The Truth about the SL-1 Accident — Understanding the Reactor Excursion and Safety Problems at SL-1

On the night of January 3, 1961, the SL-1 nuclear reactor, a prototype for a military installation to be used in remote Arctic locations, exploded, killing the three-member military crew. The crew had been performing the routine process of re-assembling the reactor control rod drive mechanisms during a reactor outage. The SL-1 was a small 3 Mega-Watt-thermal (MWt) boiling water reactor, complete with a turbine-generator and condenser designed to generate both electric power and building heat. [1]

The SL-1 was designed, constructed and initially operated by Argonne National Laboratory. It was located at the Idaho National Laboratory, then called the National Reactor Testing Station. Combustion Engineering became the operating contractor for the Atomic Energy Commission (now the Department of Energy) for SL-1 on February 5, 1959. The crew that night were young military men in their twenties, all with families. There were three 8-hr shifts working around the clock. Combustion Engineering's request for funding to provide staffing to supervise more than the 8-hr day shift had been denied by the AEC, the agency that was predecessor to the Department of Energy. The three-man crew on the evening shift had a large number of tasks to perform on the operating room floor and there was no one in the control room to keep logs books, monitor plant conditions, or inform emergency responders of plant radiation status or the whereabouts of the crew.

The SL-1 core contained 14 kg of 93 percent enriched Uranium-235 in 40 aluminum fuel assemblies. Control of the reactor used 5 cruciform-shaped cadmium control blades raised and lowered by control rods. There were 4 outer (shim) control rods, labeled No. 1, 3, 5 and 7, and one center control rod, No. 9, to regulate reactor power. In addition to the control rods, burnable poison was added in the form of aluminum-boron strips, both half-length and fuel length, that were spot welded to the sides of the fuel assemblies. See Figure 1.

Key Things to Remember About What Caused The SL-1 Accident:

1. Poor design of the reactor provided significant vulnerability to over-withdrawal of the center control rod during power operation and shutdown maintenance due to its excessive reactivity worth.
2. Radiation-induced damage to reactor materials was causing swelling and deformation in core materials at an increasing rate during the last month of operation.
3. The SL-1's history of frequent control rod sticking was downplayed and ruled out as a cause of the accident long before the damaged core was closely examined. Possible reasons for reduced clearances or caught-edges at the height the rod was at when the over-lifting occurred were not explored. Numerous weld and material discontinuities exist in the area of the rod and shroud interface where the rod was being lifted up into the shroud that could have hindered free movement of the rod.
4. The extra effort required to jerk free the 84-lb stuck rod could easily have resulted in the roughly 16 inches of over-travel to achieve the 20-inch withdrawal from normal scram position — and in less than a third of a second.

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[1] Various DOE reports released by Freedom of Information Act request about SL-1 are at http://www.id.doe.gov/foia/archive.htm

The SL-1 had first gone critical on August 1, 1958 and by July 1, 1959 had accumulated 160 Mega-Watt operating days (MWD). By August 21, 1960, the reactor had accumulated 680 MWD. The boron strips were deteriorating and bowing of the strips made removal of fuel assemblies difficult; therefore, fuel inspections simply ceased. [2]

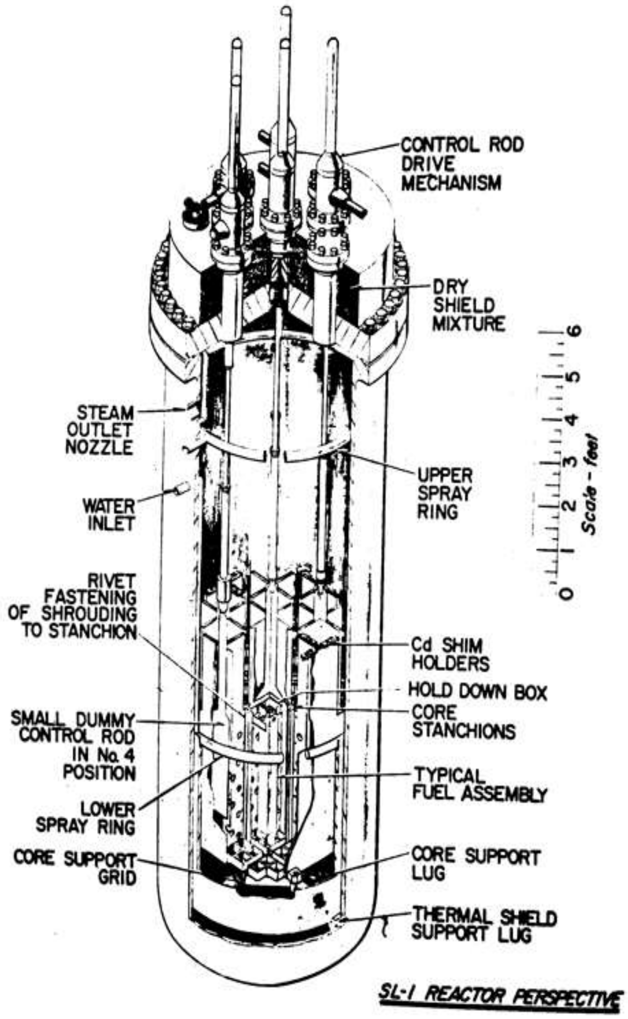

***Figure 1**. SL-1 Reactor perspective from IDO-19311.*

The routine maintenance procedure for the SL-1 to re-connect control rod drives after work above the core, such as installing flux monitoring wires, required manually lifting, twice, each of the 84 lb rods, that included the cadmium control blades, connecting and extension rods and upper-most portion called the rack.

