since the cause of the SL-1 accident has yet to be conclusively determined, the dismantling and decontamination of the reactor building had to proceed slowly in case some important evidence might be overlooked. The high radiation levels inside the reactor building also played an important part in slowing up the recovery operations. By the end of November 1961, the pressure vessel with the SL-1 core was removed from the reactor building and transported 40 miles to a large hot cell previously used to disassemble large experimental reactors. In the hot cell a more detailed examination of the disarranged core took place.

As a result of the SL-1 accident, substantial changes were made in organization and responsibility assignments within the AEC, to clarify and strengthen the safety surveillance over operating governmental reactors.

ACCIDENTS RÉCENTS AUX ÉTATS-UNIS - CONCLUSIONS QUANT A LA SÉCURITÉ DES RÉACTEURS. Au Centre national d'essai de réacteurs (NRTS) de l'Idaho, une saute de puissance s'est produite, le 3 janvier 1961, dans le réacteur à faible puissance constante No 1 (SL-1), réacteur prototype de 3 MW. Trois opérateurs appartenant aux forces armées ont été mortellement blessés et le coeur du réacteur a subi de graves dommages. De grandes quantités de matières radioactives ont été libérées à l'intérieur du bâtiment, mais une faible quantité seulement s'est échappée dans l'atmosphère. C'était la première fois qu'un accident mortel se produisait dans un réacteur aux Etats-Unis. Auparavant, le réacteur avait fonctionné pendant 931 mégawatt-jours, soit environ 40% de la durée de vie du coeur.

Les effort déployés après l'accident visaient essentiellement à retirer les victimes du bâtiment, à déterminer l'état du réacteur du point de vue nucléaire et à rechercher la cause de l'accident.

Comme on n'a pas encore pu établir avec certitude la cause de l'accident, il a fallu procéder avec précaution au démontage et à la décontamination du bâtiment, pour ne pas détruire des éléments de preuve importants. La radioactivité élevée à l'intérieur du bâtiment a largement contribué à ralentir les opérations. A la fin de novembre 1961, le caisson étanche contenant le coeur du réacteur avait été retiré du bâtiment et transporté, à une soixantaine de kilomètres, dans une vaste cellule de haute activité, déjà utilisée pour le démontage de grands réacteurs expérimentaux. Les différentes parties du coeur y ont été soumises à un examen plus approfondi.

НЕДАВНИИ АВАРИИ В СОЕДИНЕННЫХ ШТАТАХ И ИХ ПОСЛЕДСТВИЯ В ОБЛАСТИ БЕЗОПАСНОСТИ РЕАКТОРОВ. В стационарном реакторе малой мощности № 1 (SL-1), являющимся реактором-прототипом мощностью в 3 мгвт, в Национальном испытательном реакторном центре NRTS в Айдахо, 3 января 1961 года, произошло отклонение от нормального режима. Трое из обслуживающих реактор военнослужащих были убиты, а активная зона реактора сильно пострадала. Произошел

выброс значительного количества радиоактивности внутри реакторного здания; однако высвобождение радиоактивности из здания в атмосферу было незначительным. Это было первой аварией реактора со случаями смерти в истории работы реакторов в Соединенных Штатах. До аварии реактор проработал 931 мегаватт-дней, что составляет приблизительно 40% срока жизни его активной зоны.

Основные усилия после инцидента были направлены на извлечение жертв из здания реактора, определение ядерного состояния реактора и анализа причин несчастного случая.

Ввиду того, что нужно было определить причину аварии на реакторе SL-1, разработка реактора и дезактивация реакторного здания должны были производиться медленно с тем, чтобы не пропустить какого-либо важного вещественного доказательства. Высокий уровень радиации внутри реакторного здания также сыграл значительную роль в замедлении работ по восстановлению. К концу ноября 1961 года корпус высокого давления с активной зоной SL-1 был извлечен из реакторного здания и отвезен за 40 миль в большую горячую камеру, которая служила раньше для разработки больших экспериментальных реакторов. В горячей камере был организован более подробный осмотр разобранной активной зоны.

В результате аварии на реакторе SL-1 в организацию Комиссии по атомной энергии и в распределение ответственности были внесены значительные изменения для уточнения и усиления наблюдения над работой правительственных реакторов с точки зрения их безопасности.

В испытательном реакторе Вестингхауза (WTR), являющегося реактором бассейного типа мощностью в 60 мгвт, 3 апреля 1960 года произошла авария с тепловыделяющим элементом. Ни смертных случаев, ни чрезмерного облучения не произошло; однако расплавление одного тепловыделяющего элемента вызвало распространение продуктов деления в систему охлаждения реактора. Причина аварии не могла быть установлена с несомненной точностью, но вероятной причиной может быть повреждение оболочки диаметром более 1/2 дюйма в месте дефекта связи.

ACCIDENTES REGISTRADOS RECIENTEMENTE EN LOS ESTADOS UNIDOS, Y SU REPERCUSION EN LA SEGURIDAD DE REACTORES. *En el Reactor estacionario de baja potencia N° 1 (SL-1) de la Estación Nacional de Experimentación de Reactores (NRTS), de Idaho, reactor prototipo de 3 MW, se produjo un accidente de reactividad el 3 de enero de 1961. Tres operadores militares sufrieron lesiones mortales y el núcleo del reactor quedó muy deteriorado. Grandes cantidades de radiactividad quedaron libres en el interior del edificio del reactor; a pesar de ello, escapó poca radiactividad del edificio a la atmósfera. Este ha sido el primer accidente mortal registrado en la historia del funcionamiento de los reactores en los Estados Unidos. Hasta el día del accidente, el reactor había funcionado durante 931 megavatios-días, o sea, aproximadamente el 40 por ciento de la vida del núcleo.

Lo primero que se hizo, apenas ocurrido el accidente, fue retirar las víctimas del edificio del reactor, determinar el estado nuclear del reactor, e investigar la causa del siniestro.

Como quiera que aún no se ha determinado exactamente la causa del accidente del SL-1, fue preciso proceder lentamente al desmantelamiento y a la descontaminación del edificio del reactor a fin de evitar el pasar por alto algún elemento importante para la investigación. Asimismo, los altos niveles de radiación han sido una causa importante de la lentitud de las operaciones de recuperación. A fines de noviembre de 1961 se retiró, del edificio del reactor, la vasija de presión junto con el núcleo del SL-1, que se transportó a una amplia celda de gran actividad situada a 40 millas de distancia, que previamente se había empleado para desmontar grandes reactores experimentales. En la celda de gran actividad se hizo un examen minucioso del núcleo dañado.

Como resultado del accidente del SL-1, se introdujeron cambios sustanciales en la organización y deslindamiento de responsabilidades dentre de la AEC, a fin de determinar o intensificar la inspección de la seguridad de los reactores que el Gobierno tiene en funcionamiento.

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En el Reactor Experimental Westinghouse (WTR), del tipo tanque, de 60 MW, se produjo una avería en un elemento combustible el 3 de abril de 1960. No se registraron desgracias ni sobreexposiciones; sin embargo, se fundió el elemento combustible, provocando la dispersión de productos de fisión por el sistema de refrigeración del reactor. No pudo establecerse con certeza la causa del accidente, pero parece que puede atribuirse a la rotura de la vaina en una zona de contacto defectuoso de más de ½ pulgada de diámetro.

I. INTRODUCTION

Recent reactor accidents in the United States of America at the Westinghouse Testing Reactor (WTR) and the Stationary Low Power Reactor No. 1 (SL-1) are discussed in this report. The WTR accident occurred on 3 April 1960 and the SL-1 accident occurred on 3 January 1961. This paper briefly describes the facilities and the events relevant to the accidents, with a brief discussion and analysis of the damage incurred. Some of the pertinent implications as to reactor safety are discussed.

The SL-1 was operated by Combustion Engineering, Incorporated for the USAEC at the AEC National Reactor Testing Station, Idaho. The WTR is owned by the Westinghouse Electrical Corporation, Pennsylvania.

The two accidents described in this report are of greatly different proportions. The SL-1 accident was much more serious than the WTR accident. The three crew members on duty at the SL-1 were fatally injured and the recovery of the SL-1 reactor was economically infeasible from a programme standpoint. On the other hand, no one received an over-exposure of radiation as a result of the accident at the WTR and the facility was readily returned to operation.* Both accidents, of course, were thoroughly investigated. The SL-1 investigation is still continuing and probably will continue until mid-summer.

The aspects of the accidents discussed in this report, selected as those of direct interest to the nuclear power industry, are excerpted from reports, already published, by various committees, boards and other persons directly associated with investigation of these accidents.

II. WESTINGHOUSE TESTING REACTOR

A. Background

The Westinghouse Testing Reactor (WTR) is located on an 830 acre tract, approximately 20 miles southeast of Greater Pittsburgh. The surrounding land area usage is predominantly farming (Fig. 1).

The WTR is a pressurized, tank type, light water-cooled and moderated reactor. The primary function of the WTR is to test reactor materials and components. The reactor was designed for 60 MW power operation, although it was originally licensed and operated at 20 MW. The primary coolant system is a recirculating loop in which water flows from the reactor vessel to a surge tank from which it is pumped through heat exchangers to an elevated head tank, 250 ft above the ground. From the head tank, water flows by gravity back to the reactor vessel.

Each fuel assembly has 200 g of highly enriched uranium fuel as aluminium-uranium alloy in the walls of three long concentric cylinders around a central aluminium mandrel tube in which small canned specimens can be irradiated. The uranium-aluminium alloy is aluminium clad: cladding thick-

* The Westinghouse Electric Corporation announced in March 1962 that it was terminating the operation of the WTR because of lack of customer demand.

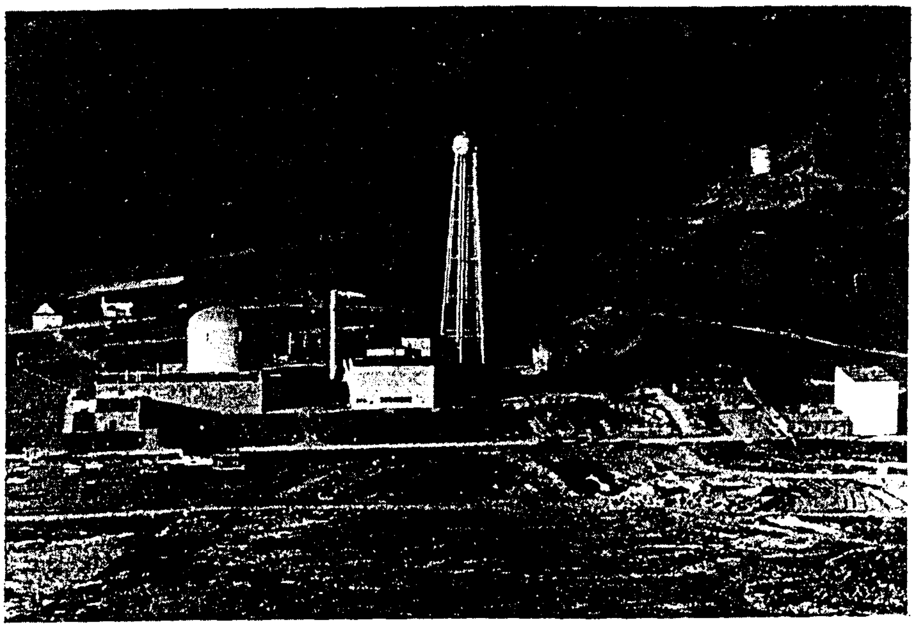

Fig. 1

Westinghouse testing reactor located 20 miles southeast of Greater Pittsburgh

ness is 36 mils; the meat, 52 mils. The fuel tubes or cylinders are 44 in long and the outside diameter of the fuel assembly is 2.5 in. Orifices at both ends distribute the coolant flow through the channels within the assembly and provide some of the static pressure required on the fuel assemblies to prevent boiling at the hot spots.

Fig. 2 shows the relationship of the reactor core to the reactor vessel. A plan view of the core barrel is shown in Fig. 3. The inner hexagon contains fuel elements, test loops, and control rods, while the outer segments are used for experimental purposes.

At the time of the accident, the reactor had been operated up to 45 MW and studies were underway to determine the effect of incipient boiling on reactor stability in anticipation of 60 MW operation. As an initial part of the experiment, tests were conducted to study the effects of bubbling helium through the core. When the accident occurred, a programme was underway to operate the reactor at incipient boiling by reduction in the primary coolant flow, observing the formation of steam bubbles using the same recorders previously tested during the helium bubbling experiment.

B. WTR accident

On 3 April 1960, the reactor had been operating at a steady state at 40 MW with a primary coolant flow of 15 000 gal (US)/min. In preparation for carrying out the reduced flow experiment, reactor power was reduced to 30 MW and appropriate reactor safety circuits were reset to permit reduction of flow to 5 000 gal (US)/min. During the experiment, it was intended

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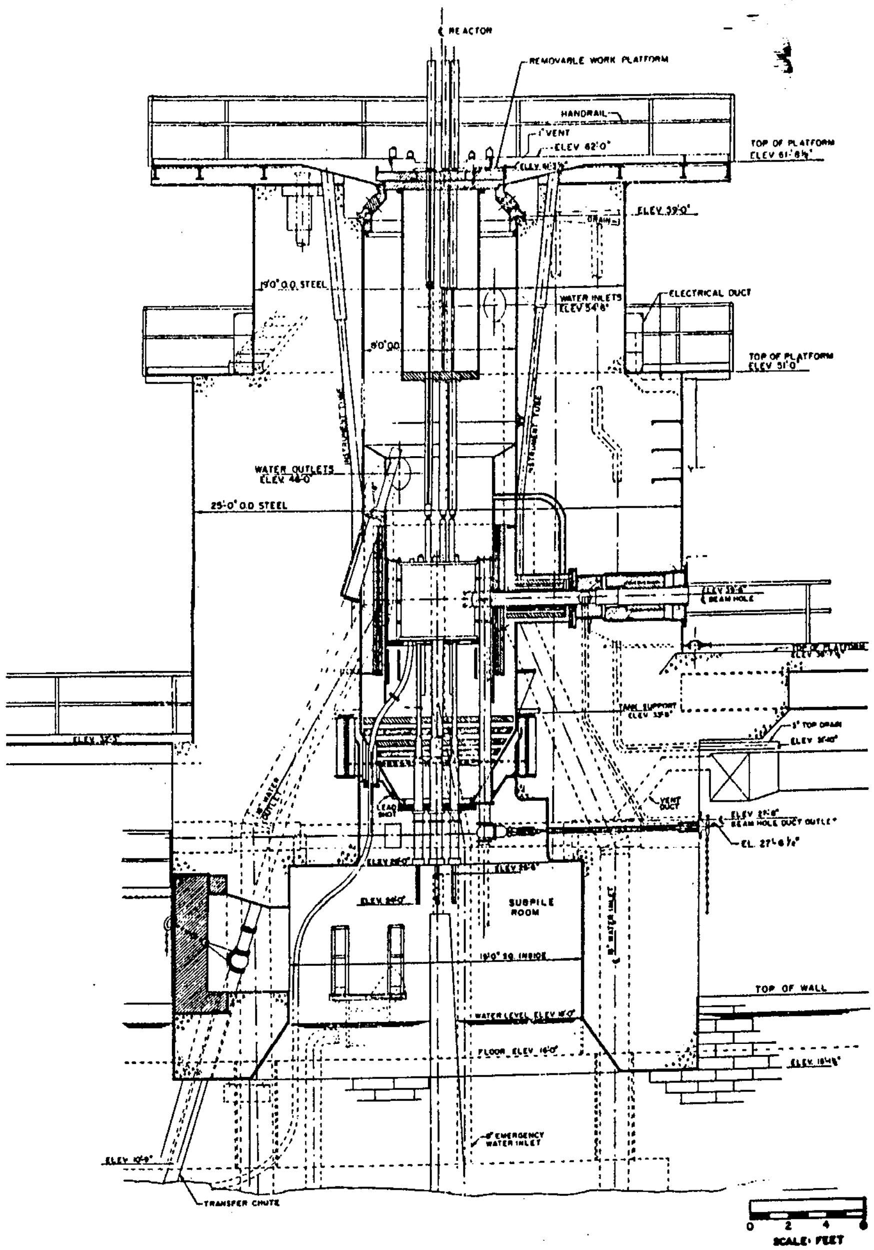


Fig. 2

Vertical drawing of WTR core

to raise the power level gradually, with continuous monitoring of the bubble measuring recorders, until a power level of 45 MW was reached or until boiling was observed.

