UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
THE REGENTS OF THE UNIVERSITY OF CALIFORNIA	)	Docket No. 50-142
	)	
(UCLA Research Reactor)	)	(Proposed Renewal of Facility License)

TESTIMONY OF MILLARD L. WOHL

Q.1. Please state your name and place of employment.

A.1. My name is Millard Wohl. I am employed by the U.S. Nuclear Regulatory Commission in the Accident Evaluation Branch (AEB), Division of Systems Integration. A statement of my professional qualifications is attached to my testimony.

Q.2. What is the purpose of your testimony?

A.2. The purpose of my testimony is to point out why I consider the three scientific analyses of the Argonaut-UTR, referenced in the SER for UCLA, to be reliable and more than adequate to demonstrate that the UCLA Argonaut-UTR is inherently safe. I was the technical manager of the contracts with Battelle (PNL) and Los Alamos National Laboratory (LANL) to perform the analyses published as "Analysis of Credible Accidents for Argonaut Reactors" (NUREG/CR-2079) and "Fuel Temperatures in an Argonaut Reactor Core Following a Hypothetical Design Basis Accident" (NUREG/CR-2198). I also worked with Ms. Mitchell, a colleague in the AEB who managed the contract with Brookhaven National Laboratory (BNL) for the study entitled "Transient Analysis of the UCLA Argonaut" (BNL Memorandum, P. Neogy 1981).

Q.3. Why were the Battelle PNL, Los Alamos National Laboratory, and Brookhaven National Laboratory studies performed?

A.3. The Staff believed that more thorough safety evaluations should be performed for ongoing NRC non-power reactor license renewal actions than those which had been done in the fifties and sixties. This was particularly true for the high seismicity areas in which some reactors are located such as UCLA. For this reason, the Los Alamos study was initiated to ascertain the consequences to the fuel in a crushed core configuration. The study provided sound evidence to us that sufficient cooling is available through natural convection and conduction that would inherently preclude melting. It is my opinion that without melting of the fuel elements, radionuclide releases to the environment would be extremely small in the event of any postulated core-crushing accident for low power, intermittently operated reactors such as the Argonaut.

The Battelle (PNL) study was performed to consider, extremely conservatively, a full spectrum of accident scenarios to which Argonaut reactors could be subjected, assuming a "worst case" event. This study demonstrated for us that the only possible accident of concern in the spectrum was the fuel handling accident, and that the offsite radiological consequences resulting from this accident were within present regulatory limits. This information is especially useful to the licensing staff when one considers that in over 500 reactor-years of commercial nuclear power reactor operation, where fuel handling is an ever-present part of operations, only about six fuel handling accidents have occurred, none resulting in any measurable release of radioactivity, and that at UCLA, fuel handling will be necessary only one or two times during the projected licensing period.

Work was initiated with Brookhaven National Laboratory (BNL) to further investigate the response of Argonaut reactors to rapid insertions of reactivity. The RETRAN-01 computer code, appropriately benchmarked against previous SPERT work, was used to study a realistic hypothetical insertion of $3.00 of reactivity in a ramp of ¼ second duration. The ¼ second ramp is the approximate time for a free fall of a sample of negative worth, such as cadmium neutron-absorptive sleeving material, through the core region to a position below the core. This was considered to be the maximum rapid reactivity insertion plausible. The BNL analysis demonstrated that the core thermal response to this insertion is well within safe limits, since peak clad temperatures remained about 250°C below the clad melting point.

In summary, the three laboratory analyses demonstrated for the Staff the safety of the Argonaut in case of any accident.

Q.4. For what reasons do you believe these three reports to be scientifically reliable?

A.4. In general, the reason for confidence in these analyses is the significant conservatisms contained in the assumptions and calculations so that the conclusions contain a wide margin of safety or accommodation of error. In addition, each study has been verified.

Most importantly, the individuals performing the studies have, in the staff's opinion, excellent qualifications and a great deal of experience which renders them highly suitable to perform these studies. Mr. Sean Hawley at Battelle (PNL) has eight years of experience working with research reactors, including being a Senior Operator, health physicist, reactor supervisor and training supervisor at Reed College and the Washington State University. Mr. Ron Kathren at Battelle also has had extensive research reactor experience and a great deal of senior health physics background. Dr. Partha Neogy of BNL, has specialized in Reactor Physics and Reactor Safety Analyses at BNL. Previously, he specialized in fuel management, and power distribution and control analyses in the commercial nuclear power industry. The staff considers him eminently qualified to have performed the transient study. Mr. G. Edward Cort of LANL is an Associate Group Leader in a fluid flow and heat transfer section at LANL. His related specialties are heat transfer, fluid flow, and HTGR fission product release.

In my judgment he is eminently qualified to have performed the heat transfer analyses in the subject study.

In addition, the methodologies used in the Battelle, BNL, and LANL studies are scientifically sound. The Argonaut and SPERT cores are similar with respect to fuel and moderator. Additionally, the transient behavior of all SPERT cores on the basis of reciprocal period is remarkably similar and the differences that do exist are predicted by a model focusing on the average void coefficient and the neutron lifetime.

In the BNL work, use of RETRAN, a versatile systems transient code, is well justified. It has been used to analyze the Semiscale and LOFT experimental facilities and the General Electric Two-Loop Test Apparatus, as well as a number of separate effect experiments on pressure drop, heat transfer, critical flow, and multi-dimensional flow. The point kinetics neutronics model used in RETRAN has even better validity for a small core such as that for SPERT or an Argonaut than for a large LWR.

With respect to the LANL heat transfer work, the computer code AYER used to perform the work has been in use at LANL for ten years. It has been used successfully in a wide variety of applications, particularly where graphite is present in substantial quantity in the system to be analyzed, such as at

UCLA. Examples of applications of AYER are the analysis of heating of graphite targets in linear accelerators, heat transfer analyses in blankets and first (plasma containing) walls in advanced fusion reactors, and the analysis of core heatup in Design Basis Depressurization accidents (DBDA) in high-temperature gas-cooled reactors (HTGR).

In sum, the staff's judgment is that the methods and computer codes used by Battelle, BNL, and LANL in performing transient and system heatup analyses for the Argonaut-UTR are well founded and verified. The studies were conducted very carefully by highly qualified, capable, experienced individuals. The methods employed have been tested and verified. The numerous conservative assumptions made in the studies lend additional credence to the overall conclusion that 100KW Argonaut-UTR transient/heatup phenomena will not lead to fuel melting or disassembly resulting in fission product release to the environment, and that the Argonaut is inherently safe.

Q.5. Please explain the reliability of the analysis of an excess reactivity excursion by Battelle (PNL).

A.5. The Battelle study employs the following conservatisms:

(a) Instantaneous insertion of excess reactivity. This is not a real possibility but a mathematical construct. Any insertion of reactivity would occur over time.

(b) Insertion of 2.6% ($4.00) excess reactivity. This is far more excess reactivity than the UCLA reactor has available or will be licensed for. Presently the available excess reactivity is less than $3.00 and the proposed technical specifications for the license renewal set a limit of $3.00. No samples used at UCLA have enough worth to create excess reactivity more than $3.00.

(c) the most conservative values for delayed neutrons, mean prompt neutron lifetime, and available excess reactivity were used in calculating the asymptotic period.