Fire alarms activated automatically from heat detectors in the building, signaling fireman and other emergency responders to drive several miles to the Army Reactor Experimental area. There had been two false alarms earlier in the day that had required firemen to drive to SL-1, but the 9:01 pm alarm would bring responders into any but a routine situation. And the fire alarms could not be distinguished from radiation alarms.

The center control rod would later be found withdrawn 20 inches relative to the normal scram position. [3] Numerous accounts would say it was greater than this distance, including the Department of Energy's "Proving the Principle" which incorrectly states it was manually withdrawn 26 ½ inches. [4] Of the 20 inches it was withdrawn inside the core, it was initially already withdrawn by at least 2 inches

[2] Atomic Energy Commission report, Idaho Field Office IDO-19300, "SL-1 Reactor Accident on January 3, 1961: Interim Report." Combustion Engineering, May 15, 1961.

[3] Atomic Energy Commission report, Idaho Field Office, IDO-19311, "Final Report of the SL-1 Recovery Operation, General Electric Co., June 27, 1962. partial center rod withdrawal of 20 inches, p. 146.

[4] Susan Stacy, "Proving the Principle – A History of the Idaho National Engineering and Environmental laboratory, 1949-1999," Washington, D.D.: US Department of Energy. p. 148. http://www.inl.gov/publications/ and http://www.inl.gov/proving-the-principle/introduction.pdf

and probably by 3 inches. [5] The operator needed only to bend down, clasp the vertical shaft and ease the 84 lb rod up an additional inch or two, wait for his co-worker to remove the C-clamp, and then lower the rod back down.

While the mechanism for the severe explosion was not immediately apparent, it was found that the center control rod (No. 9) had been lifted too high—high enough for the reactor to cause a steam explosion from the "prompt critical" rapid generation of neutrons that heated the reactor fuel, vaporized some of the fuel and flashed the water in the reactor vessel to steam.

A new core and rod drive mechanism was scheduled to be installed in the spring of 1961.[6] The new rod drive mechanism would have eliminated the need to manually raise a control rod during the coupling operation. Later examination of the core internals would also identify numerous pre-accident weld, corrosion, and material issues in the damaged core. [7]

The Rod Withdrawal Distance for Prompt Criticality Was Unknown

After the accident, reports would state that a reactivity addition of 2.4 percent delta k/k had put the reactor on a 4-millisecond period. While sounding innocent enough, the reactor design allowed manual movement of a single control rod to insert a huge amount of reactivity change rapidly enough to cause the accident. Prior to the accident, no one had computed the prompt criticality rod withdrawal distance. The 4-millisecond neutron population doubling would mean such a rapid increase in neutrons that the heat generated in the fuel could not be transferred to the coolant water before some of the plates would vaporize from the high temperatures.

The complex and irregular arrangement of burnable boron strips made modeling the SL-1 core particularly difficult. Before the accident, the calculations for predicting normal criticality for reactor operation and the corresponding control rod withdrawal positions for achieving criticality were based on greatly over-simplified computations because of the difficulty in analyzing the complex non-symmetrical geometry of the core. In fact, even studies attempted today find the complex arrangement untenable. The simplified analysis had deviated significantly from the actual observed reactor core control rod positions needed for reaching criticality for normal power operation. [8]

Reactivity shutdown margin is known to change over time with reactor burnup. Very little monitoring to compare predicted to actual reactivity shutdown margins was performed at SL-1. As a prototype, its unproven design should have resulted in more, not less attention than is ordinarily performed at reactor facilities. Such monitoring was hindered by lack of staff and inaccuracies in recorded conditions including errors in accurately zeroing the control rod drives.

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[5] ibid. IDO-19311, p. III-109]

[6] ibid. IDO-19300. p. 4.

[7] Atomic Energy Commission report, Idaho Field Office, IDO-19313, "Additional Analysis of the SL-1 Excursion: Final Report of Progress July through October 1962. Flight Propulsion Laboratory Department, General Electric Co., November 1962. p. 147.

[8] ibid. IDO-19300. p. 34-36.

Post-accident calculations would require tedious and imprecise delving into operating records to try to account for the previous month's operation. [9]

Reactivity shutdown margin and the reactivity worth of each rod are affected by core geometry, individual fuel element history, water temperature, Xenon decay and in the SL-1, also by the status of the deteriorating boron strips. Estimates of reactor shutdown margin and rod position to achieve the "prompt critical" condition that would destroy the reactor would later be extrapolated from non-identical conditions and revised in later, somewhat overlapping SL-1 accident reports. [10]

Abnormal Core Degradation Required Taking Steps to Increase Shutdown Margin

The observed degradation of shutdown margin was finally so large that in November of 1960 that special cadmium strips were installed in two positions, on the east and west sides of the reactor to provide additional neutron poison. These core positions are referred to as tee rod positions 2 and 6, although the shims were stationary. Approximately 1.1 % additional negative reactivity was inserted by the cadmium shims in two positions. It would increase the amount of required for center rod withdrawal to achieve criticality by about 2.3 inches. While addition of the cadmium shims was a necessary step to ensure the shutdown margin, the original calculation for the cadmium shims would have been at four positions I will refer to as north, south, east and west, providing a more balanced power distribution. [11]

Although this would not have caused the accident, it is identified in the post-accident examination of the core that on the east side of the core, the east position at control tee-position 6, where three cadmium strips were installed, that one of the three strips was found 2 ½ inches below the proper position. That this was a pre-accident core loading mistake is not discussed. However, the consequences of this mispositioning are suggested by the distribution of damage in the core. [12] See Figure 2.