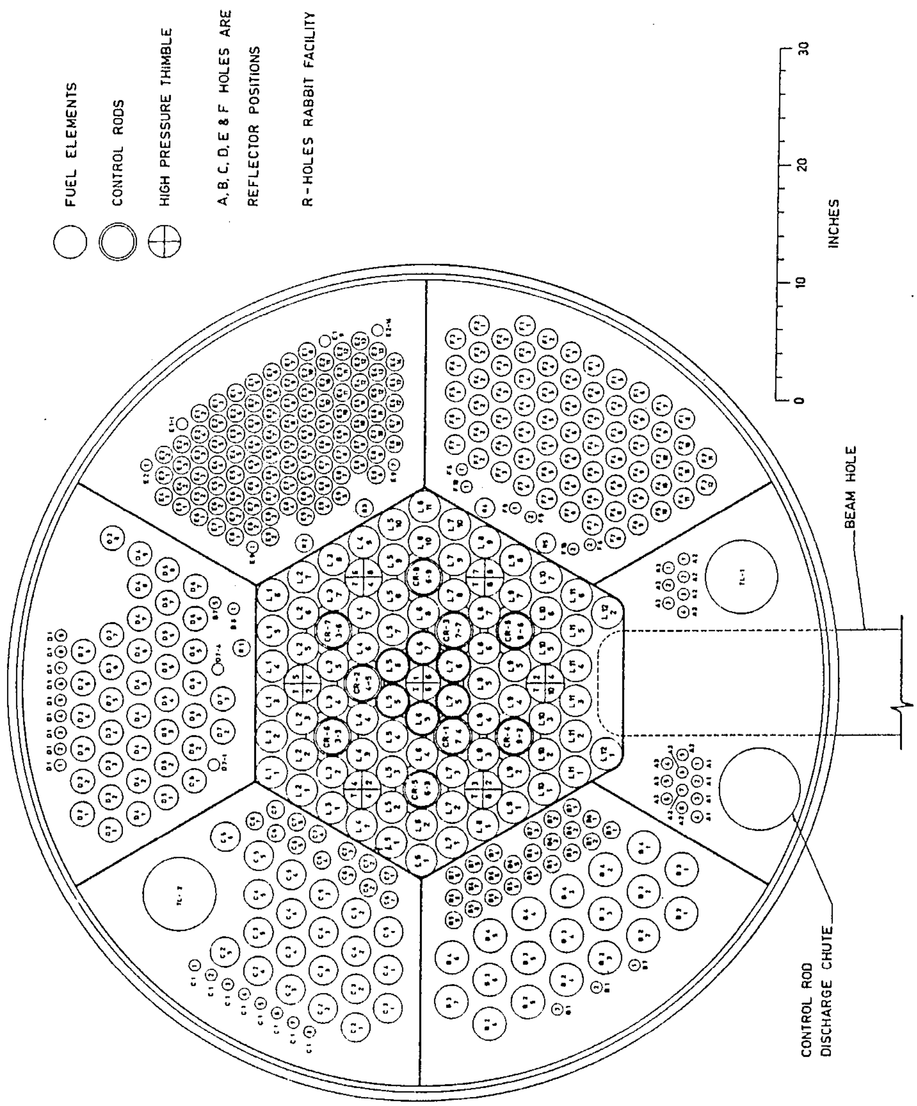
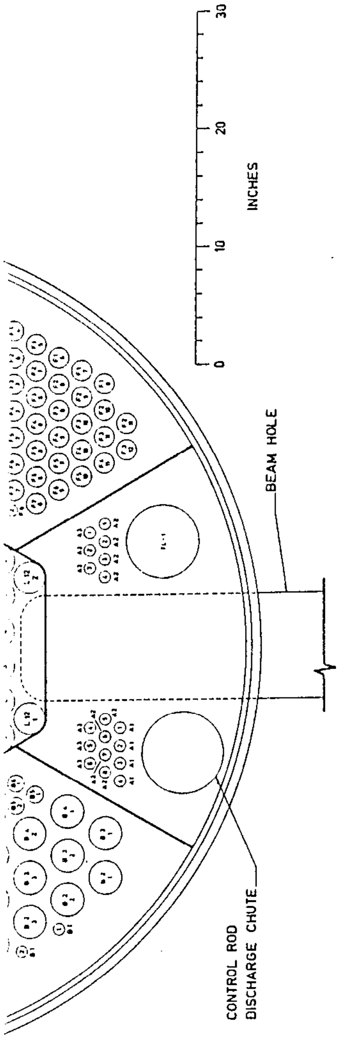


Fig.3

Plan view of WTR core

The reactor flow was reduced gradually to 5 250 gal (US)/min. At 8.20 p.m. the reactor power was increased to 37 MW (calculated). A recording of power levels observed is shown on Fig.4. At 8.33 p.m. the power demand was adjusted to raise the power to 40 MW. At 8.35 p.m. the power level began to drop rapidly, going down to 17 MW over a period of about two minutes for no apparent reason. During this period, the control rod on automatic control withdrew to its upper limit. The other control rods were with-



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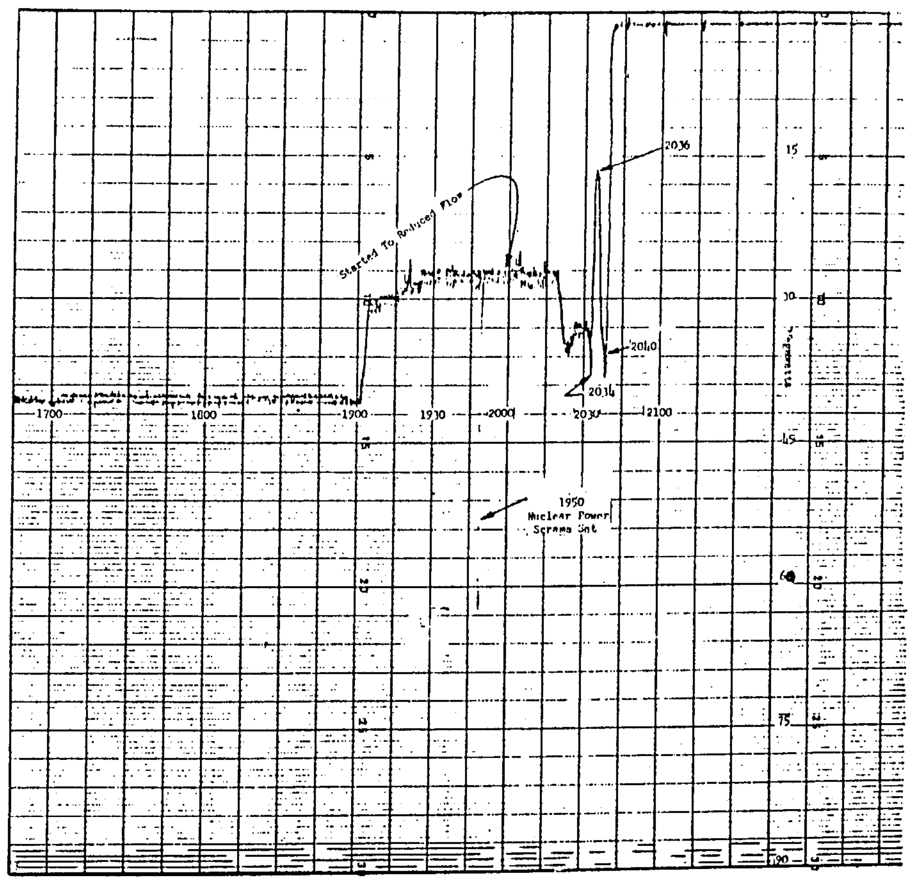

Fig.4

WTR neutron power level recorder chart for night of 3 Apr. 1960

drawn on manual control in order to maintain power, and on reaching approximately 17 MW, power level started to increase on approximately a 60 s period or greater. The power returned to approximately 38 MW. At 8.40 p.m. the radiation detector monitoring the demineralized water supply alarmed. This was followed by further alarms from other radiation monitors within a minute. Power was lowered to 15 MW and at 8.44 p.m. the reactor was scrammed manually as radiation levels continued to rise.

Immediately following reactor scram, personnel on the reactor top were evacuated to the control room. As radiation levels continued to rise on all monitoring channels, a general evacuation was begun to a remote location on the site. Certain operating and health physics personnel remained for a short time to secure the plant and to continue survey work, but were evacuated due to the high radiation levels.

The reactor primary coolant system was left in operation and one of the high pressure test loops set for cool-down. Activation of the stack gas and particulate monitors (located in the Process Building) by external radiation caused automatic recirculation of the vapour container ventilation system. The surge tank vent blower, which sweeps air from the surge tank to the top of the head tank where it is discharged, was left in operation to prevent possible blowback of fission product material into the process area. To prevent further releases of material, personnel returned to the plant to shut down this blower.

The initial radiation survey indicated that gross fission product contamination of the primary coolant system had occurred. The highest reading of 40 r/h was taken at the head tank downcomer at ground level.

C. WTR recovery operations

The major effort was to determine the cause of the failure, get the plant decontaminated and the reactor back into operation. Such problems as water storage and radiation protection occupied a considerable effort and the solution to these type problems governed the pace of the main activities.

By 9 April, decontamination efforts had proceeded sufficiently so that the reactor head was raised one foot for examination and radiation survey. Since the radiation levels close to the head were approximately 1 r/h, the head was replaced pending construction of shields and to prepare washing and decontamination equipment. A system of car-wash brushes was hooked up for continuous scrubbing during the raising of the head (Fig. 5). A 3 in

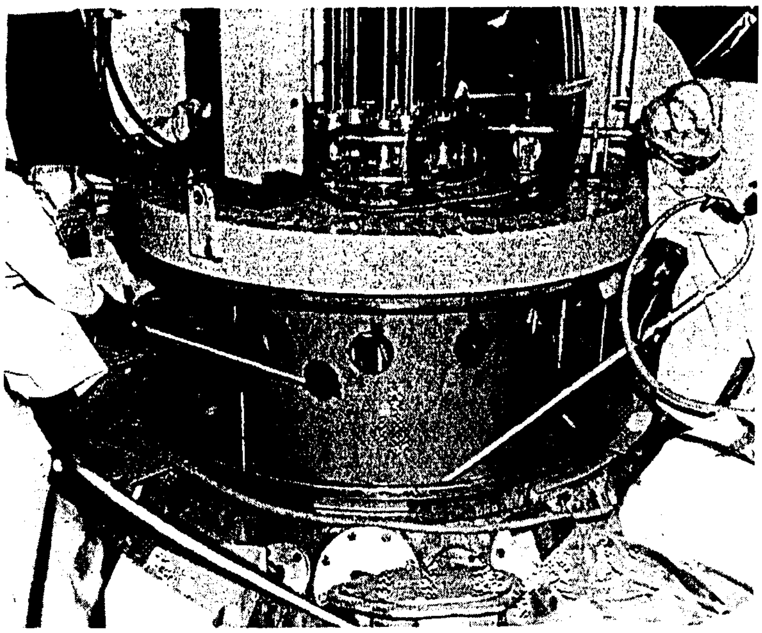

Fig. 5

WTR reactor vessel head removal and decontamination

thick iron shielding platform was constructed to permit visual observation of the core and to begin unloading the core. On 11 April, the head was re-

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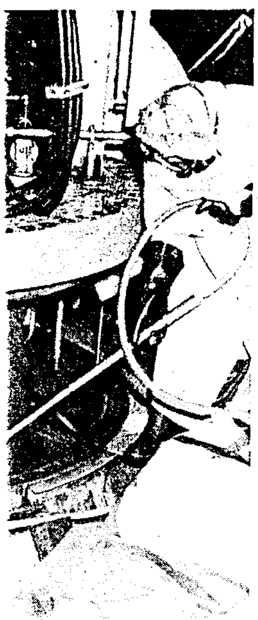

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moved. Fig. 6 shows a photograph of the core taken on this date. No visible damage was apparent at this time.

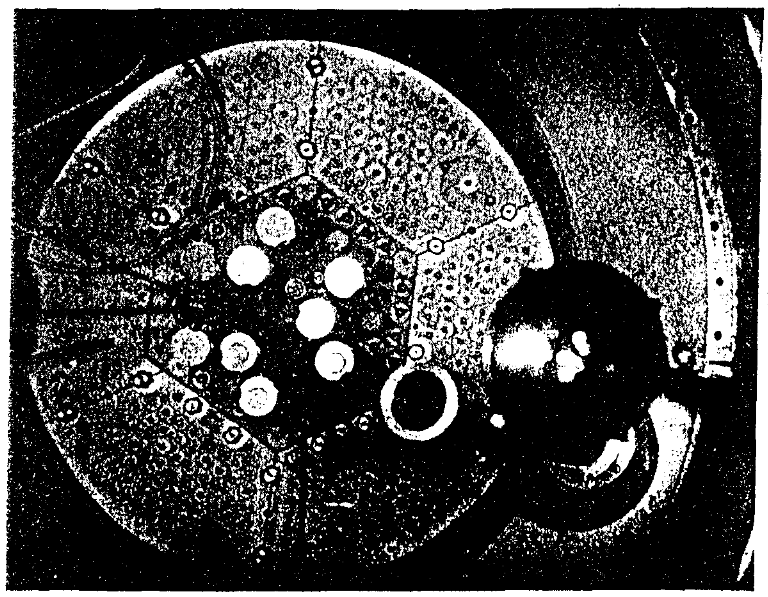

Fig. 6

View of WTR core - 11 Apr. 1960

Fuel unloading then began with elements being removed first from the outside of the core, working towards the middle. Some elements stuck slightly and were removed by a hoist with a 350 lb force limitation. Following removal of all fuel elements but one, which could not be dislodged within the above force limitation, all the control rods and their fuel element followers were removed.

Upon examination, all fuel elements thus removed from the core appeared discoloured but without apparent physical damage. The stuck element was finally removed by a 500 lb force and only the upper third of the element came loose (Fig. 7). The bottom end of the shroud tube appeared to be solidly plugged. Finally by using a specially fabricated core drill, the final portion of the damaged element was removed. A visual examination of the shroud holes and a later check with a sizing tool indicated that the core structure had not been damaged.

D. Accident analysis

The power reduction, shown in Fig. 4 is believed to have occurred as a result of a decrease in reactivity caused by the fuel element failure meltdown and subsequent blockage of the coolant channels. Production of steam and bulk boiling in the blocked element voided the water channels. It was calculated that reactivity loss by voiding the water channels and possibly by the loss of a small amount of fuel is consistent with the reactivity change which caused the power loss from 38 MW to 17 MW.

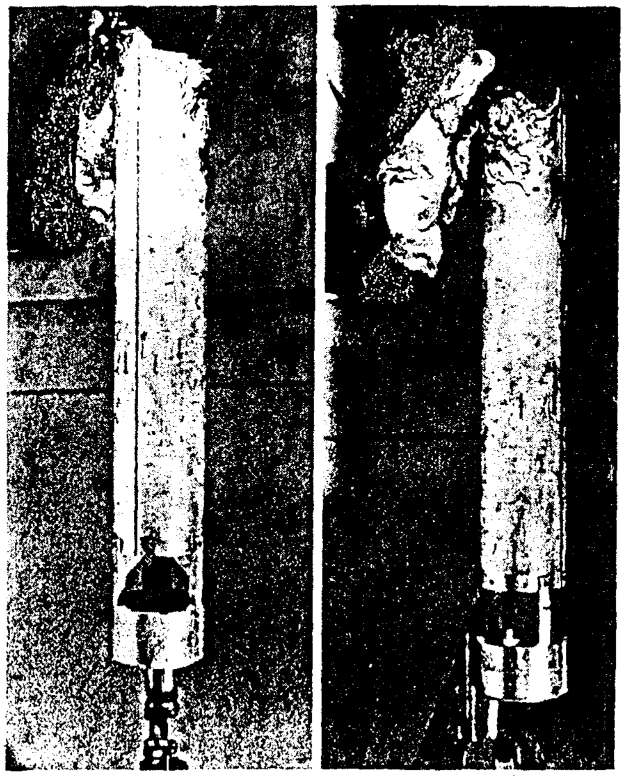

Fig. 7

Damaged WTR fuel element

A close examination of the trace made by a boiling detector Brush recorder, being used during the reduced-flow experiment in progress at the time of the accident, confirmed that the element failed prior to the power loss.