(d) two methods of calculating maximum energy release for an Argonaut, based on empirical SPERT-1D data were used, both of which assumed symmetrical power increase and adiabatic temperature increases, which are additional conservatisms. The two methods produced similar results (9MWs and 12MWs).

Q.6. Discuss the differences in T, the asymptotic period following a reactivity insertion; l, the mean prompt neutron lifetime; and the effective delayed neutron fraction used in the Battelle report (NUREG/CR-2079) and the values used in the BNL transient analysis of the UCLA Argonaut.

A.6. The value of T = 7.2 msec for the asymptotic period used in the Battelle report (NUREG/CR-2079) is intended to be generic and is extremely conservative. It is computed from a form of the inhour equation using a very conservative mean prompt neutron lifetime ($l = 1.4 \times 10^{-4}$ sec) and the most conservative effective delayed neutron fraction (0.0065), which really applies for pure or more highly enriched ^{235}U than the 93% enrichment in the UCLA fuel. The total energy release (12MWs) and resulting adiabatic temperature rise of the fuel of 240°C, are thus, very conservative numbers due to the conservative values used in the calculations, and yet show a margin of safety of about 200°C, using various calculations. Since the fuel melting point is 640°C and the latent heat of fusion would also have to be supplied at a 640°C fuel temperature, the Battelle analysis demonstrates that fuel melting is not credible in an Argonaut.

The BNL Argonaut transient study utilized a 15.8 msec period, since it corresponded closely with the transient that would be produced by the insertion of \$3.00 reactivity in ½ second in the Argonaut core. The prompt neutron lifetime used was $l = 1.83 \times 10^{-4}$ sec and the effective delayed neutron fraction was 0.00714. In the transient analysis, performed with the RETRAN-01 code, onlv moderator density and temperature effects were included as reactivity feedback mechanisms. Fuel Doppler feedback and core geometry changes due to fuel plate heating were neglected, leading to conservatism in the calculation of the peak clad temperature. The presence of the central graphite island in the UCLA Argonaut, an added safety feature, is specifically included in the determination of the prompt neutron lifetime used by BNL in the transient analysis. The analysis showed peak clad temperatures about 250°C below the clad melting point. It should be pointed out that the ¼ second ramp reactivity insertion time utilized is considered rapid in the credible spectrum of such time, and that a step insertion, being merely a mathematical construct, does not account for the time in which physical processes occur in nature.

Q.7. Discuss the conservatisms in the Battelle analysis of a fuel handling accident.

A.7. In the Battelle fuel handling accident analysis, the assumption of 7% of the total core fission product inventory, contained in one of 24 core fuel elements, is based on a 1.5 peak-to-average power density ratio. This is equivalent to asserting that the fuel element involved in a fuel handling accident is from the hottest portion of the core (with greatest neutron flux). Additionally, although a gap is assumed, Argonaut fuel plates have no gaps in which volatile fission products can accumulate for release if the fuel were to be damaged. The Battelle assumption of release of <u>all</u> gaseous activity produced within a distance from the fuel plate surface equivalent to the range of fission fragment recoil particles is likewise very conservative (2.7% of volatiles). Thus, due to the absence of any porosity in the fuel meat, or any gaps for recoil fission products to reside in and become gaseous, it is very unlikely that inert volatile fission products will escape the fuel. Therefore, in actuality only some small fraction of this amount might be expected to be available for release.

The Battelle assumptions that the reactor was operated continuously for a full year (36.5 Megawatt-days), and that releases occurred from the entire surface area of a fuel element containing 11 plates (equivalent to shredded or finely fractured plates) are extreme conservatisms. Further, no credit was taken in the analysis for internal chemical reactions, mixing, filtration, or plateout of released volatile radioactivity, nor for any period of fission product decay during cooldown. However, substantial iodine plateout would be expected in an accidental release within the reactor area. Also with respect to meteorological considerations, Battelle's use of an atmospheric diffusion and transport relative concentration factor (X/Q) of 0.01 sec/m^3 represents an upper bound to a maximum value range for this parameter at the outside of a building based on my discussion with I. van der Hoven, NOAA, in reference to the study "Near Building Diffusion Determined from Atmospheric Tracer Experiments," Sagendorf, J. F., et al, in "Fourth Symposium on Turbulence, Diffusion and Air Pollution," Jan. 15-18, 1979, (p. 597 of preprints). The staff's slightly lower upper bound

X/Q estimate of 0.007 sec/m^3 in Section 14 of the SER for UCLA is still highly conservative, based upon the same Sagendorf data.

In sum, because of conservatisms in operating cycle, release fraction, fission product removal, and meteorological parameter assumptions, the reported dose of 43.3 Rem to the thyroid is certainly high by a sizable factor. For comparison, we noted the 300 Rem guideline limit of 10 CFR Part 100, so that the result of the worst case assumptions do not give cause for concern.

Q.8. Discuss the conservatisms present in the heat transfer analysis performed by Los Alamos National Laboratory (LANL) for the crushed core configuration.

A.8. It is assumed that all the core water is absent after full power (100kW) operation for a period of time sufficiently long that the fuel fission product inventory is maximized, ensuring maximum heat generation rates and subsequent fuel temperatures. With water loss, as fuel temperatures increase, heat transfer from the fuel to the ambient material will occur by conduction, natural convection, and radiation. However, in calculating the transient fuel heat-up, conduction was allowed to occur only in the two least favorable (most conservative) directions vertically, such that the conduction distance is long and perpendicular to the fuel plates where successive layers of low-conductivity air between the fuel plates inhibit heat transfer. No heat loss was allowed from the edges of the fuel plates, which would constitute a substantial heat transfer mode.

Further, in a subsequent analysis for 500 KW Argonaut reactors, where analysis of heat flow was performed in both horizontal directions, the temperature rise was only 193°C. This study verified the conservatism in the 100 KW Argonaut study.

Q.9. Compare the SL-1 accident with a similar worst case excursion at UCLA.

A.9. Historically, one of the worst reactor accidents in the United States, involving the greatest degree of core damage with the greatest radiological release, was in a non-commercial power reactor, the 3 MW "SL-1" on January 3, 1961. Triggered by rapid insertion of reactivity due to radically incorrect maintenance action, the accident resulted in destruction of the core, lifting of the pressure vessel, expulsion of control rods, and dispersal of radioactivity throughout the reactor building. The released radioactivity was, however, confined mostly to the interior of the building, even though the building was of fairly open, non-airtight sheet metal construction which was not significantly damaged by the enormous and destructive energy pulse. The measured dose rate immediately outside the building shortly following the accident was about 25 R/hr, due mostly to shine from within the building.

A similar large sudden reactivity insertion is not possible at the UCLA Argonaut reactor due to differences in control rod design and worth. However, if a similar degree of core damage were somehow produced at this facility, radioactive releases from the core would be at least two orders of magnitude less than the comparable SL-1 releases due to lower power level and intermittent operating history. The Argonaut power level would produce a factor of 30 times less fissions per second and a Curie inventory smaller than SL-1 by a factor of 30. An even smaller Curie inventory would result due to the intermittent operation. Further, confinement of released radioactivity within the much stronger concrete building at UCLA and shielding of gamma ray shine from within the building would result in a whole body dose adjacent to the building of 0.25 R/hr at a maximum, well within the limits of 10 CFR Part 20. Additionally, people in the area would be evacuated, thus reducing exposures even more. Also, in the event of a destructive explosion, the reactor building at UCLA could better withstand a destructive force than could the SL-1 sheet-metal building.