The core, almost 26 inches high, was a checker board of square shapes but the fuel position approximated a filled cylinder shape. The geometry, looking down on the core, was symmetrical. The power in this core should have been symmetrical, unlike test reactor cores designed for varying power in various lobes in order to simulate higher powers for materials testing such as the Advanced Test Reactor. Maintaining symmetry would have reduced power peaking in different areas of the core, prolonged fuel life and put lower stresses on the fuel during an accident.

The choice to put the cadmium strips in the #2 and 6 tee positions (which can pictorially be thought of as east and west positions, see Figure 2) while providing sorely needed additional shutdown margin had the effect of reducing power on the east and west sides, but of increasing reactor power on the north and south sides of the reactor. After their installation, reactor

[9] ibid. IDO-19300. p. 49.
[10] ibid IDO-19313.
[11] ibid IDO-19300. p. 52.
[12] ibid IDO-19311. Chapter 3, Appendix D and E.

operating power oscillations had increased. The power levels and fuel damage from the accident are highest in the central area of the core. But fuel plate powers were higher in the north and south than the east and west because of the two rather than four cadmium shim positions used.

Even with the two instead of four cadmium shim locations, one would have expected the north and south fuel assembly powers of the core to be symmetrical: they were not. The effect of mispositioning one of cadmium shims, with its three strips filling the tee slots on the east side of the reactor can be seen in the higher power of the fuel assembly, No. 58 during the accident, next to what I presume is the mispositioned shim. Fuel assembly No. 58 has a higher power, peaking at a higher elevation in the core, than the fuel assembly south of it, fuel assembly No. 60. [13] [14]