Visual observations of the failed fuel element disclosed some evidence of poor bonding; hence a programme was instituted to reinspect the unused cold fuel elements on hand. A mechanical inspection of these elements disclosed many small deviations from specifications and a few elements with serious bows in the tubes or with visible blisters. An ultrasonic inspection revealed dozens of imperfections (Fig. 8). The defects ranged in size from a few thousandths of an inch to greater than 1 in in diameter.

An experimental check was made to determine whether defects grew upon temperature cycling. No significant change in size or number of the defects was noted.

Thermal hydraulic analysis, using pertinent heat transfer information (reactor power at 38 MW; coolant flow rate, 5250 gal (US)/min , etc.), applicable to the reactor when the fuel element failed, revealed that a burnout type failure of a good element did not occur. The heat transfer calculations indicated, however, a bonding defect greater than $\frac{1}{2}$ in in diameter could account for the fuel element failure.

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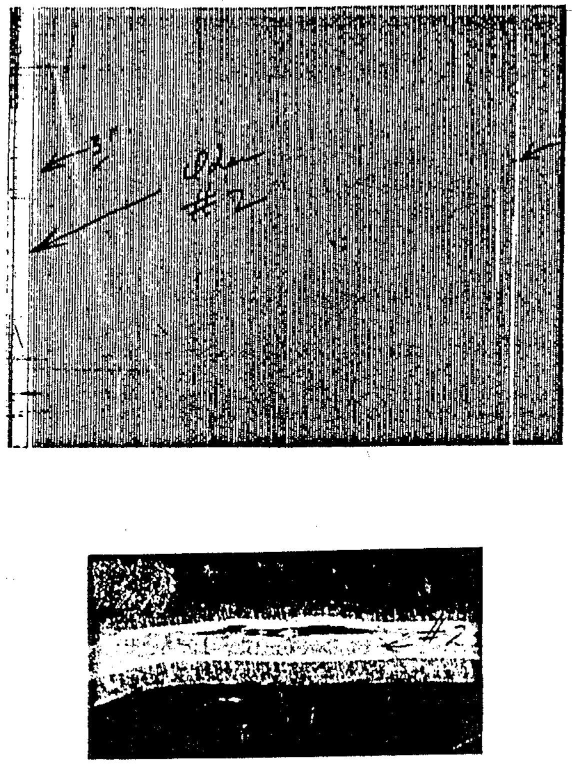


Fig. 8

WTR-typical ultrasonic trace with defect photo

Investigation by the AEC's Division of Licensing and Regulation indicated that either or both of two factors played a major role in the WTR accident : (i) inadequate coolant flow under conditions existing at the time, or (ii) defective metallurgical bonding in the fuel element. Further detailed calculations by the WTR staff indicated that the cause of the failure could not be established beyond a reasonable doubt.

E. Conclusions

A fuel element failure at the WTR on 3 April 1960, resulted in the spread of gross fission products throughout the reactor primary coolant system. The cause of the failure was not established beyond reasonable doubt, but it may be assumed that a normal fuel element operating under the conditions at the time of the accident would not have failed. A strong possibility exists that the failed element was not normal and possibly had a defect greater than $\frac{1}{2}$ in in diameter.

The rapid and spontaneous decrease in power was not recognized by the reactor operator or supervisor as being abnormal. The recovery of the specified power was not consistent with safety of operations. Apparently the fuel element failed prior to the power loss and, therefore, the following increase in power by direct withdrawal of the control rods only aggravated the situation.

Subsequent to the accident, approximately 100 cold fuel elements from the same batch as the ruptured fuel element were reinspected. The results of the reinspection disclosed dozens of defects.

Rigorous inspections cannot be done without adding costs to the fabrication of fuel elements; however, these additional costs are rather insignificant when compared to accident recovery costs.

III. STATIONARY LOW POWER REACTOR NO. 1 (SL-1)

A. Background

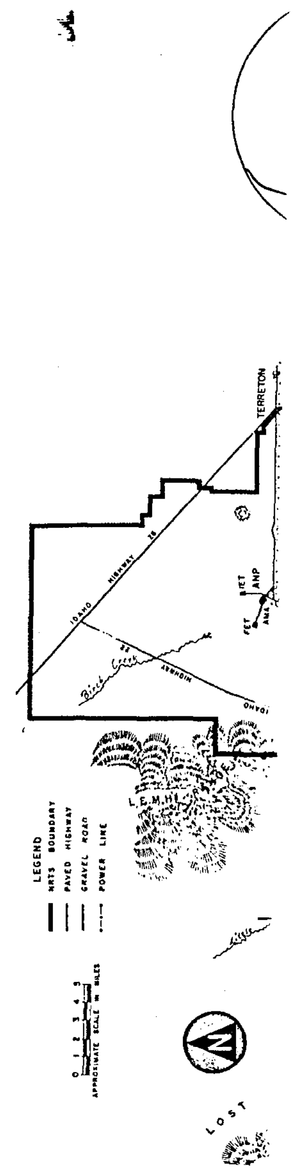
The SL-1 was a direct cycle, natural recirculation boiling water reactor designed for 3000 kW gross thermal capacity and was capable of producing 200 kW net of electricity and 1.3 million BTU per hour for space heat. Work on this plant started in 1955 in response to a Department of Defense request for a small nuclear power plant. The requirement was based on the need to develop such a plant for future use at remote military installations.

Site work began in the fall of 1956; plant construction started in 1957 and initial criticality was achieved in August 1958.

Argonne National Laboratory, the prime contractor, performed the initial criticality and start-up tests and successfully completed a 500 h, full power plant performance test in December 1958. In February 1959, the permanent operator (Combustion Engineering, Inc.) assumed responsibility for the operation of the SL-1. After start-up the SL-1 was used to furnish operating experience, develop plant performance characteristics, obtain core burn-up data, train military personnel in plant maintenance and operation, and test improved components planned for use in subsequent reactors of this type.

The SL-1 site is located at the National Reactor Testing Station about $\frac{3}{4}$ mile north of Route 20 (Fig. 9). Site facilities consisted of the reactor building, an adjoining support building which contained the control room, and miscellaneous service buildings (Fig. 10). The majority of the plant equipment was located in a cylindrical steel reactor building $38\frac{1}{2}$ ft in diameter having an over-all height of 48 ft. This building was made of steel plate, most of which had a thickness of $\frac{1}{4}$ in. Access to the building was provided by ordinary doors. The building was not a pressure-type containment shell as would have been used for reactors located in populated areas. Nevertheless, the building was able to contain most of the radioactive particles released by the explosion.

The building was erected on dummy support piles to simulate the type of construction that would be used in the Arctic, in the permafrost area, where the whole structure would be supported by piles (Fig. 11). The reactor vessel, fuel storage wells, and demineralizers were located in the lower third of the building and shielded with gravel. Gravel was used because this was a material that was readily available at the remote sites where location of such reactors was planned. A recirculating, air-cooler condenser was located in the upper third of the building. The middle third of the building contained the turbine generator, feedwater equipment, and shielding blocks located around the reactor pressure vessel head. These shielding blocks were movable by an overhead crane, permitting access to the pressure vessel head and control rod drive mechanisms.



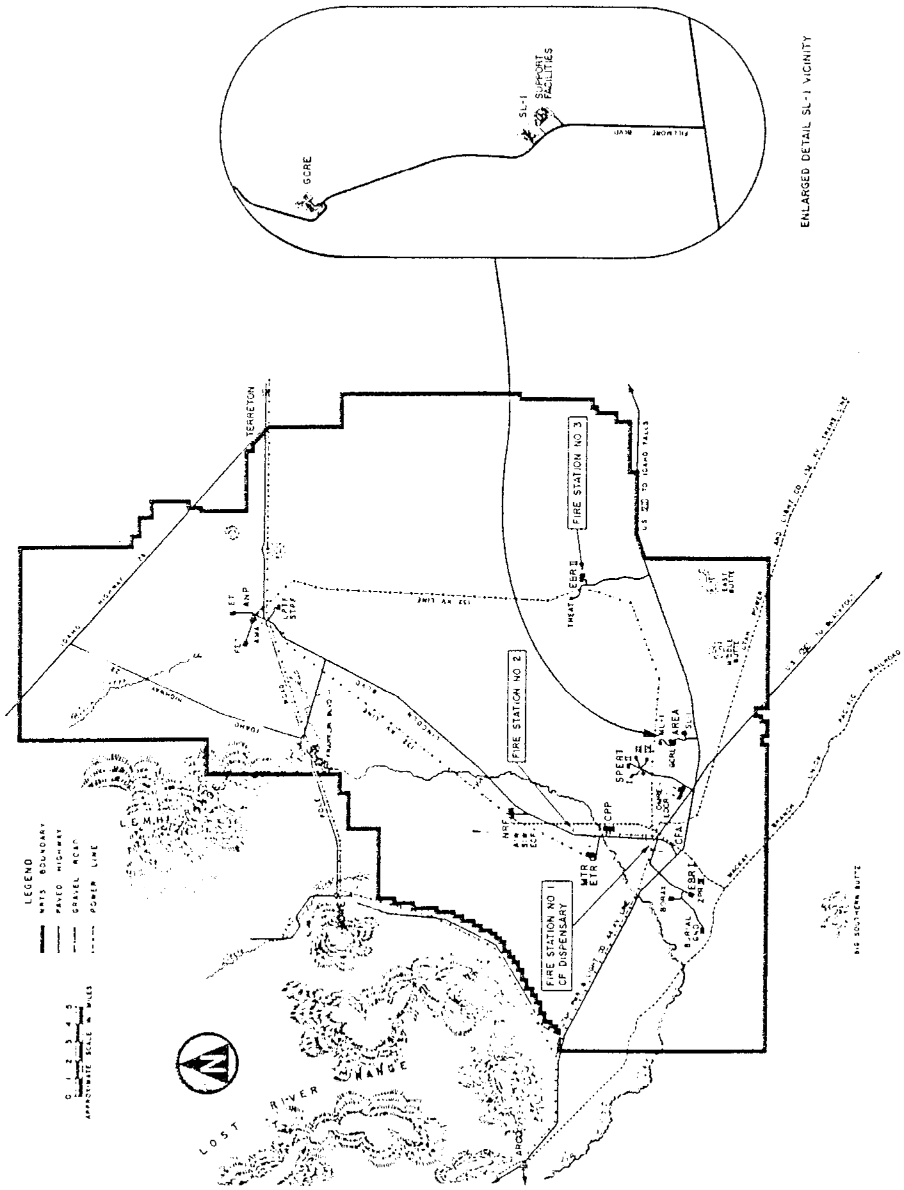


Fig. 9

National reactor testing station

The reactor core was located near the bottom of the vessel; above was the chimney section formed by the control rod shrouds (Fig. 12). Each of the five control rods was connected to a vertical extension rod and a rack which was driven by a pinion gear in the control drive mechanism located

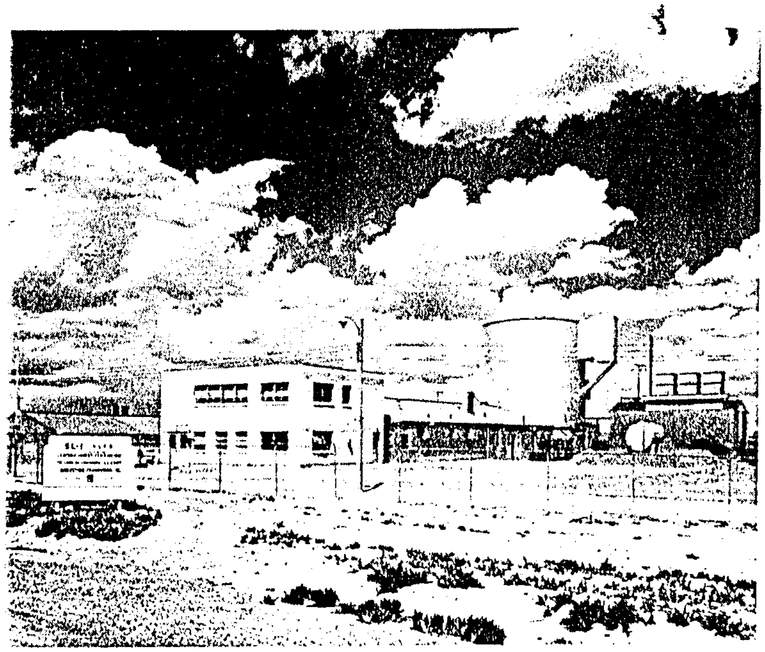


Fig. 10

General view - SL-1 facility

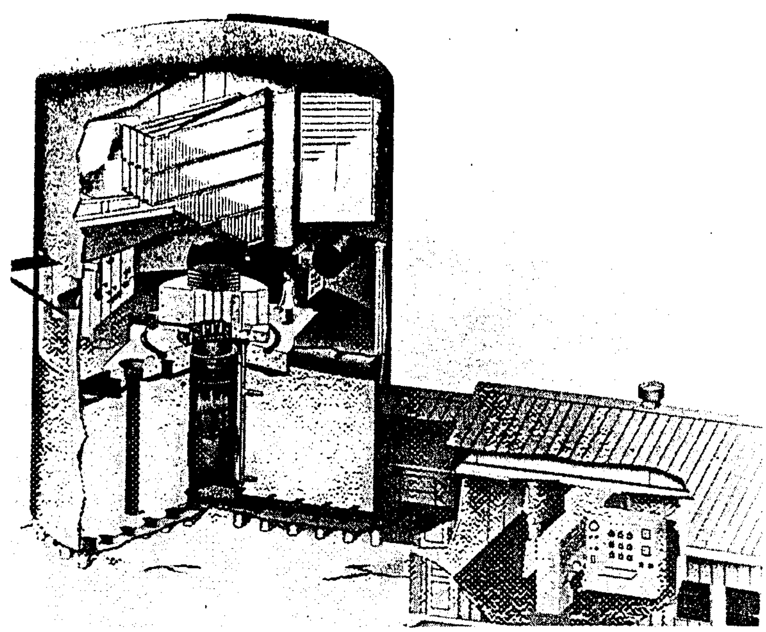


Fig. 11

SL-1 plant perspective

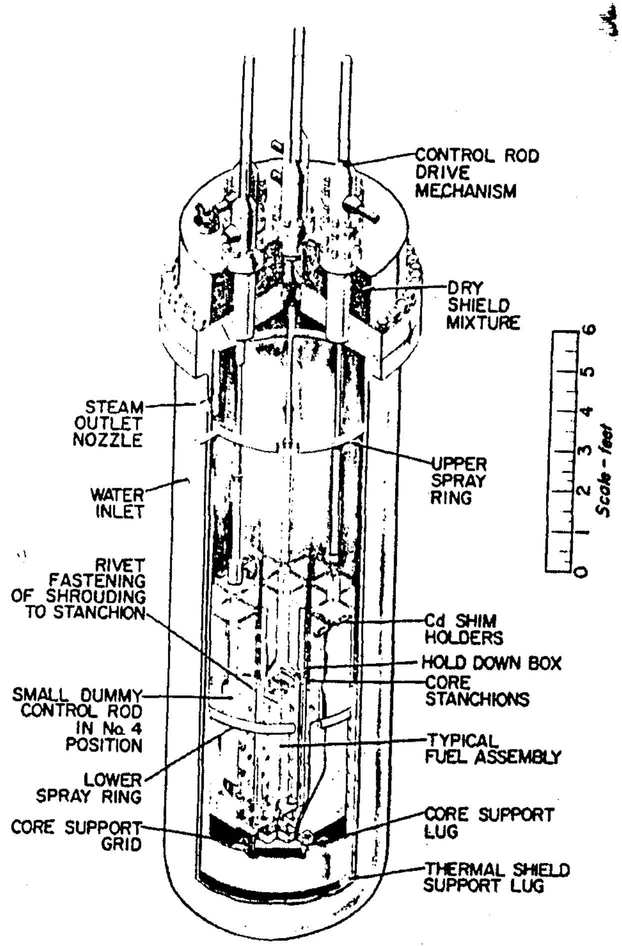


Fig. 12

SL-1 reactor perspective

on the head. Each pinion gear was driven by a horizontal shaft which extended through a pressure seal in the housing of the drive mechanism and through surrounding shielding blocks to a motor located on the outside. Over the head of the vessel was a sheet metal enclosure filled with metal punchings, gravel and boric oxide to provide shielding. A top shield cap rested on the side shielding blocks.