Therefore, even though no means are available to produce such an event at an Argonaut, an accident similar to the SL-1 accident at the UCLA Argonaut reactor, would not result in radiological consequences inimical to the public health and safety.

Q.10. Compare the Spert ID destructive test with a similar event at UCLA.

A.10. The SPERT I destructive test program resulted in a final test (D-core) which partially melted about 35 percent of the core, with large pressures suddenly produced due to a steam explosion when the reactor was completely self-shutdown after a power excursion. The excursion released 31 MW-sec of energy. The pressure pulse following the excursion demolished the core, damaged most of the associated hardware and part of the control system, and bulged the reactor vessel. Virtually no damage occurred to the open sheet metal building housing the reactor.

Measured (extrapolated) airborne fission product concentration immediately outside the SPERT I building was about 10 microcuries per cubic meter. This is approximately equivalent to a (maximum) 0-2 hr whole body cloud immersion dose of 36 mrem, slightly larger than that for a chest x-ray. This dose is almost entirely due to short-lived airborne fission products produced during the transient.

Since no mechanical damage occurred to the (weak) sheet metal SPERT-I building due to the destructive D-core test, even if this type of accident were postulated at the UCLA Argonaut site, no damage to the reactor room walls or ceiling would be expected. However, this type of accident could not occur at UCLA because of the different control blade design and the central graphite island.

MILLARD L. WOHL
PROFESSIONAL QUALIFICATIONS

I am employed as a nuclear engineer in the Accident Evaluation Branch, Division of Systems Integration, U.S. Nuclear Regulatory Commission, Washington, DC. My duties are to conduct site and accident analyses and various other safety-related studies for nuclear power and non-power reactor facilities. I was the contract monitor for the Pacific Northwest Laboratory work leading to NUREG/CR-2079, Analysis of Credible Accidents for Argonaut Reactors and the Los Alamos National Laboratory work leading to NUREG/CR-2198, Fixed Temperatures in an Argonaut Reactor Core Following a Hypothetical Design Basis Accident (DBA).

I attended Case Western Reserve University (formerly Case Institute of Technology) and received a B.S. degree in Physics in 1956. I received an M.S. degree in Physics from Indiana University in 1958. I did graduate work in Nuclear Engineering at Columbia University and Case Western Reserve University from 1962 through 1964. I was a teaching assistant in Physics at Indiana University from 1956 - 1958. I have taught physics and mathematics in the evening divisions of Baldwin-Wallace College, the Ohio State University and Cuyahoga Community College from 1958 - 1973.

In 1958, I joined the NASA Lewis Research Center in Cleveland, Ohio. My initial duties involved the writing of Monte Carlo computer codes for the determination of radiation shielding requirements and propellant heating for proposed nuclear-powered rocket designs. Other assignments involved methods development and shielding and nuclear safety analyses for numerous proposed mobile nuclear vehicle applications. Numerous technical publications evolved in the course of this work. Additionally, during the period 1958 - 1973, I had substantial research contract management responsibilities.

In 1973, I joined the General Atomic Company in La Jolla, California, as a nuclear engineer. At General Atomic I performed a variety of nuclear safety-related analyses for the High-Temperature Gas-Cooled Reactor (HTGR). These included the analysis of depressurization accidents and containment integrity studies, as well as computer code upgrading and modification.

In 1975, I joined the Accident Analysis Branch in the Division of Technical Review, U.S. Nuclear Regulatory Commission. My responsibilities involved site characteristic studies and accident analyses. Presently, I have similar but expanded responsibilities.

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THE HEALTH PHYSICS ASPECTS OF THE SL–1 ACCIDENT

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Abstract—With so few accidents in the Atomic Energy Industry, new concepts and procedures of interest to the Health Physics profession will result from a critical analysis of each major radiation accident. A brief description of the reactor, its purpose, and operating history prior to the accident is followed by a comprehensive treatment of the health physics activities. Emergency personnel working under adverse conditions in radiation fields, ranging up to 800 r/hr, received whole body exposure doses up to 27 r. Many unique pr blems were associated with the recovery and decontamination of the bodies. Environmental monitoring revealed that the airborne radioactive material was essentially all iodine 131 and was well below the Radioactivity Concentration Guide value for the offsite population. Analysis of soil and air samples indicated that the reactor building was quite effective in containing the fission products during and following the excursion. Included is data on direct radiation levels experienced in the vicinity of SL–1 and the rate of decay of the primary source. The major lessons learned from the accident are summarized.

INTRODUCTION

A ROUTINE Fire Department response to an alarm originating from a fixed temperature detector located on the ceiling of the operating floor of the SL–1 reactor building provided the original indication to the first fatal accident in the 18-year history of reactor operations. This nuclear accident involving a severe explosion and massive fission product contamination was the prologue to an extended rescue and recovery effort which introduced unique health physics and management problems with their accompanying frustrations and successes.

SL–1 DESCRIPTION AND OPERATING HISTORY

The Stationary Low Power No. 1 Reactor was the smallest known power reactor when it began critical operations in August 1958. The forty fuel elements were 91 per cent enriched uranium clad with aluminum-nickel alloy. Reactivity control was provided by five cadmium rods as well as burnable boron strips attached to the fuel elements. Fig. 1 is a cutaway layout of the reactor facility showing the control room, fan floor, operating floor, reactor vessel and top shield. This direct cycle, natural circulation, boiling water reactor was part of the Army program to develop simple and compact package power plants to be transported by air to remote Arctic sites. Following assembly by military personnel, the 3 MW (thermal) nuclear plant would produce 200 kW net of electric power for radar and other equipment, and 400 kW net of space heat for barracks, offices, and other installations. Other major objectives of the facility were reliability, 3-year core life, minimum plant costs, minimum maintenance and minimum operating personnel.

The SL–1 was operated by a cadre of twenty-five military personnel under the supervision of Combustion Engineering, Inc., with a staff of twenty personnel. Since the operating crew would be isolated by time and distance, special emphasis was placed on training personnel in multiple responsibilities. Two Army specialists were trained to be full-time health physicists capable of independent evaluation and action. In addition, all other personnel had received extensive training at Ft. Belvoir and at the National Reactor Testing Station in basic and advanced health physics and had qualified as health physics technicians. All members of the operating crew had been indoctrinated with the philosophy that operations were subservient to safety. At a remote location without fire, medical and other emergency support, the occurrence of even a minor incident was unacceptable.

Over a period of 26 months, the SL–1 operated nearly 11,000 hr for a total of 932 MW days. Before the accidental nuclear excursion, the plant had been shut down for the installation of flux wires for $11\frac{1}{2}$ days. All flux wires had been installed by 4.00 p.m. on 3 January, 1961, and the evening shift was scheduled to connect the controlled rod extensions with the drive mechanisms preparatory to start-up.

Fig. 2 is a map of the National Reactor Testing Station (NRTS) which shows the location of the SL–1 reactor with respect to other facilities at the NRTS.