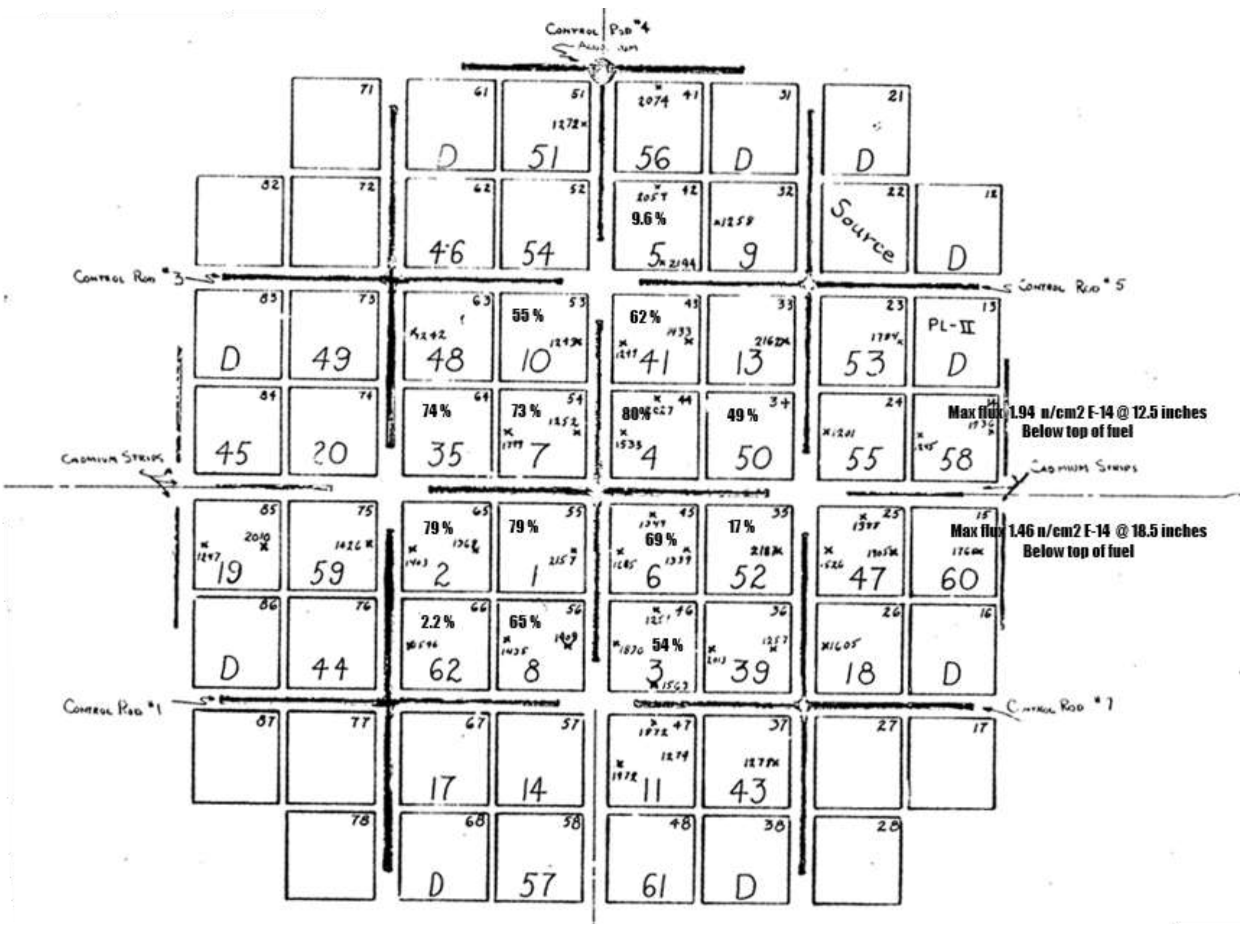

Figure 2. *SL-1 fuel assemblies, core positions, and percent destroyed, i.e. just northwest of center, fuel assembly 7 in position 54 has 73% of the fuel destroyed. Adapted from IDO-19311, figure III-89.*

There appears to be another problem with core configuration not pointed out in the reports. One of the central fuel assemblies, position 55, had been replaced with fuel assembly (Fuel

13 ibid IDO-19311. Figs III-70, III-89 through III-92, Appendix D and E.

14 ibid IDO-19313. Appendix E- Supplement to IDO-19311.

assembly No. 1) which had only 253 MWD of burnup.[15] This made it a higher power fuel assembly, generating more neutrons and increasing its power level and that of its neighboring fuel assemblies during the accident. The higher fuel plate powers generated in the south-west portion of the center of the core can be seen by comparing fuel powers achieved during the accident of fuel assemblies 2 and 52. Fuel assembly 2 was 79 percent destroyed during the accident, while the symmetrically positioned fuel assembly No. 52 was only 17 percent destroyed. [16]

While neither of these problems caused the accident, they are problems that would have increased the likelihood of melting fuel during an accident. And with this reactor, not only were materials corroding, and materials deforming from increasing neutron bombardment of power operations, the control rods were not reliably inserting. While some of the problems were attributed to control drive mechanisms, other problems involved sticking due to mechanical interferences between the blade and shroud that the blade slide inside, in the fueled region of the core. Add to this the loose tracking of reactivity margins, the wide error bands on scram setting for overpower, design vulnerabilities inherent with the high reactivity worth of the central control rod—any delay in insertion of the central rod would have been especially likely to have allowed an overpower condition during power operations to result in melting fuel.

One month earlier, the critical and prompt critical withdrawal distances would have been about 1 ½ to 2 inches lower than they were at the time of the accident. And while the reactivity loss from the degrading boron strips had been addressed by adding the cadmium shims, the bulging of the boron strips on the center control rod shroud may have contributed to the rod being stuck. [17]

"Boiling Noise" During High Power Testing

By December 21, 1960, the SL-1 had accumulated 932 MWD, and despite the flaking boron strips, the difficulty removing and inspecting fuel assemblies, and frequently sticking control rods, tests were being conducted at higher than rated power, pushing the reactor to the point of power control instability. The tests involved powers of 4.7 MWt in order to test the performance of a newly designed condenser. "The testing was limited since permission had not been granted at that time to operate the reactor at power levels over 3 MW[t]." [18]

"An approach to the limit of the stable operation range and incipient instability of the reactor occurred in November, 1960 during a program to increase the operating power level to 4.7 MW in order to test the recently installed PL type condenser." [19] At these higher reactor power levels, automatic movement of the center rod was not able to maintain a steady power. Installation of the cadmium strips in the east and west positions had worsened the instability.

[15] ibid IDO-19300. p. 46.
[16] ibid IDO-19311 p. III-96 and Appendix D.
[17] ibid IDO-19300. p. 69.
[18] ibid IDO-19300. p. 72.
[19] ibid IDO-19300. p. 4-5.

A reactor scram on overpower had occurred on November 23, 1960. The scram setting was 5.7 MW and the scram was estimated to have occurred between 6 and 8 MW. [20] These reactor power oscillations, known as "boiling noise" were swinging pens off the paper recorder charts and had prompted an automatic scram. This would likely explain the comment remembered later by the spouse of the second victim about his concern that the reactor might blow up.[21]

Obscured History of Sticking Control Rods

The issues with stuck control rods at the SL-1 were downplayed in numerous Department of Energy reports. The center control rod by design had a much higher effect on reactivity in the core than the other control rods and after the accident, it was found to have been withdrawn while the other rods were not. It was emphasized in official reports that the center control rod had performed better than any of the other rods. They failed to mention, however, that the center control rod had stuck on 7 occasions, while the other rods had each stuck between 8 and 12 times. And the frequency of control rod problems had increased from about 2.5% (up until November 18, 1960) to 13% between November 18 to December 23, 1960. On December 23, 1960, when SL-1 was shut down as planned, only two of the five rods dropped cleanly to the bottom of the core: No. 5 and center control rod No. 9. The others had to be driven in by their drive motors. [22] [23]Causes of the stuck rods varied from malfunction of the control rod drives to issues of clearances due to radiation-induced swelling of materials inside the reactor.

In addition to control rod sticking during scram and rod drop tests, the rods had a history of sticking during rod withdrawal: a total of 10 rod withdrawal sticking incidents—four of them had occurred during the last month of operation. The specific rods that stuck during withdrawal were not identified but would likely have been identified in the logbook records. [24]

Little emphasis was given to the sticking prevalent in the shutdown position when a portion of the control blade extended below the core and the shroud. As the lower portion of the control blade was pulled up into the shroud, evidence would later be found of pre-accident "scouring" marks on more than one blade. [25] It points to serious material condition and design flaws reducing the clearances needed for control blade movement.