The core structure was built for a capacity of 59 fuel assemblies, one source assembly, and 9 control rods of which 5 were cruciform rods and 4 T rods. The core in use, however, had 40 fuel elements and was controlled by 5 cruciform rods. The control rods were made of 60 mil thick cadmium, mechanically clad with 80 mils of aluminium. They had an over-all span of 14 $\frac{1}{4}$ in and an effective length of 32 in (Fig. 13). The 40 fuel assemblies were composed of 9 fuel plates each (Fig. 14). The plates were 120 mils thick consisting of a 50 mil uranium-aluminium alloy "meat" and 35 mils of X-8001

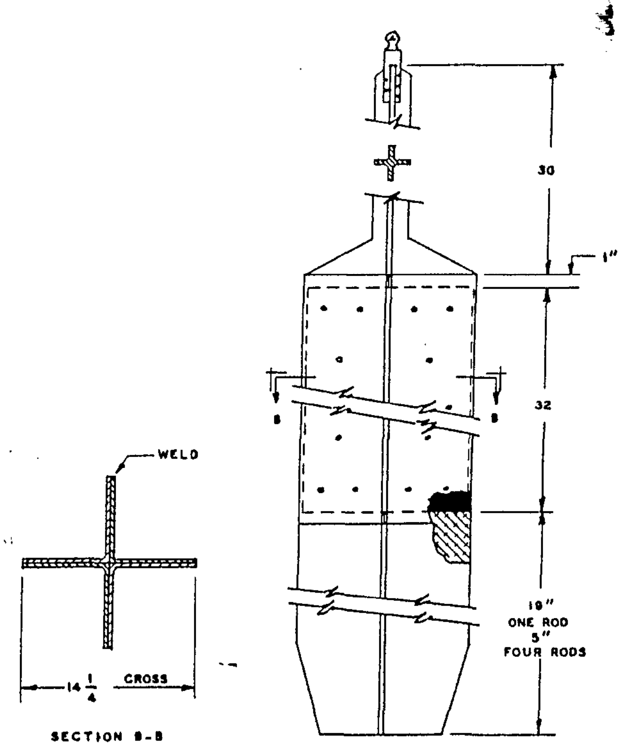


Fig. 13

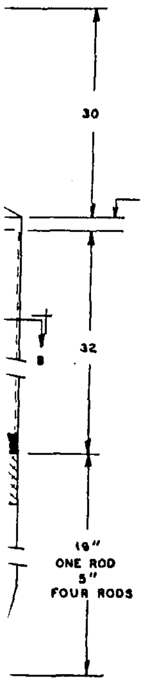
Cross type control rod

aluminium cladding. The meat was 25.8 in long and 3.5 in wide. The water gap between fuel plates was 310mils. The initial loading of the 40 assembly core was highly enriched and contained 14 kg of uranium-235.

On each of the 16 fuel assemblies in the centre of the core (Fig. 15), a full length burnable poison strip was spot welded to one side plate, (shown by dashed lines) and a half length strip to the other side plate (shown by solid lines). The remainder of the fuel assemblies had a full length strip only on one side plate. The strips were aluminium-nickel, containing boron-10. The half length strips were 21 mils thick, and the full length strips, 26 mils thick. The core contained a total of 23 g of boron-10 as burnable poison.

The fuel was calculated to provide about 15% excess reactivity (Fig. 16). The burnable poison was calculated to provide negative reactivity of 11.2%. Reactivity of about 10% was expected to be burned out in 4 ½ yr at normal power operation. The fission products were expected to provide additional negative reactivity of up to 2% over core life. The combined excess reactivity (or the reactivity held down by the rods) was calculated to be 3% at beginning of life, rising to over 3 ½% in just under one year, then decreasing gradually yielding a calculated life of over 3 yr.

At the time of the accident, the SL-1 had been in operation for over 2 yr. The reactor had produced 931.5 MWd of thermal energy which was approximately 40% of the design life of the core.



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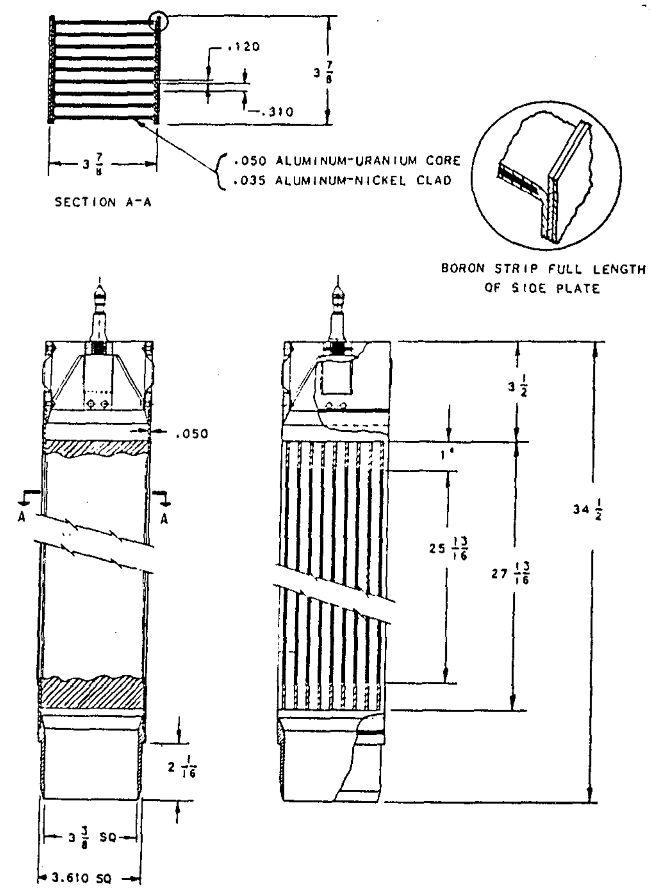


Fig. 14

Fuel element

B. SL-1 accident

On 23 December 1960, the reactor was shut down for routine maintenance, instrument calibration, installation of auxiliary system valves, minor plant modifications, and installation of flux wires in the core. During the period 27-30 December 1960, the maintenance, calibration, and modification work was performed. Work on the installation of the flux wires started after midnight on the morning of 3 January 1961. This work involved moving the shielding blocks back from the reactor, raising the water level to the top of the reactor vessel, removing selected control drive mechanisms, and inserting the 44 flux wires into predesignated water channels within the fuel assemblies. The flux wires were aluminium, containing cobalt-aluminium alloy slugs, and were to be used to measure flux distribution within the core as part of an investigation of reactor core power distribution. By 4 p.m. on 3 January 1961, installation of the flux wires was completed. The three-man, 4 to 12 p.m. shift on 3 January 1961, was directed to pump the water

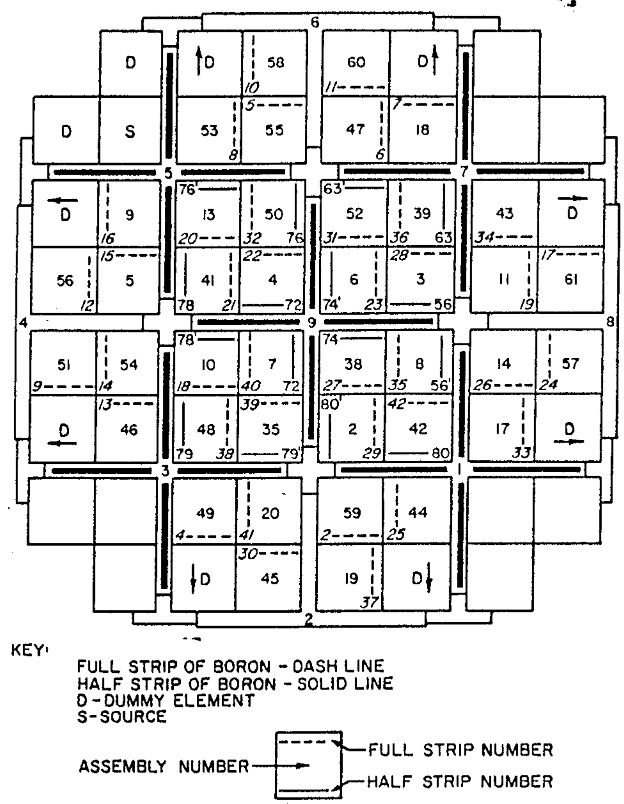


Fig. 15

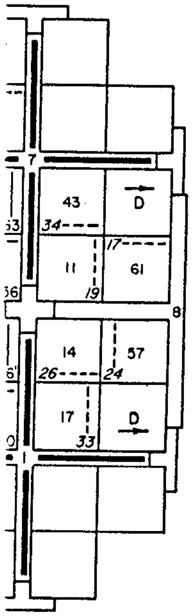
SL-1 loading for 40 element core

down to the normal operating level, install the shield plugs in the top head of the reactor pressure vessel (around the control rod extensions), reassemble the control rod drive mechanism, replace the shield blocks, and connect the motors in preparation for resuming operations the following morning.

The operating log disclosed that the crew had pumped the water down to a level 2 ½ ft below the reactor head. The recovery evidence obtained so far indicates that the crew had installed all the shield plugs and was completing the reassembly or "hook up" of the central rod when the accident occurred at 9.01 p.m. The three crew members on duty, working in the reactor room, received fatal injuries from the explosion. Two crew members died instantly; the third, a few hours later.

C. SL-1 recovery operations

The post-accident SL-1 investigation and dismantling operations, which are expected to be completed by midsummer 1962, consisted of three phases. Phase I (3-9 January 1961) included the emergency operations mainly con-



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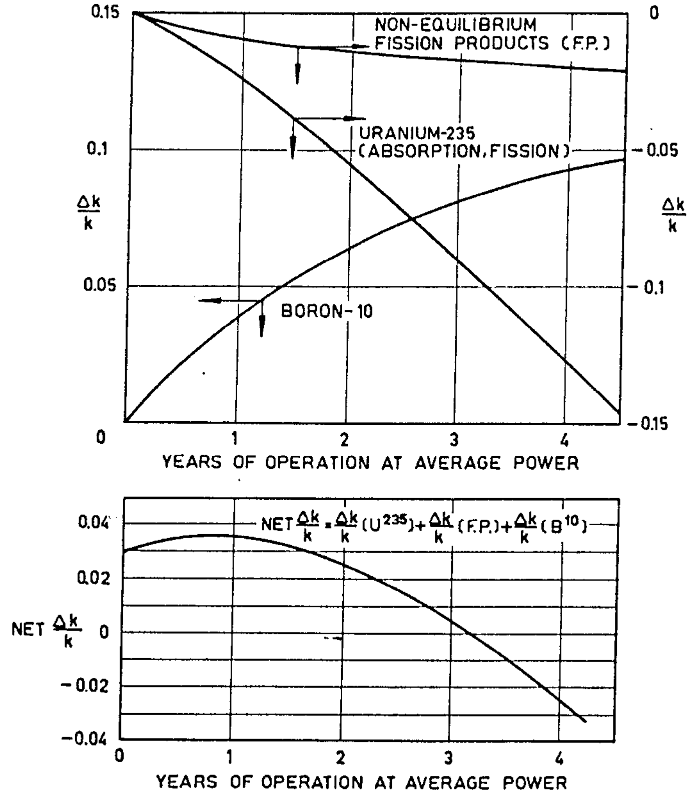

Fig. 16

Reactivity variation during SL-1 core lifetime

cerned with the recovery of the victims from the SL-1 reactor building. Efforts also were made to determine whether a nuclear excursion had taken place and the status of the reactor facility. Phase II (9 January 1961 through 21 April 1961) consisted basically of efforts to determine the gross extent of the accident and the nuclear status of the SL-1 reactor. Phase III (21 April 1961 to midsummer 1962) consisted of the detailed investigation of damaged reactor components and effects of the excursion in an effort to determine the cause of the accident. The clean-up and dismantling operations of the SL-1 site also took place during this phase.

1. Phase I (3-9 January 1961)

During the emergency operations, it was determined that a neutron excursion had taken place. The following are some of the analytical results which supported this conclusion:

(a) Bare gold foils from a Hurst dosimeter which was located near the entrance to the operating floor indicated a neutron exposure of 1.2×10^8 thermal neutrons per cm^2.

(b) A brass screw taken from a cigarette lighter indicated a neutron exposure of 9.3×10^9 thermal neutrons per cm^2.

(c) A brass watch band buckle indicated a neutron exposure of 1.8 x 10^{10} thermal neutrons per cm^2.

(d) Gold jewelry indicated a neutron exposure of 9 x 10^9 thermal neutrons per cm^2.

(e) Analysis of samples taken from the clothing of the victims indicated the presence of uranium and strontium--quantitative analysis of these samples yielded a yttrium-91 activity of 2.4 $\times$ 10^4 deceys per minute per militer.

(f) Soil samples from within the area, clothing samples from personnel that entered the reactor room, and air samples from the control room all exhibited a gross fission product spectrum.

Photographs (Fig. 17 and 18) were taken of the reactor head area to

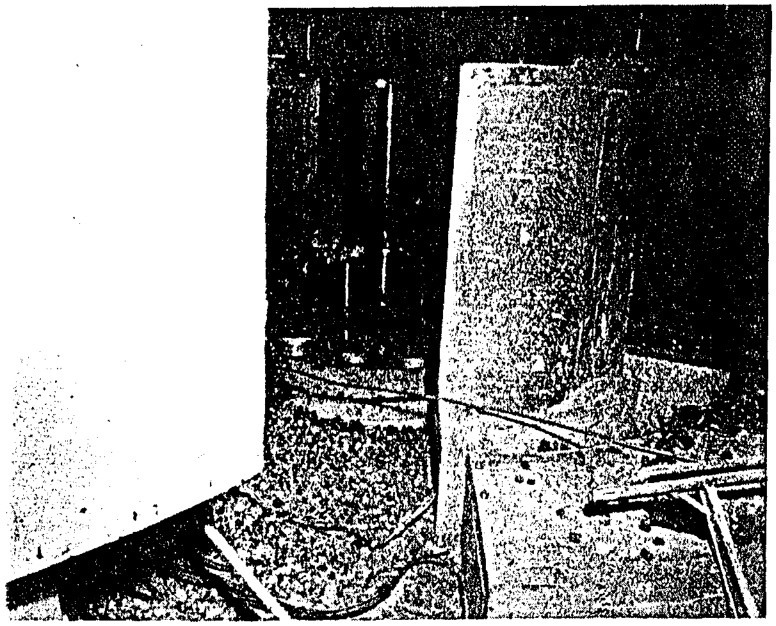

Fig. 17

View of SL-1 reactor head area after accident

assist in the recovery of the third victim. The photographer, permitted to enter the building for only 30 s, took these photographs in a 500-1000 r/h field. Fig. 17 shows that the metal cover of the pressure vessel head shield was forced upward and the metal punchings and gravel forced out covering the floor area in the foreground. Control rod racks are protruding from nozzles 1 and 7 (see Fig. 19 for nozzle positions) and are about $\frac{1}{2}$ ft further out than they would normally be during a shut-down. Across the top of the head is a shield plug with a portion of the control rod extension shaft still in this plug, later identified as the No. 9 shield plug. Fig. 18 shows the various control rod drive components which had not been assembled.