POST ACCIDENT EMERGENCY ACTION

At 9.10 p.m. on 3 January, 1961, the seven man Fire Department crew from Central Facilities Area (CFA), responding to an automatic heat sensing alarm for the SL–1, found radiation fields of 200 mr/hr at the gate to the area which was located 200 ft from the reactor

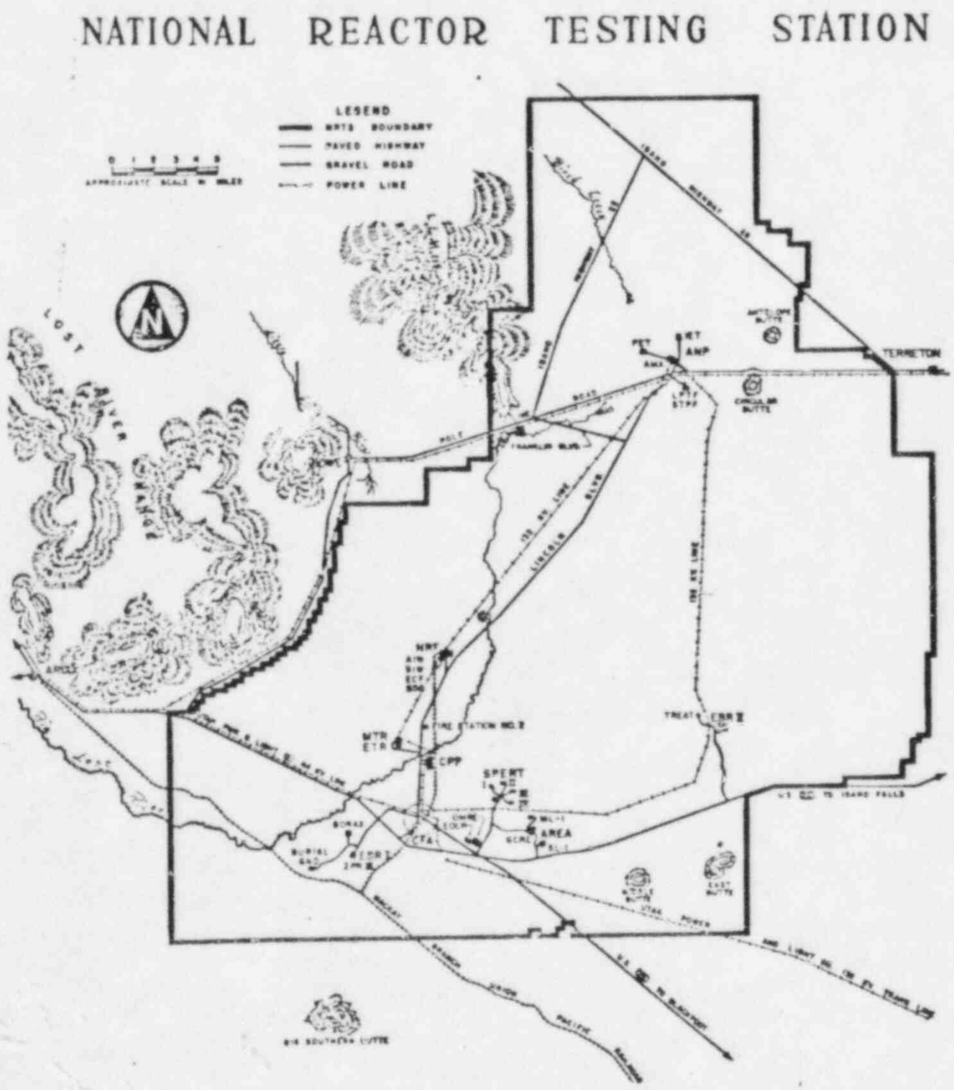

FIG. 2. Relative location of NRTS facilities.

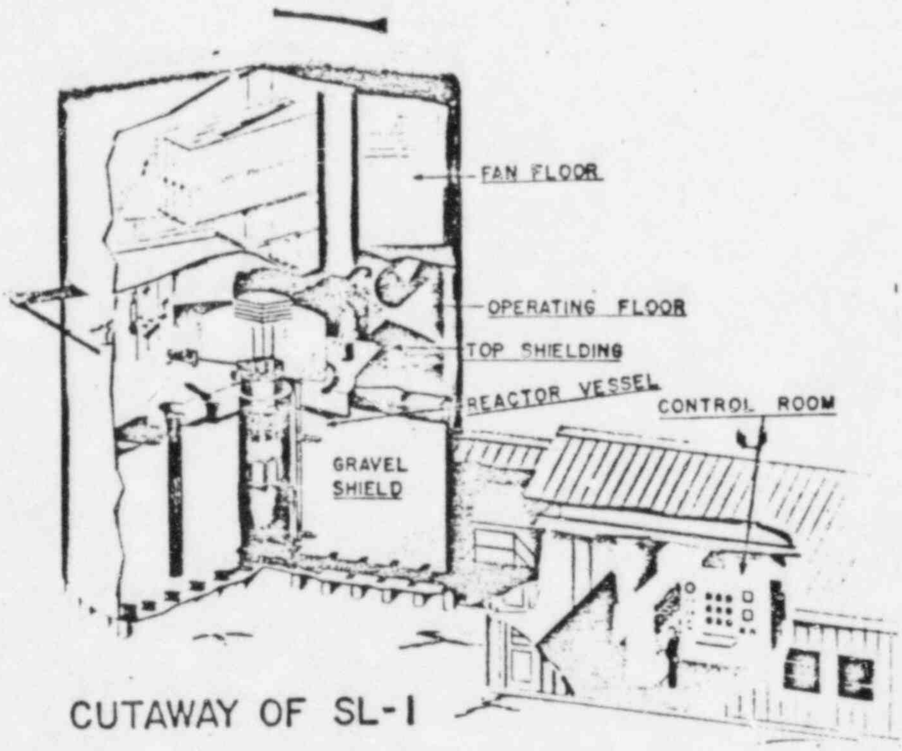

FIG. 1. Cutaway of SL–1 reactor building and control room. Stairway and entrance to operating floor are not shown.