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[20] ibid IDO-19300. p. 74.

[21] William McKeown, *Idaho Falls: The Untold Story of America's First Nuclear Accident.* Toronto: ECW Press, 2003. p. 201.

[22] IDO-19300, p. 62-63, Table V, p. 62, and Appendix A (half of pages missing in online report as of 12/2014).

[23] Todd Tucker, *Atomic America — How a Deadly Explosion and a Feared Admiral Changed the Course of Nuclear History*, Free Press, 2009.p. 115.

[24] ibid. IDO-19300. Table V, p. 62.

[25] ibid. IDO-19311. p. III-57 and III-62.

Numerous Conditions to Cause Sticking of the Control Rods During Withdrawal

The boron strips spot welded to fuel were buckling and also flaking off—and completely missing in some places. This reduced reactor shutdown reactivity margin and also created debris that could bind a control rod blade. Any work above the open reactor tank could also allow a small piece of material to fall into the tank and block a fuel channel or restrict movement of the control blades.

The control blades slid up and down in metal shrouds that had been designed with rows of 2-inch diameter weep holes cut in the sides. These openings in the shroud could allow debris such as flaking bits of the boron strips to interfere with control blade movement.

Center control rod No. 9 was completely surrounded by fuel assemblies each containing two aluminum-boron strips. It was known that control rod shrouds could have been distorted inward, creating frictional resistance for the control blades to slide freely from the swelling or buckling of the boron strips. [26] Official reports would say that deterioration of the boron strips played no role in the accident, but it was one of several credible causes of control rod sticking relevant to the accident.

The non-poison bottom follower ends of the control blades were designed to be lowered below the core, exiting the shroud that encased the cadmium-filled portion of the blades in the core when the rods were fully inserted during shutdown. With the control rod drives disengaged, the upper portion of the blades rested 3.7 inches lower than the normal scram position relative to the core. The outer rods had a 5-inch extension below the cadmium poison part of the rod, and the center rod had a 17-inch bottom extension which extended 15 ½ inches below the shroud. The follower of the center control rod had various welds and discontinuities that may have been especially likely to have caught on an upper edge of a 2-inch weep hole in the shroud as the control blade was being lifted during control rod drive re-assembly.

Interference by catching an edge of a 2-inch diameter shroud weep hole near the bottom of the shroud would provide a reason for the rod to get stuck in the lowered position during rod re-assembly despite its free movement upward several inches from the disengaged position.

The drawings of the two-year-old reactor were already acknowledged to not be up-to-date. [27] There are differences in how the length of the center rod follower are measured, but apparently the follower was about 17 inches long. [28] [29]

The operating contractor had been aware of the problems with the boron strips and with control rod sticking. The early conclusion that boron swelling and rod sticking had no role in the accident was made prior to knowledge that the vessel had jumped 9 ft and before close examination of the damaged core. Given how often the control rods had been sticking during two

[26] ibid. IDO-19300. p. 68-70.
[27] ibid. IDO-19300. p. 16.
[28] ibid IDO-19300. Figure 39 (and Figure 10 depicting 19 inches of follower to the active cadmium).
[29] ibid IDO-19311. p. III-57 states that that 17-inch follower extended 15 ½ inches below the shroud.

years of operation, the likelihood of a stuck rod even during rod withdrawal should have been expected; it was anything but a remote possibility.

The key conclusion that it would require an intentional or highly reckless act in order to manually overlift a stuck rod enough to cause the accident is not actually stated nor is any technical description given to support this conclusion. The AEC analyst, C. Wayne Bills, later interviewed in *Idaho Falls – The Untold Story* [30] said that the mock-ups for control rod lifts showed that a stuck rod would not be overlifted by more than about 10 inches. But the mock-up did not actually have blades in a shroud, and the man pulling on the rod did not have to free it. To simulate a stuck rod, they freed the rod "unexpectedly" while the operator was attempting to lift it. There were plenty of problems with the simulation that I can see, and if it had a more solid basis, they would have described it in the technical reports. [31]There was plenty of incentive to reach the conclusion that freeing a stuck rod was not the cause of the accident so they could discount the importance of pre-accident conditions that they were aware of.

The results of the control rod mock-ups for measuring the distance of rod travel and the speed of travel are documented in a table and figure. The finding was that a casual effort as well as a maximum effort in lifting the control rod achieved adequate speed for the prompt criticality condition and that there would be no time for correcting an overlift because it would occur in less than a third of a second. [32]

The black-and-white photos available on-line of the post-accident examination of the SL-1 core are difficult to examine, but the series of 2-inch diameter weep holes in the shroud can plainly be seen. [33] These holes provided ample opportunities for an edge to catch the sliding control rod, causing it to stick, requiring a hefty jerk on the 84 lb rod. As stated above, four of the ten rod withdrawal sticking incidences had occurred in the last month of operation when cycling high power operations would have accelerated the radiation-induced deteriorating material conditions of the numerous structures in the core. [34]