Other photographs taken during this phase of operation indicate physical damage, other than to the pressure vessel and core, was confined to the area directly above the reactor. Tools lying on the shielding blocks were a neutron exposure of 1.8

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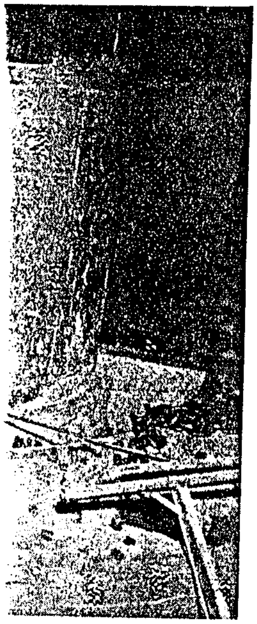

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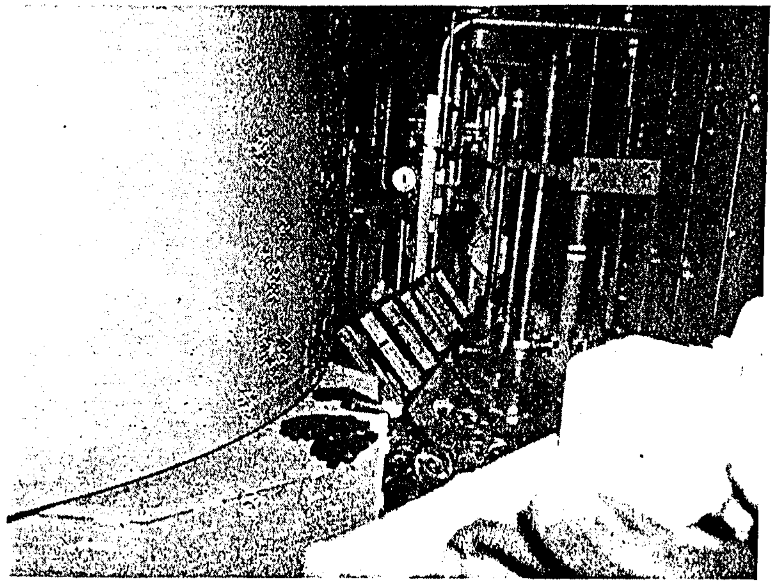

Fig. 18

View of SL-1 reactor room after accident

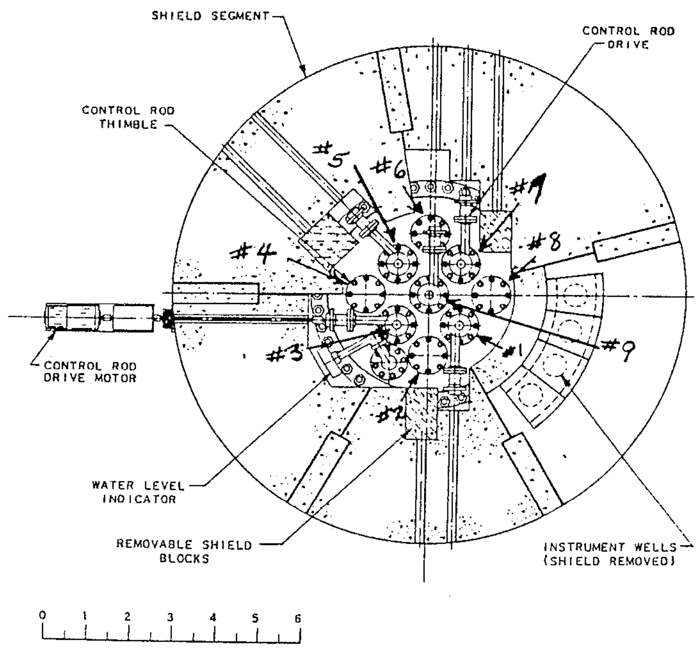

Fig. 19

Control rod drives and top shielding

essentially unmoved and only one light located directly above the reactor head was broken.

As mentioned previously, the radiation levels in the vicinity of the reactor head were 500-1000 r/h and at the building walls the levels were approximately 100 r/h.

2. Phase II (9 January-21 April 1961)

With emergency operations completed, no one was allowed to enter the reactor building due to the high radiation fields and because the nuclear status of the reactor had not been determined. It was not then known whether or not water was in the vessel, whether portions of the reactor fuel were precariously balanced and might be dislodged into another nuclear configuration, etc. Hence all penetrations into the reactor room.were accomplished remotely.

For these remote penetrations, several devices were used which disclosed valuable though not always conclusive information.

A mock-up of the reactor building, reactor head, vessel, etc. was constructed whereby the recovery crews could practise the intricate manipulations required to handle photographic and television cameras and associated lighting in order to view the reactor head area and inside the pressure vessel. Also various probes were used to measure the radiation fields in the reactor building and inside the pressure vessel, the temperature over the reactor head and core, and the water level in the pressure vessel.

The specially shielded crane with a movable boom used throughout the remote operations is shown in Fig. 20 performing an entry in the reactor

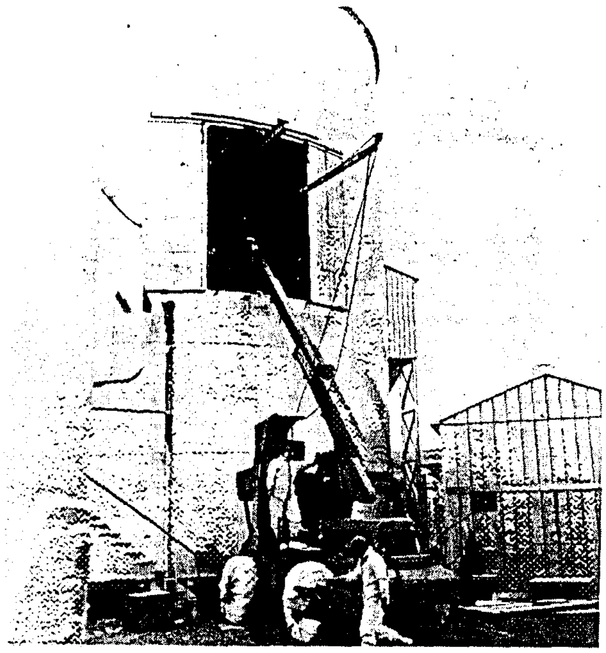

Fig. 20

Performing entry into the SL-1 reactor building directly above the reactor

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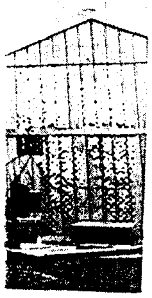

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building. Photographs of the various cameras and probes used during this phase are shown in Fig. 21 to 26.

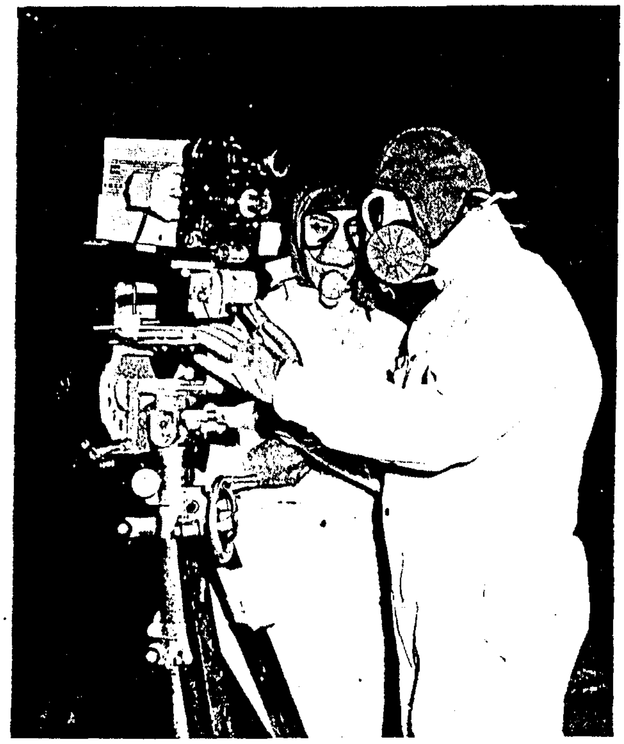

Fig. 21
Recovery crew adjusting TV camera

Fig. 27, a frame from the first movies taken directly over the reactor head, indicated that six nozzles were open to the atmosphere and that nozzle No. 8 appeared to be free of any obstructions. Hence most of the remote penetrations into the pressure vessel were made through No. 8. Although the television shots were not too clear, valuable information was obtained as to the condition of the core. Photographs taken of the core using a Minox miniature camera added significantly to our knowledge of the condition of the core (Fig. 28).

On 15 April 1961, the shielded miniature camera assembly was used in conjunction with a chemical probe which reached within 3 in of the bottom of the pressure vessel (Fig. 28). The probe gave no indication that water was present in the vessel and hence the reactor was declared nuclearly safe as long as the core remained unmoderated.

Aside from determining the nuclear status of the reactor, significant information was obtained from the numerous photographs, movies, etc. taken inside the pressure vessel. It was determined that the four outside control rods (1, 3, 5, and 7) were essentially in place and that the central

Fig. 22

Minox camera and shielding assembly

rod, No. 9, had been ejected upward and was lying across the top of the core. These observations clearly indicated that the core and core structure were severely damaged.

3. Phase III (21 April to present)

With the nuclear status of the SL-1 reactor known, the recovery operations could proceed more deliberately. By the end of April, radiation levels within the reactor building had decayed to approximately 200 r/h. The primary objective of this phase was to determine the cause of the accident. Complete photographic and radiation surveys were a necessity before removing debris and reactor components from the reactor building. As these surveys progressed, some of the reactor components (excluding those inside the pressure vessel) were removed from the building. Limited personnel access to the reactor building was eventually allowed when the radiation fields became better known. A hole cut into the side of the reactor building at the fan room level (above the reactor room) permitted access to that area for completion of surveys of the interior of the building.

Careful examination of the photographs taken and the debris recovered

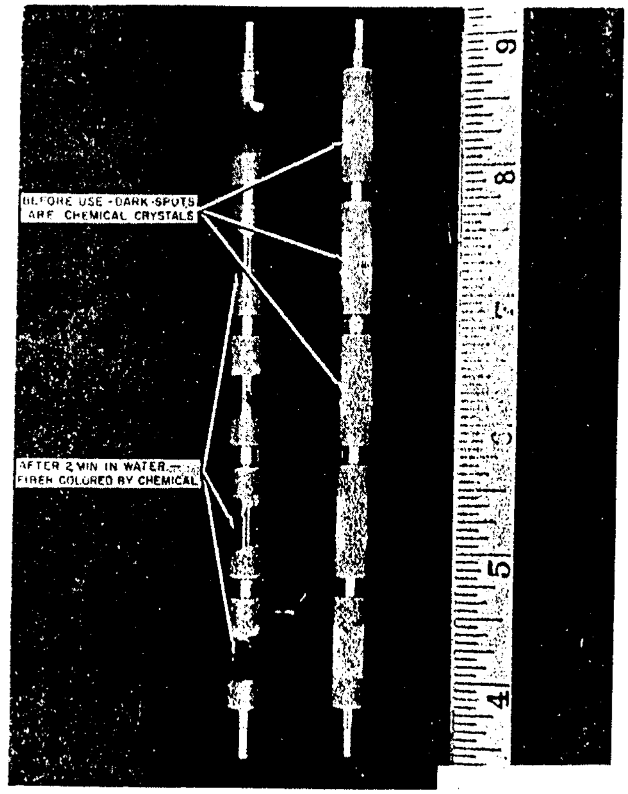

Fig. 23

Chemical water prove (section)

from the reactor room led us to believe that the pressure vessel as a whole might have been physically dislocated upward as a result of the nuclear excursion. The most notable evidence which supported this belief was the presence of block insulation lying on the reactor room floor (Fig. 29). This insulation was originally wrapped around the pressure vessel and held in place by a $\frac{1}{4}$ in steel jacket. The most likely explanation to account for such large pieces of insulation on the reactor room floor was that the vessel must have been forced upward. Early in November 1961, a trial lift of the pressure vessel confirmed that the vessel had indeed been projected up by the explosion, shearing the steam nozzle and other pipes (Fig. 30), and had then fallen back approximately into its normal position.

Before the pressure vessel was lifted, a $2\frac{1}{2}$ in hole was drilled into the side of the reactor building and through the wall of the pressure vessel at a level below the core. Through this hole, photographs were taken, using a boroscope, which disclosed severe damage to the lower core structure (Fig. 31). Also, four of the five control followers were identified, confirming that the four outside rods were essentially fully inserted into the damaged core.

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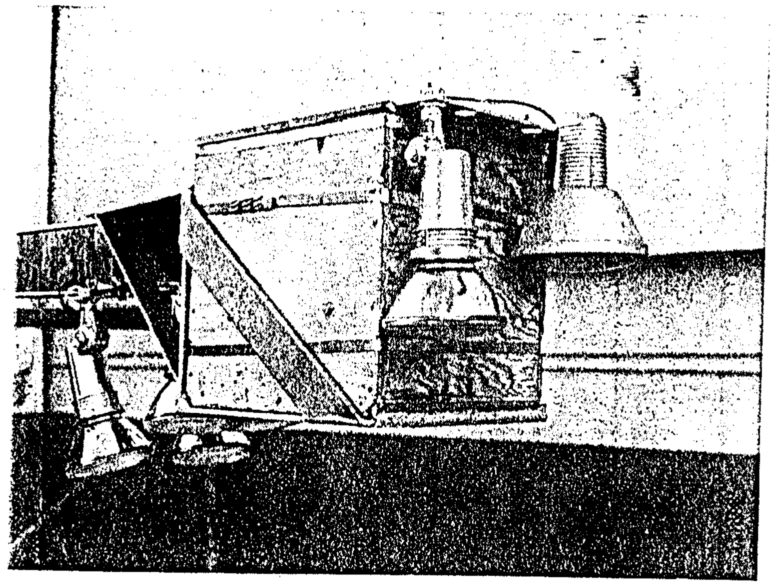
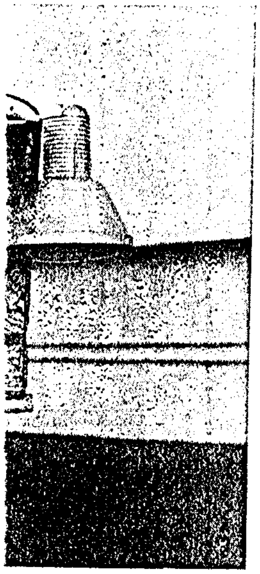


Fig. 24

Shielded movie camera mounted on crane

Fig. 25

Special barrel mounted television camera



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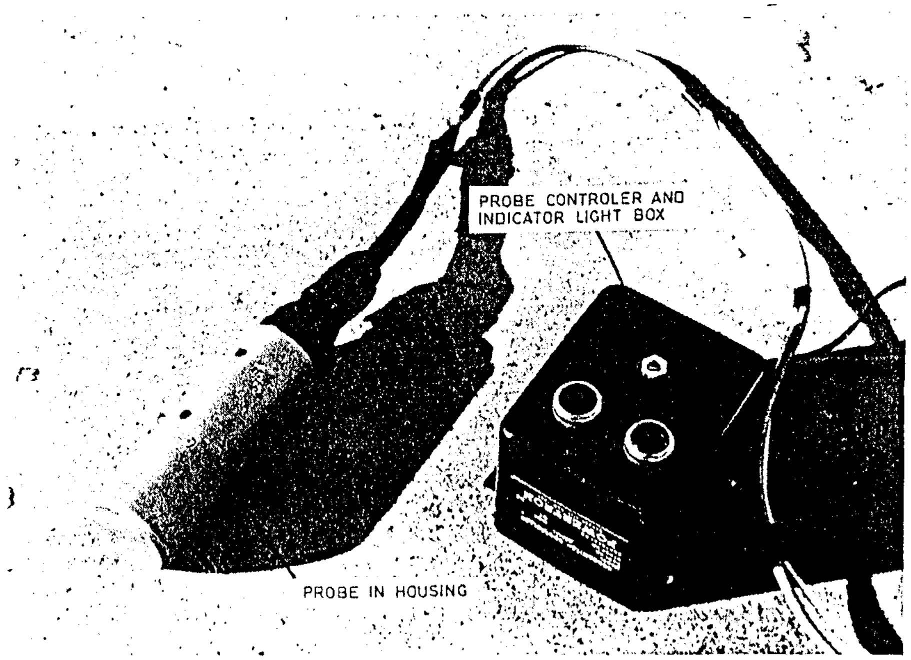

Fig. 26

Ultrasonic probe and housing for water detection

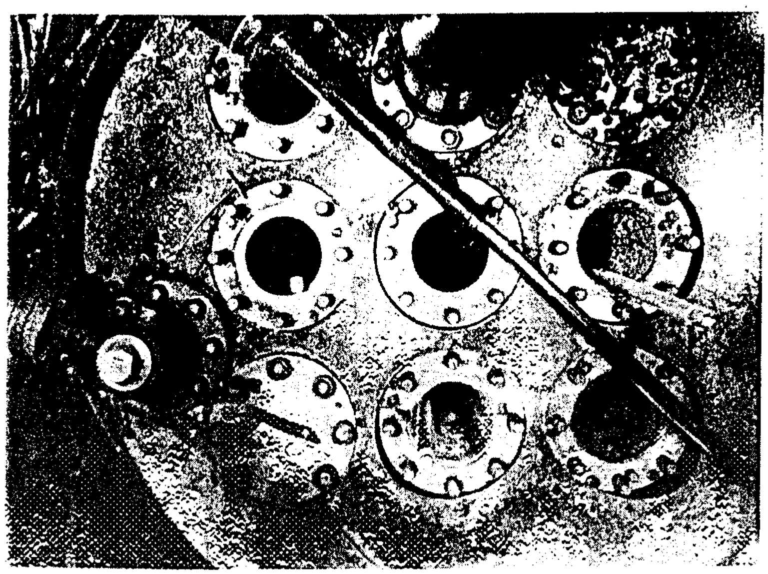

Fig. 27

View of SL-1 reactor vessel head after incident

Fig. 28

Photographic evidence of chemical probe penetrating core structure through control rod shroud No. 8