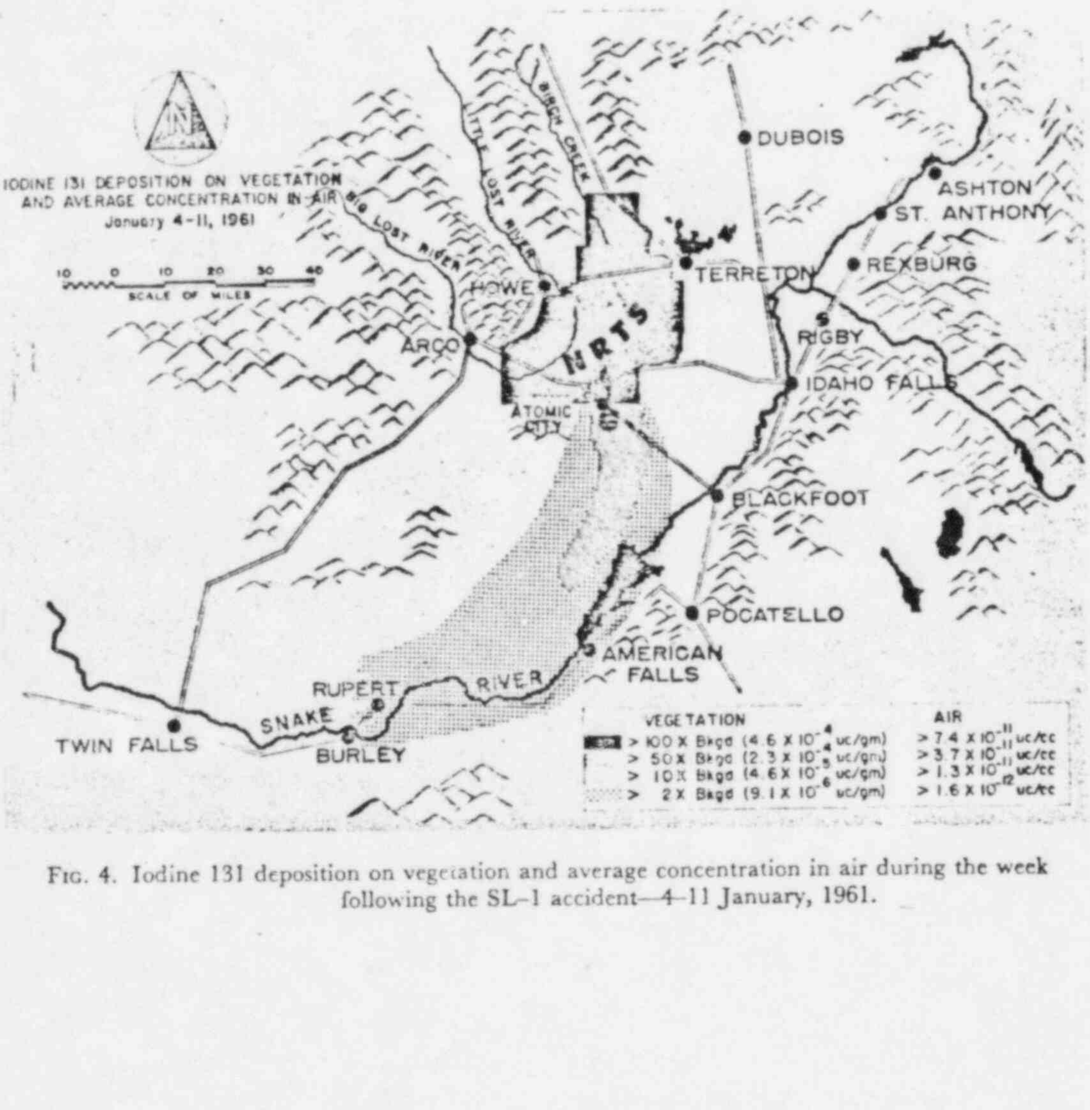

FIG. 4. Iodine 131 deposition on vegetation and average concentration in air during the week following the SL–1 accident—4–11 January, 1961.

building. A small quantity of steam was observed being discharged from the exhaust on the fan floor, and drifting to the west. A few minutes earlier the guard on duty at the gatehouse at the Gas Cooled Reactor Experiment facility, about 0.9 miles north-northwest of SL-1, noted the gateway monitor alarmed and could not be reset on any of the three ranges for approximately a 4-min period. A maintenance check revealed no malfunction of the equipment, but no particular significance was attached to the event.

Fig. 3 lists the radiation levels detected by the

RADIATION LEVELS IN THE SL-1 AREA

January 3, 1961

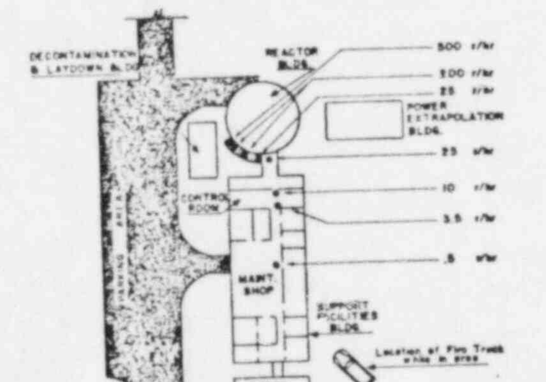

FIG. 3. Radiation levels in the SL-1 area.

emergency personnel at various locations. Attempts to contact the operators by telephone from the guard house were unsuccessful. Security patrolmen used pass keys to open the gates in the perimeter fence and the doors to the administration building. Firemen trained in nuclear fire-fighting entered the main buildings in an attempt to contact the operators and to search for fire. In addition to their regular Fire Department turnout clothing, these men were equipped with film badges, 200 r direct reading pocket dosimeters, rubber gloves, latex boots, and self-contained air supplied masks. Although every room was searched in fields up to 10 r/hr, none of the operators were found. Upon the arrival of experienced health physicists from the Materials Testing Reactor, a series of entries was made to progressively higher fields, until a penetration had been made to the top of the spiral stairs leading to the floor of the reactor building in fields of 500 r/hr. A brief scan of the room revealed silence, no response to shouts, limited damage, no fire and no personnel. Most of the emergency team believed the crew had evacuated; however, a telephone and radio check of neighboring facilities failed to locate any of the three-man crew.

At 10.30 p.m. a two-man rescue team entered the reactor floor and, beyond the 8-ton shield blocks, found the first two victims, one showing evidence of life. After re-grouping, a team of five determined that the second victim was dead and removed the critically injured operator. The AEC shift nurse from the Central facilities Dispensary was the first medical representative to arrive at the accident scene. She administered first aid until relieved by an AEC physician who arrived a few minutes later and pronounced the man dead

Radiation exposure to the 11 Fire Department personnel ranged from 0.03 to 1.07 r, indicating their ability at self protection in an emergency complicated by high radiation fields. Similar response by an untrained crew could have resulted in radiation injury and possible fatalities. Unfortunately, in the excitement to expeditiously rescue any survivors with minimum loss of time, the Fire Department personnel, trained in rescue operations, were overlooked. Instead, the rescue team was composed of supervisory personnel whose special competence would have been of greater value during later phases of the recovery effort.

Another four-man team, comprised of two Phillips Petroleum Company health physicists and two Army personnel, returned to the reactor building and found the third victim pinned to the ceiling directly above the reactor vessel. Since this man was obviously dead, the team immediately left the area. At this time, approximately 11.00 p.m., all action at the disaster scene was suspended and emergency personnel proceeded to the Gas Cooled Reactor Experiment area, about one mile north of SL–1, for initial decontamination.

During the first 2 hr, only the initial emergency crews observed routine contamination control procedures. Many other individuals entered the area wearing their ordinary street clothes because of the nature of the emergency. As a result, there was extensive spread of contamination from the reactor building into other buildings and adjacent areas and roadways. Thereafter, it was extremely difficult to re-establish effective contamination control. More contamination was spread through the area in the vicinity of the SL–1 by human and vehicular traffic than by the accident itself.

Everyone who entered the reactor building, or had come in contact with the first victim, was highly contaminated. For instance, radiation levels of 5 r/hr at 6 in. were common from hand contamination. After initial decontamination had been completed, these men were taken to the Central Facilities Dispensary or the Chemical Processing Plant for final decontamination.

All personal clothing worn in the area had to be confiscated and eventually disposed of in the burial ground. Contaminated spots measuring approximately 25 r/hr at 6 in. made any decontamination attempts impractical. The self contained breathing apparatus which was worn by everyone who entered the area was also highly contaminated; however, decontamination attempts were generally successful.

Although radiation fields just inside the entrance to the operating floor exceeded the scale indication of the survey instruments used (500 r/hr), it was estimated that the maximum field encountered was less than 1000 r/hr. The highest radiation exposures were estimated to be 20–25 r.