Destructive BORAX-1 Testing Experience Not Heeded Prior to SL-1 Accident

After the accident, the experience inferred from testing boiling water reactors, including destructive testing at BORAX and SPERT was studied along with tests of a mock-up of the SL-1 control rod assembly, and it was concluded that, yes, the amount of reactivity needed to go prompt critical could be added manually by lifting the center control rod. The speed of a manual lift was a sufficiently rapid rate to produce a reactor "excursion." [35] Note, however, that some of the discussion in this early portion of the report is in error because later examination of the core confirmed that the center control rod was bound in its shroud at 20 inches of withdrawal. The

[30] ibid. *Idaho Falls: The Untold Story of America's First Nuclear Accident.* Toronto: ECW Press, 2003. p. 185, 179-186.
[31] See evaluation of manual rod withdrawal in IDO-19300, p. 154-157 and also IDO-19311.
[32] ibid. IDO-19311. p. III-109 through III-111, Table III-10, Fig. III-96.
[33] ibid. IDO-19313.
[34] ibid. IDO-19300. p. 62.
[35] ibid. IDO-19300 p. 155.

active length of the fuel was 25 7/8 inches. The initial position of the rod, relative to the normal "zero" or scram position for reactor operation also must be taken into account. [36] Statements such as the rod had to be withdrawn "nearly the full length of the rod" were based on the initial look at the center control rod and shroud which had been expulsed from the center core region.[37] With 2 to 3 inches of height above the normal scram position in order to install the washer and nut, less than 18 inches of travel remained to reach the 20 inches length.

A 50s vintage documentary film by the AEC presenting the BORAX tests [38] states, "The [BORAX] experimental reactor was built for the purpose of testing this self regulation [reactor power reduction due to steam formation] and its most important consequence—the inherent safety of the reactor. The reactor is inherently safe against the accidental addition of any amount of excess reactivity ***which can be removed by the formation of steam before the power rises to a dangerous level***. [Emphasis added]" The need to pay particular attention to the last caveat would be demonstrated again by the SL-1 accident.

In numerous non-destructive BORAX tests, the rapid power increases would result in a geyser of water out of the tank that was placed out in the open air. The final test, designed to reach prompt criticality and expected to be catastrophic, was observed from a safe distance and did not disappoint. The rapid prompt critical condition results in rapid fuel melt and steam explosion with an upward expulsion of the core and control rods high into the air. The release of fission products caused about as much concern as would the setting off of fireworks at the fourth of July.

Interestingly, many of the BORAX tests increased reactivity by dropping the water temperature in the reactor tank. Investigators of the SL-1 accident would later comment that the SL-1 accident, with water initial temperature of 90 to 100-degree F increased the peak power by a factor of 10 what it would have been had the water been at saturation temperature. [39] [40]

The narrator in the BORAX film states: "Extension of experimental data to such a condition was considered important **even though the accidental addition of so much excess reactivity to an operating reactor has almost negligible probability**. **Addition of so much reactivity is not easy, for unless the ejected control rod is very large and is moved rapidly**, **the reactor will shut itself down** by steam formation before the desired amount of reactivity has been added. [Emphasis added]"

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[36] ibid. IDO-19300, Fig. 39.

[37] ibid. IDO-19300. p. 156 "so large a withdrawal of the rod – corresponding to nearly the full length – might have been made" is based on the as-found position of various portions of the center control rod after the accident. But, rod movement during the accident must be distinguished from rod movement that initiated the accident. At the time the statement is written, it is not known that the vessel jumped 9 ft nor that the blade is bound in the shroud at 20 inches of withdrawal. See IDO-19313. p. 146.

[38] Borax – Safety experiment on a Boiling Water Reactor. Film produced by the Argonne National Laboratory. Operated for the U.S. Atomic Energy Commission by the University of Chicago. circa late 1950s. The destructive BORAX-1 experiment was conducted in 1954 at the Idaho site.

[39] ibid. IDO-19313, p. 151.

[40] ibid. IDO-19300. p. 132 in contrast to IDO-19313, says the temperature in the reactor vessel was 73 F based on log entries, an even worse situation for providing heat transfer to reactor fuel plates.

The safety analysis for the SL-1 did not include consideration of any accident involving melting of fuel and release of fission products, let alone destruction of the reactor from a prompt criticality achieving a total energy release of 133 MW-sec. [41] [42]The fuel cladding of the SL-1 reactor was twice as thick as the BORAX design—and other aspects of the fuel design had made it more susceptible to reaching a prompt critical condition than the BORAX reactor. It would be determined that the SL-1 reactor needed only 2.4 percent delta-K compared with the 3.3 percent delta-K reactivity insertion for the BORAX-1 destructive test. [43]

It was known with the BORAX experiments that movement of a rod of sufficient reactivity worth, in a few tenths of a second, could result in increasing the reactor power so rapidly that neutron population doubling occurred in milliseconds. Such rapid power increase in the fuel from fission heated the fuel plates in the SL-1 reactor "to a point near or above melting, depending upon location in the core. In the center regions of maximum neutron flux, the fuel within the plates experienced vaporization temperatures and burst the plate cladding. Thus, the spewing of hot vaporized fuel rapidly produced steam in the surrounding water. The steam was generated at a rate far faster than could be dissipated. . ."

The SL-1 Reactor Vessel Had Jumped 9 Feet

It would not be discovered until months after the SL-1 accident that not only had the reactor ejected various missile projections, but the vessel had jumped 9 ft, shearing connected piping. The engineers discovered that they would not need to cut piping in order to remove the vessel from the building, for transport of the damaged core to a hot cell facility, now available after cancellation of the Aircraft Nuclear Project.