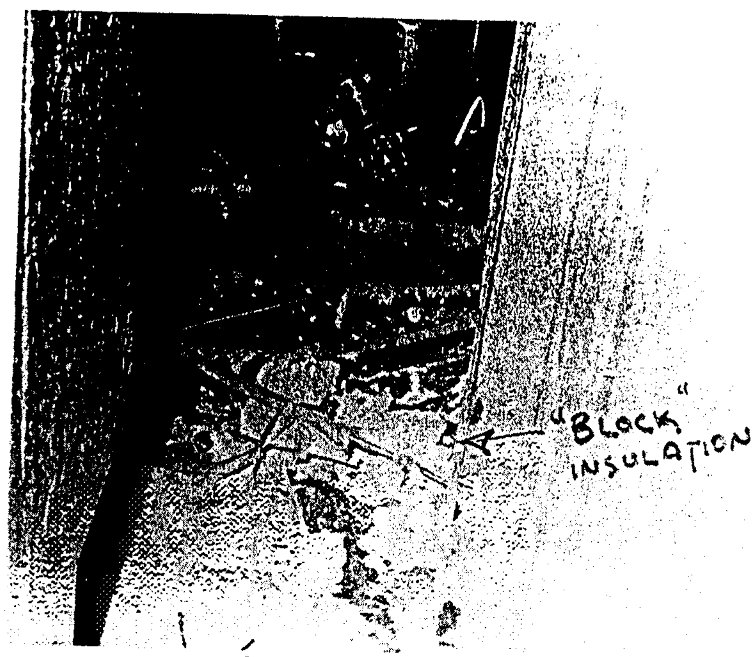

Fig. 29

Pieces of block insulation on SL-1 reactor operating floor

From June through November 1961, clean-up operations proceeded rather slowly since water or any other moderating material could not be used to decontaminate the interior of the reactor building. Vacuum cleaners, ıre through control rod shroud No. 8

BLOCK INSULATION

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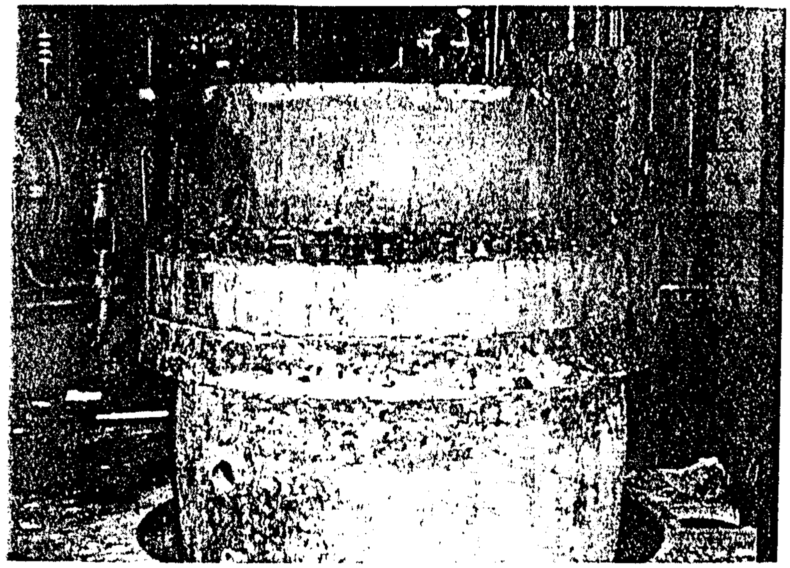

Fig. 30

Trial lift of pressure vessel showing distorted flange, bulged vessel and sheared steam line

a remotely controlled electromagnet, and manual labour (on a rapid, large scale turnover rate to avoid over-exposure to any individual) were the techniques used to remove the debris from the reactor building. The radiation levels within the building were substantially reduced using these techniques and by placing several thousand pounds of steel and lead sheet and lead shot over the reactor head.

By late November 1961, all preparations for removal of the pressure vessel, with the core left inside as it was, were completed. On 29 November 1961, the pressure vessel was successfully removed from the reactor building and transported in a large concrete shipping cask to a large disassembly hot cell located 40 miles north of the SL-1 site (Fig. 32).

In January 1962, preliminary hot cell examination of the pressure vessel and core disclosed that the vessel was not ruptured but was bulged about 4 in in diameter just below the head flange and was bulged about 1 in above and below the core (Fig. 33). The reactor head nozzles were also found to be bulged. The pressure vessel flange was so distorted that the head could not be raised off the head bolts after the nuts had been removed. It was necessary to force the head upward using wedges. After the reactor head was removed, it was clear that the central rod, within its own shroud, was entirely out of and above the core. The rod with shroud was lying approximately 45° to the horizontal across the top of the core (Fig. 34). When the central rod and shroud were removed, it was quite evident that the centre of the core suffered severe melting and destruction (Fig. 35).

Dismantling the reactor building and decontamination of the SL-1 site proceeded quickly with the major sources of radiation removed. At the present time, the building components, the gravel shield and most of the equip-

Fig.31

Underside of core showing location where core has been lifted from support bracket

ment in the building are being buried at a site approximately ¼ mile from the SL-1 site. The remaining buildings on the site are being restored for future use.

IV. PRE-ACCIDENT CONDITION OF THE SL-1 REACTOR

A. General

This section is concerned with certain circumstances and conditions of the SL-1 reactor which are relevant to discussions of the accident. There is no evidence to indicate that any of these circumstances had a direct relationship to the SL-1 accident. Each of the factors mentioned has a logical explanation as to why it existed, though there has been debate as to whether

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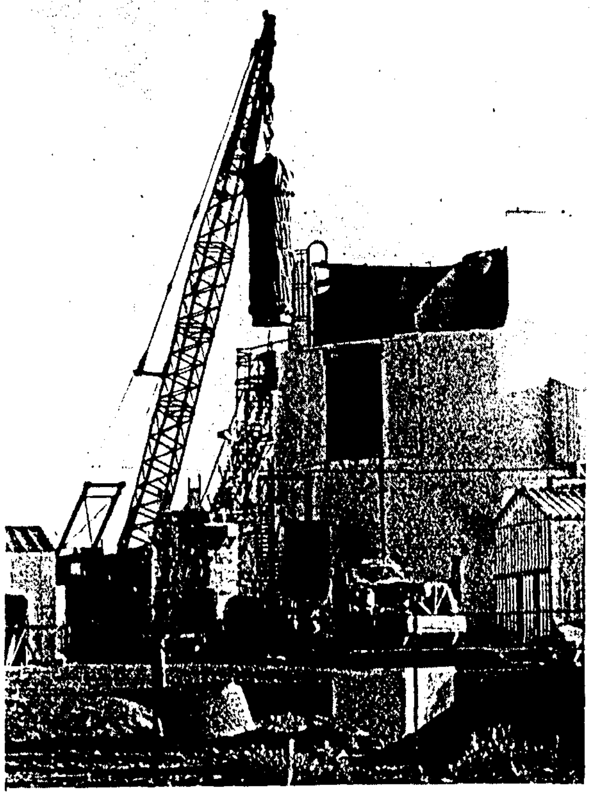


Fig.32

Pressure vessel being removed from SL-1 reactor building

some of these circumstances and conditions should have existed. Factors underlying various design features, conditions, procedures, etc., include such intangibles as operating and design philosophy, engineering judgement, state-of-the-art of reactor development at the time, budgetary and programming considerations, administrative procedures and organization.

B. SL-1 core design

1. Reactivity worth of the central control rod

With the reactor at ambient temperature and pressure and with the four outside rods fully inserted, the reactor could be made critical by the withdrawal of the central rod alone.

The central rod (No. 9) critical position, measured early in the core life, was 19.2 in at 83°F; in February 1960 this position was 16.1 in at 83°F. In September 1960 the position was measured at 14.3 in at a temperature of 106°F. In November 1960 additional cadmium was added to the core which decreased the core reactivity by about 1% as indicated by the change in rod

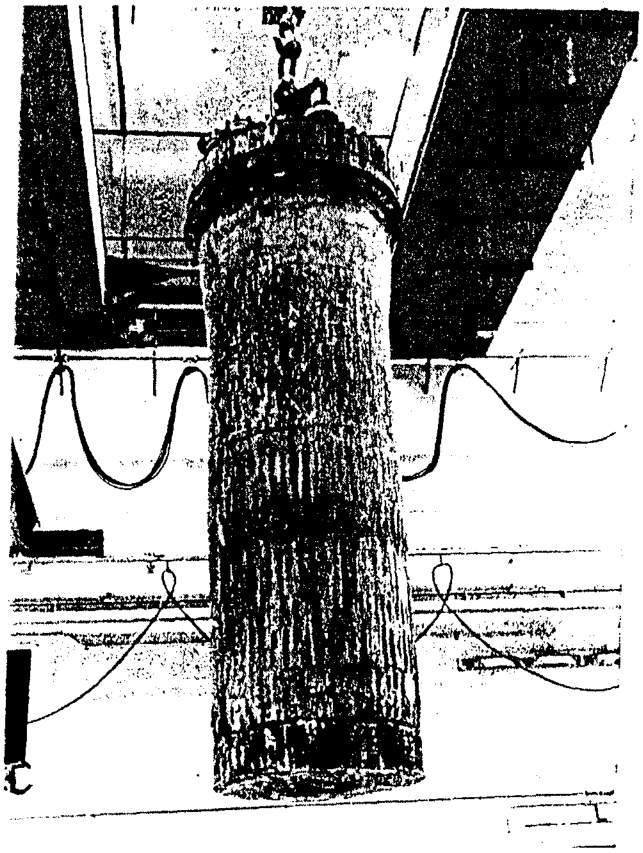


Fig. 33

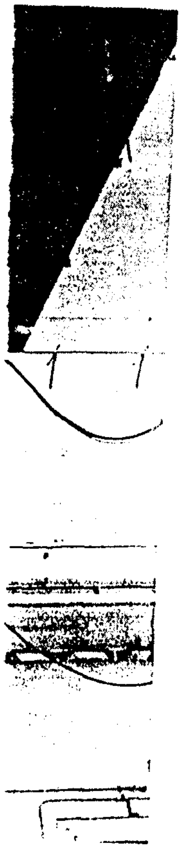
Outside of pressure vessel before bottom skirt was removed

bank position. This presumably would have also raised the critical position of rod 9 a slight amount, but this was not measured.

For remote site applications, it is necessary to keep the size and weight of the reactor to a minimum in order to minimize transportation and installation costs. This requirement made it necessary to optimize for compactness, efficiency, and reliability. The SL-1 reactor was designed to accommodate 59 fuel elements, one source assembly, and 9 control rods. However, during the initial zero power experiments, it was evident that a 40 element, 5 rod core would adequately meet the basic design criteria of 3 MWt operation with a 3-yr core life. It was this deliberate effort to minimize the size of the core which gave the central rod an abnormally large reactivity worth.

2. Boron burnable poison

In order to obtain a 3-yr core life at 3 MWt, burnable poison was required to compensate for the heavy loading of uranium-235. Attempts to



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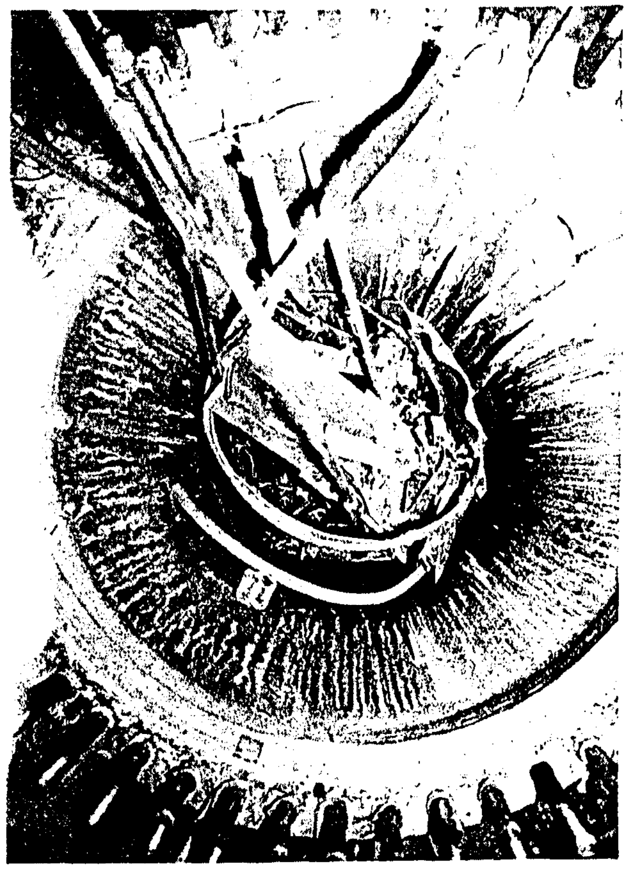

Fig. 34

Inside of pressure vessel just after head was removed - 22 Jan. 1962

include this poison in the aluminium-uranium fuel matrix proved unsuccessful. As was done in Boiling Reactor Experiment No. 3 (BORAX III) where boron strips were used to assist rod control, boron strips were fusion welded to one or both side plates of designated fuel assemblies. The flexibility of this method proved to be very useful since the final boron loading could be readily changed during the zero power experiments which immediately preceded full power operation.

During the fabrication of these strips, the aluminium-boron meat was placed in an aluminium jacket. Pressing and rolling were calculated to result in a 2 mil clad. Strips were then cut from large rolled sheets leaving the meat on the edges exposed and, subsequently, these strips were fusion welded to the fuel assemblies.

In the operation of the SL-1 reactor, there had been considerable concern that swelling of the aluminium fuel elements might occur as a function

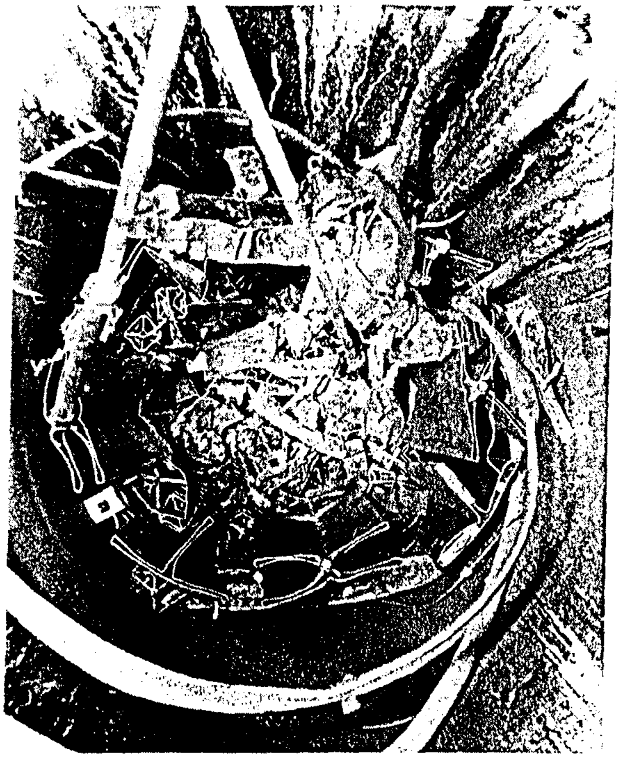

Fig. 35

View of core after the No. 9 shroud was removed

of irradiation damage. A schedule for the removal and inspection of selected fuel elements was established to check for fuel element swelling. During such an inspection in August 1959, the aluminium fuel elements were in good condition, but there were indications that the boron strips were bowing. During a similar inspection in August 1960, the boron strips had bowed between the weld joints and had wedged the elements tightly together. Much force was required to remove one of the centre fuel elements. On one element it was found that boron side strips had bowed up to 170 mils between the welds. This element, photographed under water above the core, is shown in Fig. 36. On another element, both the half length and full length boron strips were missing when removed. Portions of these strips plus a loose boron strip from an adjacent fuel element were subsequently recovered from the core. The appearance of these strips, in comparison with an unirradiated strip, is shown in Fig. 37.