Everyone who entered the reactor buildings that night was handicapped by a visibility problem. The outside temperature was approximately 10° below zero, causing the respirator face-piece to fog over badly. In two instances condensate apparently froze in the supply valve, completely cutting off the air supply. This problem has been overcome by installing anti-fog inserts in the face masks of the respirators. This insert forms a seal across the nose and cheeks channeling the air back out through the exhaust valve.

By midnight, two emergency trailers had been moved to the control point which had been established at the junction of Fillmore Boulevard and U.S. Highway 20, about one mile south of the SL–1. All personnel entering or leaving the area were required to pass through the decontamination trailer where they were provided with health physics coverage, such as personnel metering and protective clothing, before entry, and monitoring on exit from the area.

The telephone company responded very efficiently and established telephone communication at the control point within hours of the incident. Flood lights and sanitary facilities were installed the following day. The flood of newspaper and magazine reporters, TV and press photographers, the curious and unofficial visitors plagued the early operations. It became necessary to relocate the control center 200 yd up Fillmore Boulevard and to establish a security post at the highway to limit access to the control point to those personnel having official business.

Dry runs and detailed personnel briefings prior to each major operation in the SL–1 area were used quite effectively after the initial rescue operations, and in some instances the use of mock-ups enabled individuals to practice their assigned functions and develop techniques which resulted in reduced radiation exposure doses. Photographs taken on the reactor operating floor after the accident proved to be an invaluable aid in briefing individuals emotionally for the grim task of body recovery.

The body of the second victim was recovered the evening of 4 January by a relay of two 2-man teams. An exposure limit of 12 r was established for this operation. However, the first team functioned beyond expectations and received doses of 9 and 4 r respectively, while the second team received less than 400 mr.

Recovery of the third body, which was pinned to the ceiling above the reactor, presented many difficult problems and was not completed until the early morning hours of 9 January. For the most part, this recovery was handled remotely and resulted in a maximum exposure of 6 r. As of that date, a total of 263 personnel had participated in recovery operations and received an integrated gamma exposure of 375 r.

SUBSEQUENT ACTIVITIES

The recovered bodies had been completely saturated with highly contaminated water expelled during the excursion. In addition, particles of fuel had penetrated the skin, resulting in large open wounds due to blast effect. Clothing removal had little effect in reducing the radiation levels which ranged from 100 to >500 r/hr at 6 in. Scrubbing with detergents was limited by personnel exposure considerations and was, therefore, only mildly effective with a decontamination factor of 2–3. Over a period of days, the continual melting of ice used to preserve the bodies was surprisingly successful, producing a decontamination factor of 20–40. Sheets of 1/8 in. lead were utilized to reduce the levels at the surface of the burial vaults, which were generally <300 mr/hr with hot spots up to 1 r/hr. A team of medical and health physics specialists from Los Alamos Scientific Laboratory performed the major role in the autopsies and decontamination of the bodies.

The detailed post-mortem examinations indicated the cause of death to be the direct result of injuries resulting from severe blast.

The bodies were flown by military aircraft to the airfield nearest the cemeteries selected by the families. Each body was escorted by an AEC Health Physicist, who supervised the handling of the burial vault until interment, making certain that no one received any significant exposure. The following requirements were placed on the burial sites:

(1) Burial in a perpetual care cemetery with adequate records of grave locations.

(2) The graves would not be reopened without the expressed permission of the AEC.

(3) The burial vaults would be surrounded by at least 12 in. of poured concrete and at least 3 ft of packed earth.

The total body and thyroid doses received by the 14 individuals who exceeded 5 r of penetrating radiation as a result of the SL–1 accident are presented in Table 1. The Radiation Exposure Guide established by Idaho Operations Office, Health and Safety Division, for emergency personnel was 25 r if the situation involved major loss of property, and 100 r if the loss of life was at risk. Although personnel entered radiation fields of 300–800 r/hr, the highest exposure from penetrating radiation was 27 r. The highest thyroid dose was 5.5 rad. Other internal exposures were 10 mrad over the first year from strontium-90, and 15 mrad total body exposure from cesium-137.

Table 1. Wholebody and thyroid doses received by emergency personnel immediately following the SL–1 accident

SL–1 Exposures greater than 5r Individual	Penetrating radiation in r	Thyroid dose from I^{131} in rads
(1) AEC Health Physicist	27	4.2
(2) Contractor supervisor	27*	1.2
(3) Contractor supervisor	25	0.6
(4) Contractor supervisor	25	1.2
(5) Contractor health physicist	23	5.5
(6) AEC project officer	21	0.0
(7) Cadre supervisor	18	2.0
(8) AEC physician	16*	0.5
(9) AEC nurse	15	0.6
(10) Support patrolman	11	0.5
(11) Support health physicist	11	0.4
(12) Cadre supervisor	9	0.7
(13) Support health physicist	7.4	0.6
(14) Army support	5.9	0.0

* estimated exposure

The operational philosophy of Idaho Operations Office following the initial rescue effort was premised on the following priorities:

(1) Positive and complete protection of all personnel involved in the SL–1 effort to prevent further injuries or loss of life and to minimize radiation exposure.

(2) Assurance that another nuclear excursion would not occur in the SL–1 reactor.

(3) Collection, analysis, and interpretation of essential data to assist in determining the cause and extent of the accident.

ENVIRONMENTAL MONITORING

Surprisingly, no serious exposures were received by rescue personnel, but even more striking were the results of environmental monitoring.

For 5 days during and following the accident, a general anti-cyclonic weather pattern existed. During 7 hr of daylight, light variable winds blew from the north to northeast direction with a capping inversion above 1000 ft. The remainder of the time, strong, deep inversions extended from the surface to 3000 ft.

A 6-hr aerial monitoring flight to the south-southwest of the NRTS was launched at dawn on 4 January, surveying over 500 linear miles of land surface. No contamination could be detected with the sensitive single channel pulse-height analyzer from an altitude of 500 ft. Three days later, during another flight, the highest activity recorded was 130 c/sec above a background of 200 c/sec. This was approximately 3 miles southwest of SL-1.

One of the first actions taken following notification of a radiological incident was the activation of an 11-station, high-volume air sampling network by means of a telephone signal. These sampling units consisted of a Staplex air sampler fitted with an MSA-2133 prefilter for collection of particulates and an activated charcoal filter for the iodine isotopes. One of these stations was located at Atomic City, the nearest off-site population (140 people). Gamma spectral analysis of the first sample collected there indicated that the cloud was essentially all iodine-131. At that location, about 5.3 miles downwind from the SL-1, the average iodine-131 concentration in air was approximately 5×10^{-11} $\mu c/cm^3$ for the first 16 hr following the accident.

Sagebrush sampling gave the first positive indication of cloud trajectory and low-order contamination to the south and southeast beyond the NRTS boundaries. Through 21 January a total of 220 vegetation samples were collected and analyzed for gross gamma activity. Vegetation samples collected on 4 January revealed a maximum contamination level off-site of approximately 4×10^{-4} μc of iodine-131 per g of sagebrush. At Atomic City, the vegetation contamination level was about one-third of the maximum, which was found 2 miles to the east. This indicates that the maximum iodine concentration in air off-site was about 1.5×10^{-10} $\mu c/cm^3$ for the first 16 hr following the accident. This was 50 per cent of the suggested Radioactivity Concentration Guide value of 3×10^{-10} $\mu c/cm^3$ for continuous exposure to the off-site population.