Subsequent evaluations determined: "The steam being generated pushed upon the water that was above the level of the core forcing the slug of water upward from the core zone. It was stopped by the vessel head with the resultant water hammer causing peak pressures of about 10,000 psi. While the water was moving upward, the core structure jumped reaching a height of 7 inches above its supports when the water hammer hit the head. As the water was decelerated upon striking the vessel head, the forces generated collapsed the shield plug guide tube. It also deformed the vessel wall and the vessel head nozzle. Additionally, the momentum of this water as it stuck the vessel head transferred its energy to the reactor vessel imparting a vertical motion to the shield plugs and to the vessel itself. . .. The vessel jumped approximately 9 ft shearing the connecting pipes and expelling some of the surrounding thermal insulation. Simultaneously with the vessel lift, the pressure within the vessel expelled the unbolted shield plugs." [44]

[41] ibid. IDO-19300, p. 170.
[42] ibid. IDO-19311. Table III-I.
[43] ibid. IDO-19311. p. IV-25.
[44] *The SL-1 Accident: Phases 1 and 2*. Film produced by the Idaho Operations Office of the US Atomic Energy Commission, *The SL-1 Accident: Phase 3*. Film produced by the Idaho Operations Office of the U.S. Atomic Energy Commission, circa 1963.

Center Control Rod Blade Withdrawn 20 Inches

Months after the accident, a lower portion of the metal shroud that surrounded the center control rod would be found compressed around the control rod blade and described as 20 inches of withdrawal relative to the normal scram position for power operations. [45] Of the 20 inches (plus or minus a ½ inch) it was withdrawn, it was initially already withdrawn by at least 2 inches for the rod drive re-assembly. The operator needed only to bend down, clasp the vertical shaft and ease the 84 lb rod up an additional inch or two, wait for his co-worker to remove the C-clamp, and then lower the rod back down.

The earlier step in the procedure that required lifting the center control rod (attached to the blade) about 6 inches above the low disengaged position had already been performed, because the stop washer and nut were found installed on the center rod. The handling tool had been reinstalled on the rod for the final lift required in order to remove the C-clamp.

The control rod initial position would still have remained the 2 to 3 inches above the normal scram position. This left an additional 17 to 18 inches to reach the as-found control rod blade withdrawal position. A possible ½ inch of rod movement during the excursion was acknowledged as possible and the 20-inch withdrawal distance uncertainty was plus or minus ½ inch. So, jerking the center control rod if it did not move freely, an over-travel of as little as 16 inches in less than a third of a second was all that was needed for the accident to take place. See Figure 3.

Reading various reports can cause plenty of confusion on these distances. First of all, the rod movement for achieving criticality is always described from the point of view of a fully assembled rod drive in the zero or normal scram position rather than the position during maintenance. Next, the various components of the control rods have different positions relative to the core (the region of the fuel assemblies) and the reactor vessel which is the cylinder-shaped tank that houses the fuel and most of the control rods that extended up to the top of the vessel, exiting the vessel to connect to control rod drives.

The cruciform-shaped control blades are inserted in the core (or the region of the fuel assemblies). The blades are connected to a connecting rod, extension rod, and the rack which protruded from the vessel. They were all connected and lifted as a unit prior to the accident. But during the accident, after the center control blade was lifted to the height causing the prompt criticality, the resulting heating and melting of fuel, expulsion of melted fuel caused rapid heating of water that caused a steam explosion. The steam explosion moved the upper connecting rods and shield plugs that enclosed the upper portion of the control rods above the reactor vessel in a complex series of events. The guide tube which collapsed around the extension rod and resulting scratch marks indicated that the upper portion of the rod had moved 26 ½ inches relative to the shield plug. The complex discussion would lead many readers astray

[45] ibid. IDO-19311, p.III-57 and Figure III-58.

because what mattered was the distance the control rod had been manually lifted (20 inches) to initiate the accident. [46]

Attempts to derive how high the control blade had been lifted based on the position of the extension rod and the rack after the accident caused considerable confusion, as the fact that the vessel had jumped 9 ft was not known when the first official report was written. [47] At that point in time, they were trying to figure out what had caused the accident, and they could see the withdrawn center control rod No. 9 in its shroud, lying above the core in the reactor vessel. The derivation of rod withdrawal distance based on evidence of the upper portions of the connecting rods was complex and often misleading.

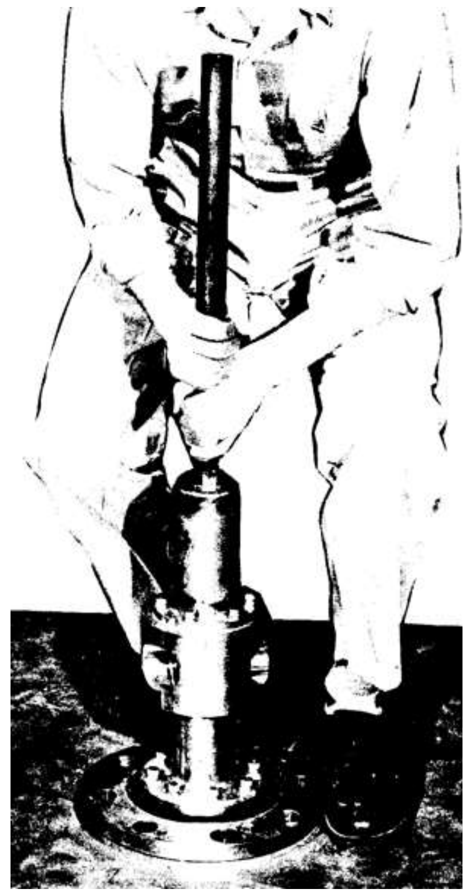

__Figure 3__. Mock-up of manual rod lift from IDO-19300.

Scratch marks made by the upper portion of the connecting rod when it hit the ceiling and was jammed in were initially thought by some people to be clear evidence that the rod had been maliciously yanked out of the shutdown position. [48] [49]