Prior to finding the corroded boron strips, it was noted that the operating rod positions were deviating from those that had been predicted analytically by the window shade technique (Fig. 38). It has been calculated that over the core life the rod bank positions would first move in, then level off, then move out. Actually, the bank positions were moving in but at a faster

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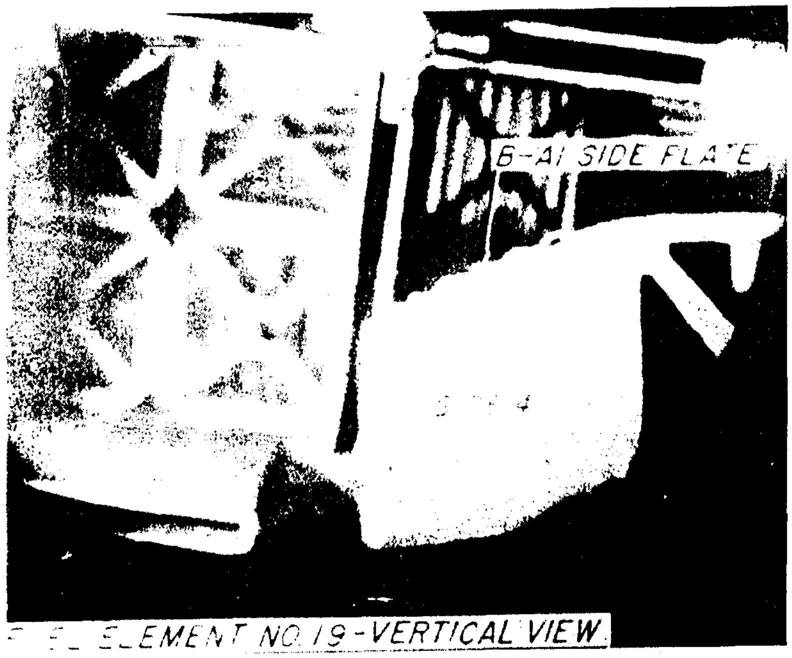


Fig.36

View of boron strip bowing - Aug. 1960

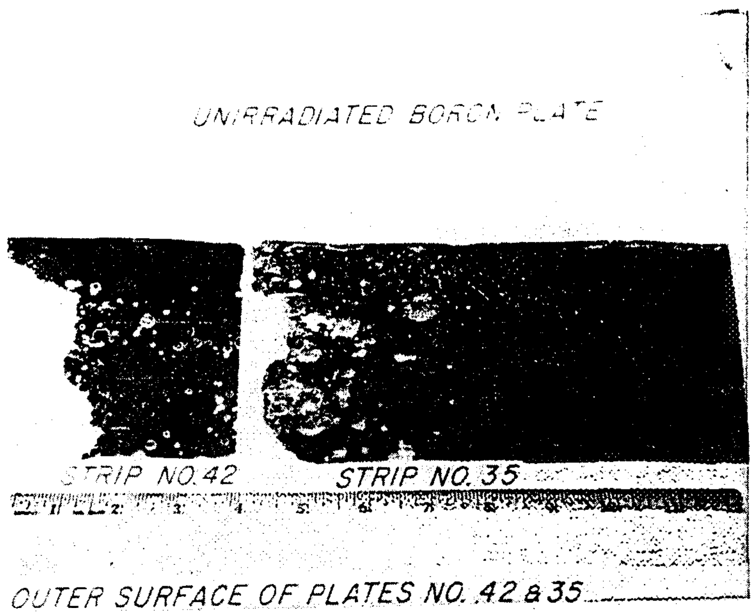


Fig.37

Comparison of remaining pieces of boron-aluminium strips with unirradiated strip - Aug. 1960

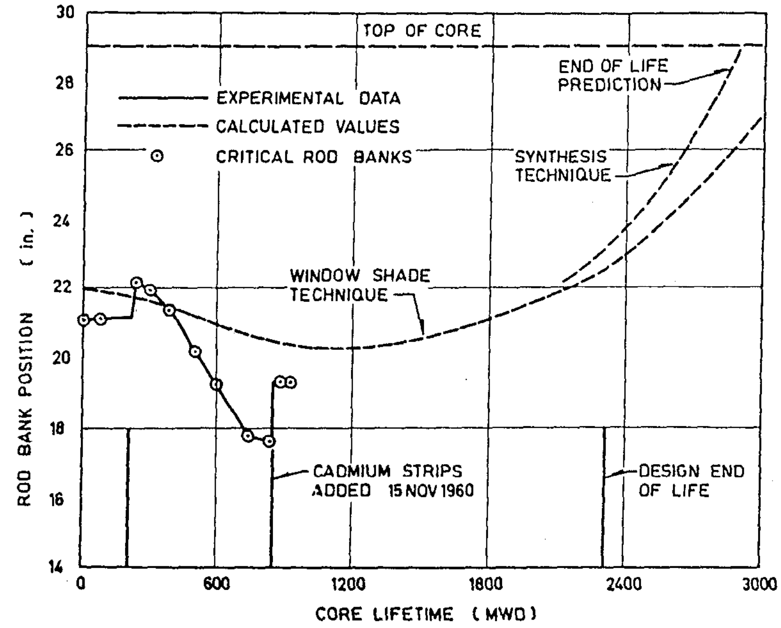


Fig. 38

Critical rod bank position with equilibrium xenon concentration at 2.56 MW

rate than expected, indicating a more rapid gain of reactivity then expected. This could have been caused by the loss of some of the boron.

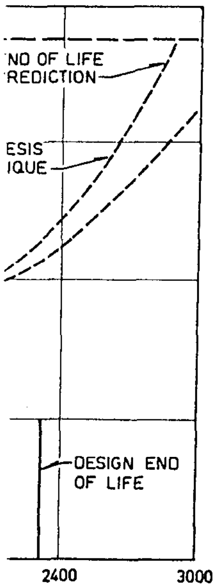
In September and October 1960, an experimental and analytical programme was conducted to investigate the reactivity gain and the corrosion of the burnable poison. As a result, additional shut-down margin was provided by the addition of 60 mil cadmium strips in two of the T slots. It was estimated that this increased the shutdown capability of the reactor by approximately 1% reactivity.

Except for additional boron burn-up, there is no information which indicates that the condition of the boron strips changed during November and December 1960; hence, the above was supposedly the approximate burnable poison status of the core at the time of the accident.

C. SL-1 control rod drive mechanisms

1. Performance of the SL-1 control rod drive mechanisms

The control rods were driven by a rack and pinion mechanism located in a pressure housing (also called a "bell housing") on the head of the reactor vessel as shown in Fig. 39. The control rod blades were guided by shrouds within the core. At a ball joint, the blades were connected to vertical control rod extensions and racks. The racks meshed with a pinion gear. The horizontal pinion shaft penetrated the wall of the thimble through a rotating seal and was driven by a motor through a gear-box and magnetic clutch. By de-energizing the clutch coil, the pinion was released from the



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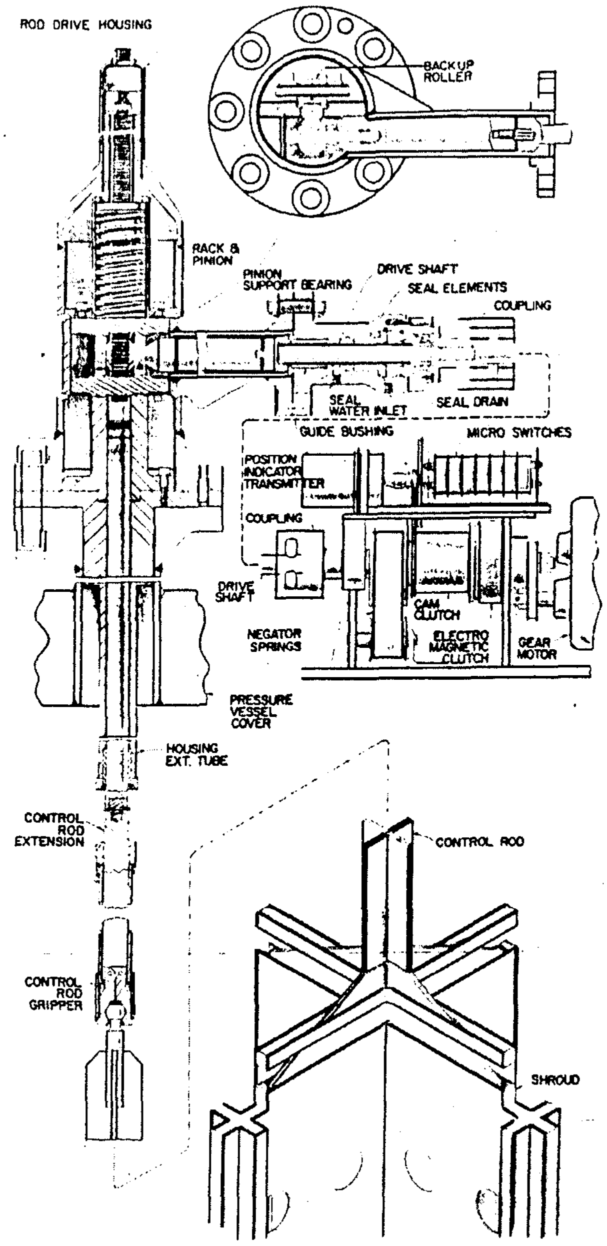

Fig.39

SL-1 control rod drive mechanism

motor, and the rod could then fall by gravity, with the rack and pinion gear "free-wheeling". Any friction in the seal on the horizontal pinion shaft would tend to impede the fall of the rod. An auxiliary clutch permitted the motor to drive the released rod downward and, if necessary, prevented upward rod motion after release.

A detailed investigation of the SL-1 operating logs disclosed that the SL-1 control rod drive mechanisms performed a total of 4300 movements. In 98% of these cases, the mechanism operated satisfactorily. In 84 instances, or 2% of these cases one or another of the 5 mechanisms operated in a less than satisfactory manner. Forty-six instances were noted where a rod did not fall freely in a scram and required the mechanical drive to assist or drive the rod in. During November and December 1960, 33 instances of sluggish or sticking performance were experienced.

The 84 instances mentioned above include instances (1) when a control rod did not meet specified minimum drop time requirements during "free" fall, (2) when a power assist from the drive assembly was necessary to enable a control rod to reach its zero position, or (3) when it was not possible to withdraw a rod prior to startup.

Cases of unsatisfactory performance occurred in a sporadic and erratic manner. Because of the erratic operation, it is difficult to indicate any mechanism which by itself could have caused sticking to occur. In a few of the sticking instances noted, it was known that crud accumulation around the rotating seals and pinion bearings was the cause. Other instances can be attributed to other mechanism problems; however, the cause of the majority of the instances was not identified.

The SL-1 Board of Investigation considered several other possible causes of control rod sticking, but found no evidence for any one cause. Among these was the possibility that the control rod shrouds may have closed in on the blades, because of bowing of the boron strips resulting in crowding of the fuel elements against the shrouds, adding to the friction of the system; crud accumulation within the shrouds may have caused the erratic performance of the control rods.

Very few incidents of rod sticking were formally reported. There had been some trouble with the control rod mechanisms from the beginning, and the crew was accustomed to slight rod irregularities. The increasing frequency of difficulties just prior to the accident were not reported to the AEC.

1. Other design considerations

(a) 17-4 PH steel

The use of 17-4 PH steel in the fabrication of some of the SL-1 control rod drive components was consistent with the state-of-the-art at the time. The control rod racks recovered from the SL-1 reactor building subsequent to the accident show many surface cracks. In other reactors, it has recently been found that 17-4 PH steel can only be used in reactor components if it is fabricated and processed through carefully controlled heat treatments and manufacturing procedures. Otherwise, progressive stress cracking leading to eventual failure might occur. This was not known when the SL-1 was constructed. Some of the components in SL-1 showed stress cracking (Figs. 40 and 41), but so far there is no evidence any cracks had progressed to the point of failure.

(b) Manual movement of rods during disassembly and assembly

During the disassembly and assembly of the SL-1 control rod drive mechanisms, it was necessary to move manually the control rod blades

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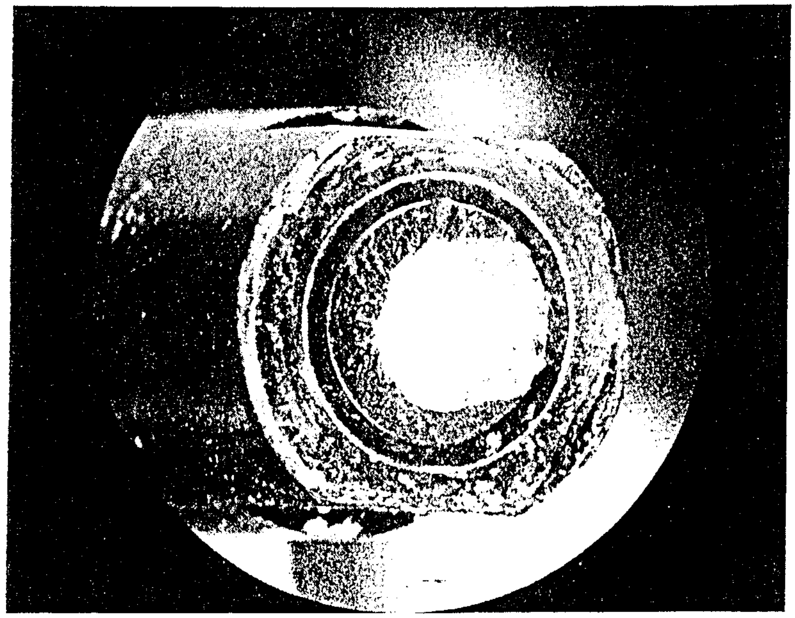

Fig. 41

Recovered SL-1 control rod extension showing fracture. Dark outer area of fracture caused by pre-accident fatigue. Bright area caused by impact tensile

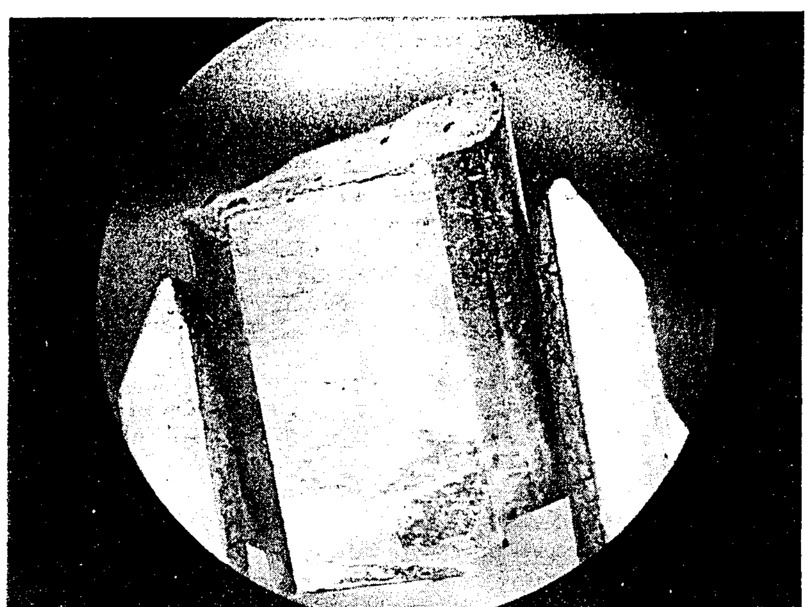

Fig. 40

Recovered gear rack showing transverse stress cracks on flat of rack within the core. As noted earlier, each control rod in the core is connected by a long, upward projecting control rod extension to a rack and pinion gear drive located on the top of the reactor vessel head. The rack and pinion gear is inside of a tall, bolted-on bell housing. The horizontal drive shaft from the pinion gear to its drive motor outside the bell housing extends through a rotating reactor pressure seal in the wall of the housing.

When an SL-1 drive mechanism was disassembled, all the drive components were removed from the reactor head and, hence, access to the core was possible. This situation is shown in a pre-accident photograph, Fig. 42, with only the control rod rack protruding through the reactor nozzle.

In the reassembly of these mechanisms, the shield plug was lowered into the reactor head nozzle over the rack. The pinion support and spring housings were then lowered over the rack and bolted to the shield plug. A lifting tool was attached to the threaded end of the top of the rack, down inside the spring housing a few inches. At this point the rack and, hence, the control rod were lifted so that a "C" clamp could be attached to hold the rod in a raised position. Very explicit instructions had been given to all operators that this manual raising of a rod should not exceed 4 in. However, the operator was expected to exercise judgement estimating this height. There was no position stop; it was possible for the operator to raise the rod higher-- even to complete withdrawal.