Fig. 4 shows the iodine-131 deposition on vegetation and average concentration in air at the NRTS and surrounding area for the first week following the accident. This is also thought to be the approximate path of the initial cloud. Of particular interest is the excellent correlation between vegetation contamination and the average iodine-131 concentration in air. This air data was obtained from a network of low volume air samplers which surrounds the NRTS. Each station is composed of a continuously operating pump which pulls air through a small carbon cartridge at a rate of 1 ft^3 per min. The cartridge consists of a 2-in. section of 3/4 in. plastic tubing packed with twenty mesh activated charcoal and fitted with an MSA-2133 prefilter. These cartridges are changed weekly and analyzed for gross beta activity and iodine-131. The vegetation data is the result of the vegetation samples which were collected by the Ecology Branch during that period.

Dairy farms are sparse in this section of sagebrush desert and lava rock terrain. A raw milk sampling program was initiated on all eight farms in the 150 square mile area where iodine-131 contamination of vegetation exceeded ten times background. Four out of seventy samples indicated 2×10^{-7} μc of iodine-131 per cm^3 of milk, which was the analytical detection limit. At this time of year, dairy herds are fed in feed lots, thus minimizing the effect of iodine deposition on the range land.

Two sheep from a herd grazing for 10 days 8 miles south of SL-1 were purchased and sacrificed. Thyroid-131 content was 0.1 $\mu c/g$. Thirteen jack rabbit thyroid samples were also analyzed. The maximum level was 0.14 $\mu c/g$ of wet tissue from an animal 4 miles south of the reactor.

With the assurance that there was no off-site hazard, attention was focused on the research and long-term studies in the immediate vicinity of the SL-1; namely, soil contamination and decay rate determinations. Since the reactor contained approximately 1 Mc of medium to long half-life isotopes and approximately 1200 gal of water had been lost from the pressure vessel, considerable interest centered around the effect of the 1/4 in. corrugated skin silo in containing the fission products during and following the excursion. It should be remembered that the reactor building was not designed to be a containment vessel.

Air and vegetation sampling indicated that ~10 c of iodine-131 were released during the first 16 hr, and ~70 c over the remaining 30-day period. From continuous air samples collected at Atomic City during this period, the calculated infinity thyroid dose for an adult was approximately 35 mrad, which is slightly greater than 1 per cent of the annual 3 rad recommended Radiation Protection Guide value for off-site population (FRC Report No. 2 dated 20 September, 1961 reduced this guide value to 1.5 rem/year). Fig. 5 indicates the rate at which this dose was accumulated.

A total of eighteen soil samples were collected from the 3-acre plot composing the project area

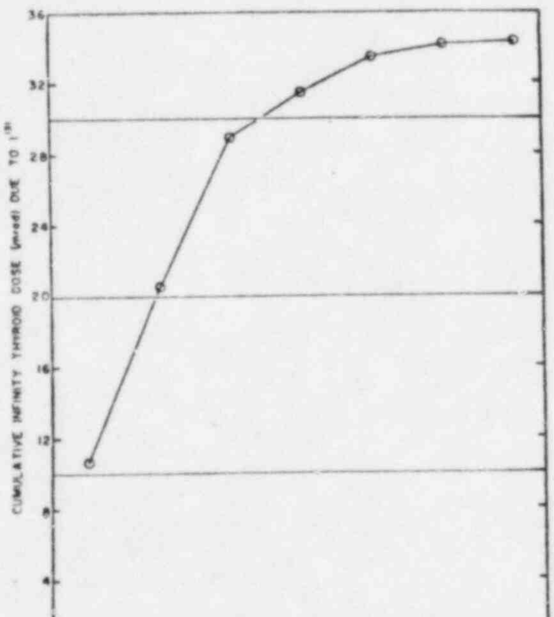

Fig. 5. Cumulative infinity adult thyroid dose at Atomic City due to inhalation of iodine 131.

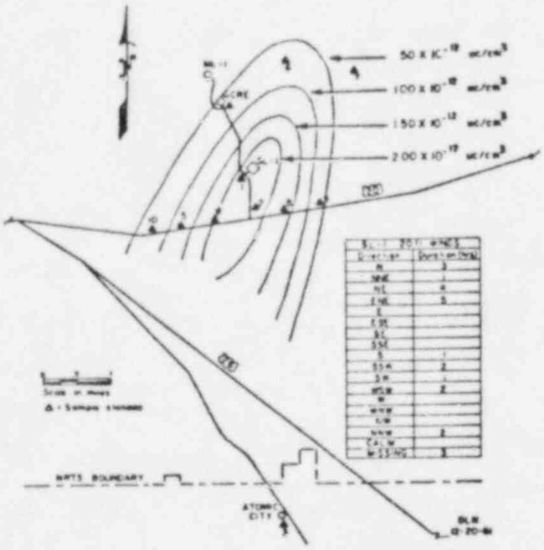

Fig. 6. Average iodine 131 concentration in air for a typical day approximately 2 weeks after the SL–1 accident.

within the SL–1 perimeter fence. An additional thirty-six samples were collected radially to a distance of 1/2 mile beyond the fence. Essentially all the radioactive material, with the exception of iodine-131, was contained within the 3-acre plot. Our best estimates indicate the release of approximately 1/10 c of strontium-90 and 1/2 c of cesium-137, with appreciably lesser quantities of zirconium-niobium-95, cerium-144 and barium-lanthanum-140.

Following the final recovery operation, a second air sampling network was established around the SL–1. Figure 6 shows the station locations and the average iodine-131 concentrations for a typical day during that period. Detectable concentrations of radioactivity in air existed at distances greater than 1 mile for about 8 weeks following the accident. However, at no time did this present a health hazard or operational problem within any of the other NRTS areas.

Film dosimeters located along U.S. Highway 20 for the purpose of measuring direct radiation from a radioactive cloud indicated an external exposure of less than 10 mr one mile south of SL–1.

A radiation survey grid was established around the SL–1 area on 5 January, 1961.

Direct radiation measurements were made at twenty fixed stations at varying distances in four directions, out to 1200 ft from the perimeter fence, utilizing Juno and G.M. type survey meters. Figure 7 presents the results of the

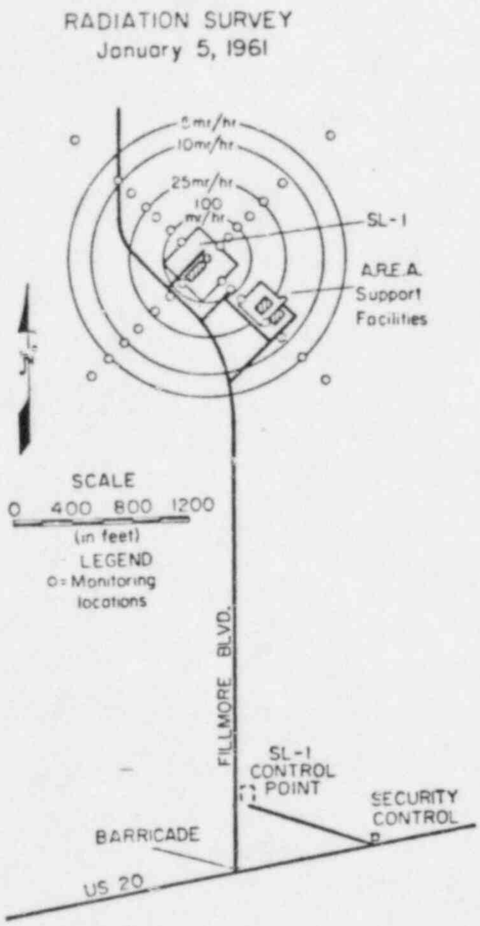

FIG. 7. Results of direct radiation survey performed in vicinity of the SL-1 on 5 January, 1961.

initial survey. At periodic intervals, data were obtained during the following 6 months until decontamination work was initiated on the reactor floor. Figure 8 is the composite decay curve for the four fence locations, and indicates the effective half-life had changed from about 30 days in January to about 120 days by the latter part of June.

EXPERIENCE APPLICABLE TO DISASTER PLANNING

Pre-planning is the recognized key to rapid and efficient disaster control and recovery. Unfortunately, it is psychologically impossible to motivate personnel to devote the amount of time to think, organize, train and equip themselves for some vague disaster which they believe will never occur, when real problems and work assignments are at their fingertips daily. Most individuals cannot gear themselves for pre-planning until they have personally experienced a major accident. In addition, there is the innate tendency to prepare for the accident which has already happened either last year at Oak Ridge, or Los Alamos, or Windscale instead of abstracting lessons learned from each experience and preparing an extremely flexible plan which can be tailored to any incident. The AEC Radiological Assistance Program has offered a rare opportunity to prepare such a flexible plan for any eventuality, from the recovery of a licensee's lost source to the recovery of the Nation from nuclear attack.

Disaster planning at the NRTS had been geared primarily to criticality-type of maximum credible accidents involving the release of thousands of curies of fresh fission products or iodine-131 to the atmosphere. Many of the unique types of problems experienced or suggested by the SL-1 accident had not been considered: performing recovery operations in radiation fields of hundreds of r/hr, medical treatment and decontamination of highly contaminated survivors and casualties, performing field operations around the clock for an indefinite period of time. Supplies and equipment, with but a few exceptions, were adequate. One of the first lessons was that survey instruments with a maximum range of 500 r/hr are inadequate for emergency use. Instruments with a maximum range of 5000 r/hr should be available. Available health physics personnel were depleted due to overwork rather than overexposure before all equipment had been committed, and this despite the full support of 5 NRTS contractors and radiological assistance from 4 AEC or military organizations outside the State of Idaho. During the first 11 days, 81 health physicists were utilized in the field and over 130 other personnel to provide them with field

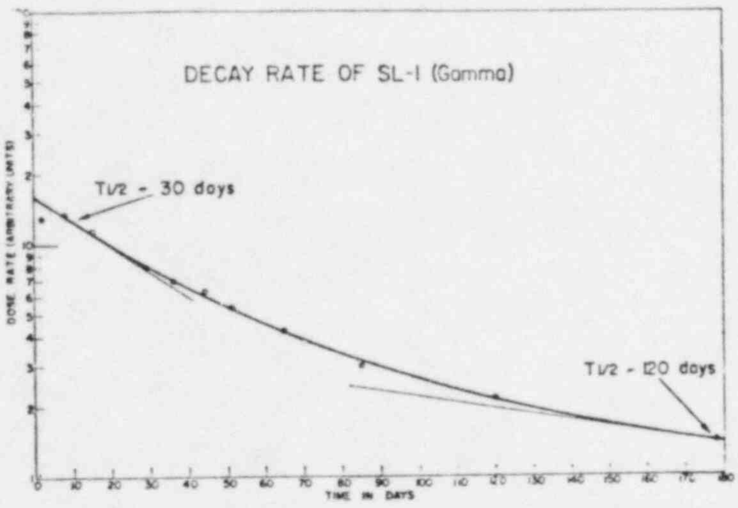

FIG. 8. Decay curve based on data obtained from periodic gamma radiation surveys on the SL-1 grid.

or laboratory support. Many individuals worked in excess of 120 hr per week, and yet sufficient personnel were not available to do the innumerable data collecting and research items which were desirable. The twenty-six-man radiological detachment from the Army Chemical Warfare Center at Dugway, Utah, provided early and effective field support.

In the training of health physicists, one must tinker with their brains and install "flip-flop" circuitry with a micro-second delay constant for a factor of 10^3 transition from routine operations to emergency response. In the flash of a second, their mental processes must switch from standards of 50 mr/day to 50 r in a few minutes, from minimum exposure to maximum performance short of injury, from slide rule calculations and considered opinions to estimates and instant decisions, from always safe standards to the brink of danger. In addition, this paragon of efficiency must have circuitry which is completely reversible. Once the emergency phase is consummated, he must make the rapid and difficult return to normal standards and practices. Some individuals could not accommodate themselves to emergency standards; others who made the transition had considerable difficulty in returning to pre-accident criteria. This situation of not understanding and appreciating emergency standards is also common to management, operating personnel, labor, the press and the general public.

Respiratory equipment must be the best available. In extreme cold weather, moisture from exhausted air may condense on the glass of the face-piece and freeze over, leaving only a dime-size opening just above the air intake holes. A nose cup is available which prevents any exhalation air from contacting the eyepiece. Thus, no vapor can condense on the glass. Communications between respirator equipped members of an emergency team is difficult, particularly in the dark. Light-weight radio units are available through vendors for use by personnel.

Exhaust valves freezing or sticking in the open position due to moisture condensing and freezing on the valve can occur during prolonged operations in cold weather, or rapid and extreme temperature changes. This can be overcome by cleaning the valve, holding the face-piece tightly and blowing, opening the emergency bypass valve, or installing special low temperature exhaust valves.

In the selection of personnel for emergency operations in hazardous fields of radiation, consideration should be given to the employment of older personnel when other considerations such as experience and agility are equal. Employees over 45 years of age offer the following advantages: a backlog of unused radiation exposure, less risk of genetic damage and a lesser impact upon their professional careers if future exposure restrictions are required.

Another problem experienced was the amount of negligence and abuse to which the health physics instruments were subjected. In order to combat this in the future, an equipment trailer for better control of field instrumentation will be established. This too, is a problem which requires indoctrination of personnel.

The SL-1 accident pointed up the shortcomings of the health physics training which had been received by the NRTS firemen; however, one shudders to think what might have happened had they not received any training. These people are now being trained to work in higher radiation fields, and to be completely self-sufficient if the situation should require it.

Another major problem was contamination control. As a result of vehicle and personnel traffic in and out of the SL-1 exclusion area, the access road and area immediately surrounding the SL-1 became highly contaminated. The contamination was primarily particulate and composed of aged fission products.

A considerable amount of effort was required to decontaminate the access road. The two most effective decontamination methods used were vacuum cleaning and high pressure water.

The prompt and factual release of information on the accident played a major role in the lack of sensationalism in the press and the acceptance by the public of the credible fact that accidents will occur in the nuclear industry since an element of risk exists in every human endeavor.

Acknowledgment—The authors gratefully acknowledge the assistance of the numerous members of the Health and Safety Division of Idaho Operations Office who have contributed to this work in various ways.