With the "C" clamp on the rack, the lifting tool was removed and a washer and nut were placed on the rack. This nut and washer acted against the spring to hold the rod in the zero operating position and to absorb the force of scrams. The lifting tool was again attached and the rod lifted to free the "C" clamp. The rod was then lowered to the spring. This point in the reassembly is shown in Fig. 43. Fig. 44 shows the cadmium overlap in the active core at various positions during the reassembly procedure.

Based on the last measurement of critical position of the centre rod, there should have been at least a 12 in margin between criticality and the position to which the centre rod is normally raised to during this operation.

D. SL-1 operating and maintenance procedures

Prior to the accident, the SL-1 control rod drive disassembly and assembly procedure was considered routine by all concerned and had been done many times. Hence, a reactor engineer was not scheduled to be present while this procedure was performed on the night of the accident. The written procedure for the disassembly and assembly of the SL-1 control rod drive mechanisms did not have a precautionary note to indicate the danger involved in withdrawing the central rod, but this procedure and the administrative precautions relating thereto were well covered in the training of all operators. The Board of Investigation found that all reactor operators were well aware of the danger associated with this procedure.

The established procedures did not require a crew member to be in the control room during maintenance on the reactor. The SL-1 control and nuclear instrumentation was adequate. However, at the time of the accident, the recorders associated with the nuclear instrumentation (with few exceptions) were turned off. The operating procedures did not require that all recorders be turned on. The constant air monitoring system was on. However, this system would not have responded to the difficulties within the reactor.

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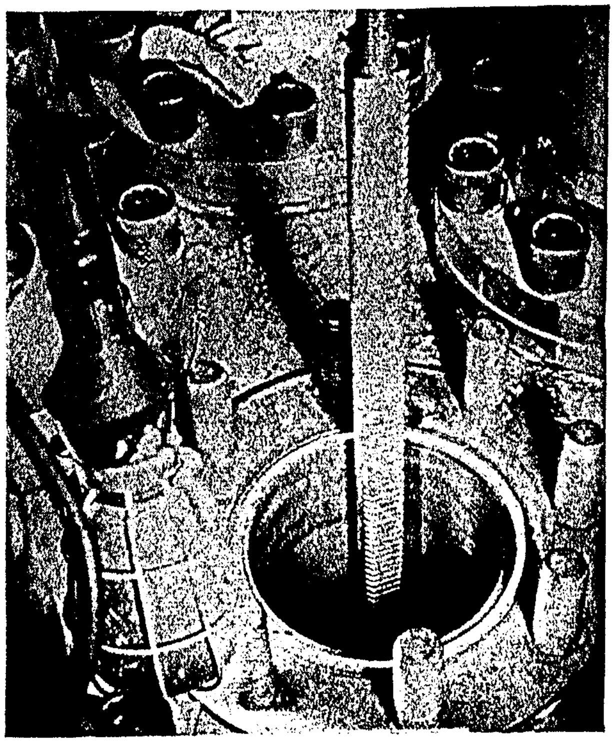

Fig. 42

SL-1 control rod drive rack (pre-accident photo)

E. Reactor safety reviews and inspections

During the operation of the SL-1, many safety reviews and inspections were performed by groups directly associated with the operations and programming. However, most formal inspections were concerned with health physics, radiation protection, and industrial safety problems. No nuclear or reactor engineers were included on the formal inspection teams. Hence, the reactor safety aspect was not adequately covered.

There were only two truly independent over-all safety reviews of the SL-1 facility including the reactor. The first review was made by the AEC's Division of Licensing and Regulation and the Advisory Committee on Reactor Safeguards. This review was made prior to the initial operation of the SL-1. The second review was accomplished by an independent group from the operating contractor's organization at the time this firm assumed operational responsibility for the SL-1 in February 1959.

V. PROBABLE INITIATION AND COURSE OF THE SL-1 ACCIDENT

The investigation into the cause of the accident is still underway by the SL-1 Board of Investigation. The final report by the Board will probably be completed this summer. The Board has released several interim reports; the latest on 3 April 1962 follows:

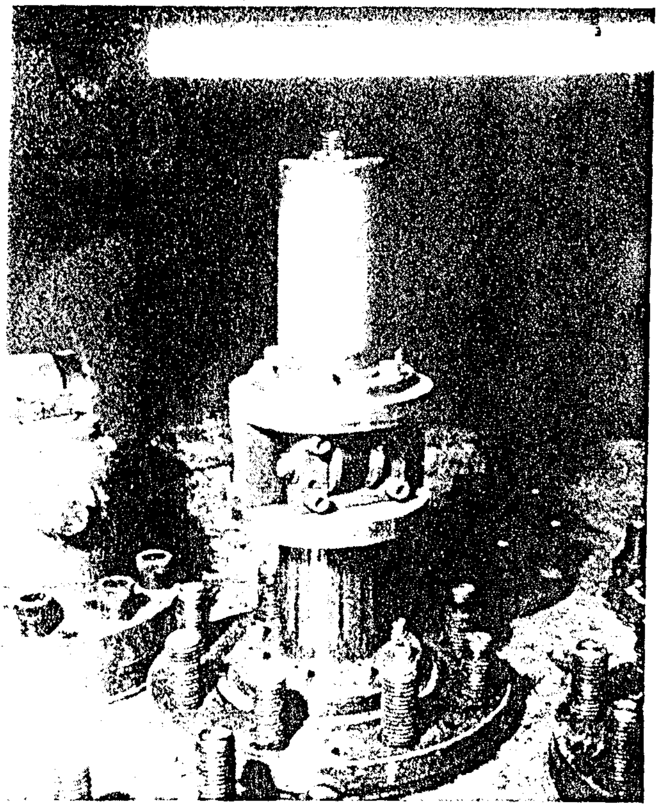

Fig. 43

SL-1 control rod mechanism (pre-accident photo)

"A meeting of the Board of Investigation on the SL-1 reactor incident of 3 January 1961, was held on 7 March 1962. The purpose of this meeting was to review evidence which has been brought to light since the Board's last report of May 1961. During this period, the reactor has been moved to a large "hot cell" and partially disassembled to facilitate careful detailed viewing and study of each component and bit of evidence which might bear upon the cause of the incident.

The Board finds it is not in a position to submit a final report but does wish to reaffirm the conclusions reached in its report of 10 May 1961. A great deal of additional evidence has been developed since that report, touching particularly on conclusion (H) * . While the Board has not made a complete review or study of all the new evidence, it finds none which appears to change its conclusions materially, but rather finds further support for those conclusions.

The following observations are based, in large part, on information obtained by the General Electric Company during the recovery and disas-

* Conclusion H in the May report states: "At this time it is not possible to identify completely or with certainty the causes of the incident. The most likely immediate cause of the explosion appears to have been a nuclear excursion resulting from unusually rapid and extensive motion of the central control rod. As yet there is no evidence to support any of several other conceivable initiating mechanisms".

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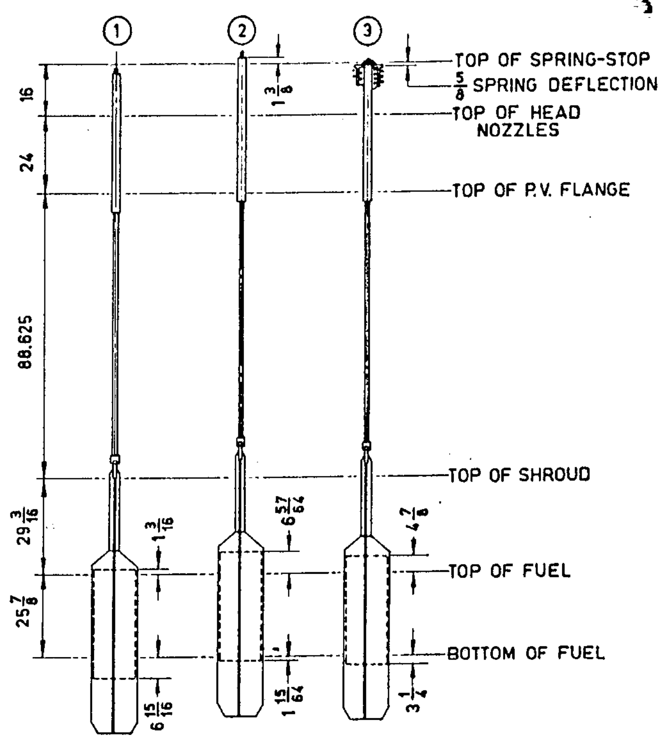

Fig. 44

SL-1 control rod cadmium overlap in active core for various positions

sembly of the SL-1 reactor vessel and core over the past several months under a Commission contract administered by the Idaho Operations Office:

(1) When the explosion occurred, the reactor core was destroyed and a pressure wave or water hammer followed which apparently trapped the central control rod (No. 9) within its shroud at a 20 in, plus or minus ½ in, withdrawn position.
(2) The radial dislocation of the core components indicates that the explosion emanated from the centre axis of the core or that part of the core controlled by the central rod.
(3) Severe meltdown of the centre and lower portions of the central fuel elements was experienced.
(4) Preliminary flux wire measurements from wires which were in the core at the time of the incident indicate that the magnitude of the energy released from the resulting nuclear excursion was sufficient to cause the observed damage and effects.
(5) Direct measurements of the critical position of the central rod with the core in a cold condition were few; however, from an analysis of the history of the SL-1 core, it appears that the critical position was between 14 in to 16 in. Hence, with the known reactivity worth of the rod, its withdrawal to 20 in appears sufficient to cause the effects observed.
(6) Evidence accumulated so far from within the reactor vessel points

to no self-propagating metal-water reaction or any other type of chemical explosion.

(7) All the observed damage to the reactor building, vessel and core can be reasonably accounted for as a result of the withdrawal of the No. 9 (central) control rod.

The reason for the withdrawn position of the central control rod is unknown. It is a principal and final objective of the Board to find this reason, if possible, or a reasonable hypothesis of the withdrawing mechanism, and to report other evidence of value to reactor safety through a detailed evaluation and analysis of the reactor core and reactor components. A final report will be written upon conclusion of this work.

The conclusions in this interim report are based on the following most probable sequence of events, believed to have occurred during the course of the accident, and which, at the present time, reasonably explains all of the observed damage.

It is believed that the SL-1 accident was caused by the rapid withdrawal of the central control rod (No. 9) above its critical position, 14 to 16 in, to a position of approximately 20 in, thus taking the core above prompt critical.

This nuclear condition rapidly increased the fuel plate temperature to a point near or above melting. The simultaneous generation of steam throughout the centre of the core produced a relatively large steam void and high pressures in the core in the order of 500 lb/in^2. Consequently, the core experienced considerable damage by the expansion of this steam and by the high pressure. At this time, the central rod was probably seized by the shroud surrounding it at about a 20 in withdrawn position. The 500 psi steam pressure apparently forced a slug of water upward from the general zone of the core. This water slug was accelerated by the steam and was suddenly stopped by the reactor vessel head, causing a high pressure, water hammer phenomenon with pressures probably as high as 10 000 psi. The forces generated by the decelerating water slug collapsed all the shield plug housing extension tubes (Figs. 27 and 39) and deformed the reactor vessel wall (Fig. 33). Additionally, the momentum of the water slug was transferred to the reactor vessel, imparting a vertical motion to the vessel itself and the shield plugs, which were not bolted to the vessel head. The vessel was projected upward sufficiently to shear the steam nozzle and water lines and to expel onto the operating room floor whole blocks of insulation which originally surrounded the vessel. Subsequently the vessel fell back approximately to its original position.

It has been calculated that the energy released was about 300 MW sec.

For further details concerning the WTR and SL-1, the reader is referred to the bibliography.

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DISCUSSION I. 4.

K. KONOPLEV: Were any cases of leaks observed in the fuel elements of the WTR reactor, either before or after the accident in April 1960? If so, how often?

A.N. TARDIFF: No, fission product release to the coolant was not observed before the accident. It is true that the fuel elements had not been adequately inspected prior to the accident. The investigation that I referred to took place after the accident and included a test to determine whether there had been any flaws or defects in the bonding.

G. RADA: An issue of Nucleonics reported the finding of a locking piece for the central control rod assembly. This suggests that the rod remained caught in its lower position, thus eliminating the possibility of its accidental withdrawal. Could you comment on this finding in the light of the fact that the fundamental assumption in the investigation seems to be that the central control rod was accidentally withdrawn?

A.N. TARDIFF: With reference to the report in Nucleonics, the USAEC in September 1961 released information to the public to the effect that the No. 9, or central, control rod extension had been found trapped within its shield plug extension guide tube, indicating that the No. 9 control rod was in a 3 in withdrawn position at the moment the "water hammer" effect took place. This tended to cast doubt on the theory that the nuclear excursion was a result of the control rod being withdrawn. However, subsequent detailed examination of the same components indicated that we had released the information prematurely. Apparently there had been differential movement between the shield plug and the central rod extension. We found identical or matching impact points on the inside surface of the shield plug guide-tube and on the surface of the central rod extension (24 in apart). This indicated that the No. 9 control rod was 27 in (24 in + 3 in) withdrawn at the time the water hammer effect took place. Further information revealed that the control rod was withdrawn 20 in at the time the core was destroyed.

G. RADA: I understand that the shift supervisor on duty was absent from the reactor site at the time of the incident. Do you think that this was a decisive factor in the chain of events that gave rise to the incident?

A.N. TARDIFF: I must point out, first of all, that the reactor, or shift, supervisor was in fact present: he was the man who was pinned to the ceiling. The crew consisted of a shift superviser, a reactor operator and a trainee. The operations supervisor and plant superintendant were not on duty. (The operation or maintenance which was being carried out was considered routine and hence the normal operating crew was given the responsibility of re-assembling the control rod drives.) The shift supervisor on duty was duly qualified by the operating contractor and the military cadre responsible for the training of the crews.

G. RADA: Could you comment on the distribution - before the SL-1 accident - of responsibilities as between Combustion Engineering, the Army and personnel of the National Reactor Testing Station?

A. N. TARDIFF: During the operation of the SL-1, the operating crews and the plant superintendant were military. The crews were responsible to the plant superintendant and the latter was responsible to the operations supervisor, who was an employee of Combustion Engineering, Inc., the operating contractor. The contractor was responsible to the Idaho office of the USAEC. The on-site military personnel did not report to the USAEC but were working directly for the civilian contractor. Normally, such an arrangement can be maintained as long as the personalities involved are compatible. In the case of the SL-1, the relations between the military and the civilian contractor appear to be good.

E. STAUBER: Was it possible to estimate, or conclude, how much activity, especially fission product activity, was released from the reactor core, and from the containment during the SL-1 accident? What were the values in both cases?

A. N. TARDIFF: I should indicate first of all that, except for the pressure vessel itself, the SL-1 was not contained. Work on determining the amount of fission products released from the reactor core to the reactor building and to the site has not been completed. Original estimates indicated that about 1% of our total inventory of fission products in the core, amounting to approximately 10^6 c, was released. When the pertinent analyses have been completed, reports will become available. (With the exception of the Board of Investigation reports and the Congressional reports, the official SL-1 accident reports are the IDO-193 series, which are available at the United States Office of Technical Information, Washington 25, D. C.)

J. A. BOURGEOIS: In your oral presentation you mentioned that a radioactive cloud was formed at the time of the SL-1 accident and that passage of the cloud had been recorded in a facility, situated to the north. How high was the radiation recorded during the passage of the cloud?

A. N. TARDIFF: The facility to which I referred is the Gas Cooled Reactor Experiment (GCRE) located 0.4 miles northwest of the SL-1 site. Shortly after 9 p.m. (the accident occurred at 9.01 p.m.,) the GCRE gate house personnel monitor sounded an alarm and all meter needles registered off-scale. The instrument could not be reset. Hence, a direct measurement of the cloud was not made. It is still a matter of uncertainty whether the cloud caused the alarm. Further information on the cloud and its disposition is given in IDO-19302, "Nuclear Incident at the SL-1 Reactor".

J. H. COLLINS: Although it is felt that the WTR incident was due to poor bonding between the cladding and the fuel, would you consider reducing the overall cooling again if you were carrying out another experimental programme of similar type?

A. N. TARDIFF: According to the WTR staff, there was nothing wrong with the experiment. The problem was one of co-ordination between the experimenters in the reactor room and the crew operating the reactor. In any future experiment of that kind, it would be necessary for the two groups to work in close contact with each other.