It would finally be determined that the center control rod had been withdrawn about 20 inches from the core and that by manually lifting the center rod, it could be withdrawn far enough and fast enough to cause the prompt criticality that would destroy the reactor. [50]

Official reports would emphasize that the center control rod needed only to be lifted a small amount in order to remove the C-clamp. And that the procedures and training were clear that the lift height must be restricted. The mock-up tests subsequent to the accident of a man lifting the 84 lb rod showed that achieving the necessary distance and speed did not require maximum effort. The entire lifting motion would take less than a third of a second—time enough for the rod to move roughly 20 inches—and no time to respond to prevent the over-lift. [51] The reports describe in depth the mock-up results of time to withdraw the rod, requiring modest not extreme effort and that there was no time available to respond to prevent overlifting.

The careful statements that "we may never know why the rod was lifted too high" implied that only by intentional effort would

[46] ibid. IDO-19311, p. III-14.
[47] ibid. IDO-19300.
[48] ibid. IDO-19311. p. III-14, III-18, Table III-1 explain why scratch marks from the guide tube on the upper portion of the control extension rod do not reflect the distance the control rod was manually withdrawn from the core.
[49] ibid. *Proving the Principle*. p. 148 provides the initial but later proven to be incorrect information about the scratches on the upper portion of the control rod as well as the incorrect rod withdrawal distance of 26 ½ inches.
[50] ibid. IDO-19311, p. III-18, see also IDO-19313, p. 146.
[51] ibid. IDO-19300 p. 155-156.

the rod be lifted too high. In *Proving the Principle* people interviewed continue to assert that the rod was pulled too high maliciously. [52]

The official reports mention but dismiss the possibility that the rod was inadvertently jerked upward causing a prompt critical condition—preferring instead to insinuate that only a deliberate action could have caused such a large rod withdrawal.

It was only after the accident that calculations were performed to estimate the speed and distance that the center rod would have to travel in order for the accident to occur. Results of these estimates evolved in the different reports developed over the two-year course of investigations. At cold shutdown, the estimate of the needed center Control Rod (No. 9) position (with all other rods inserted) was a center rod withdrawal distance from normal scram position of approximately 16.7 inches for the reactor to become critical and 17.6 inches for prompt critical which would cause immediate destruction of the reactor, consistent with the as-found 20 inch withdrawal. [53] The estimates of the position that would result in prompt criticality were acknowledged to be uncertain due to the inability to model the complex core configuration and the estimated distances vary in different documents.

Rumors of Deliberate Excess Rod Withdrawal

The rumors of a love triangle or of a man distraught over marital problems were widespread and live on in books and nuclear on-line blogs today. The rumors would naturally result from the months of delay while investigation proceeded. But the shifting of blame by official reports was rather subtle. In an AEC film describing the SL-1 accident, it is stated: “Direct cause of the accident clearly appears to have been manual withdrawal of the central control rod blade by one or more of the crew members —considerably beyond the limits specified in maintenance procedure. However, there was insufficient evidence to establish the actual reason for such abnormal withdrawal.” [54]

The careful omissions and the downplaying of the serious design, operational, and oversight problems with the SL-1 reactor reveal the strong undercurrent to avoid accepting blame for the accident.

The bodies of the three men were misidentified until autopsy many days after the accident. When it was initially thought that the man impaled to the ceiling was the new trainee, it was assumed that he had made a mistake. When it was learned that the supervisor of the crew, was the one impaled, it was unofficially assumed that his co-worker with marital problems had deliberately pulled the center rod. While the initial report following the accident concluded based on autopsy that the third victim, the crew supervisor, had lifted the control rod, [55] this report

[52] ibid. *Proving the Principle*. p. 148-149.
[53] ibid. IDO-19311, b, p. III-107.
[54] *The SL-1 Accident: Phases 1 and 2*. Film produced by the Idaho Operations Office of the US Atomic Energy Commission and *The SL-1 Accident: Phase 3*. Film produced by the Idaho Operations Office of the U.S. Atomic Energy Commission, circa 1963.
[55] ibid. IDO-19300. p. 101-102.

would have been accessible to few people. The unofficial presumption that the second victim had lifted the rod lived on despite the autopsy report that his hands were undamaged and relatively uncontaminated, inconsistent with having his hands on the control rod. [56]

Death Three Different Ways

The men, initially misidentified following the accident, were victim number 1, Richard L. McKinley, trainee, age 27, who was given attempted resuscitation in the ambulance; victim number 2, John A. Byrnes, Specialist 5, U.S. Army, age 22, the man with marital problems who was thrown from reactor top area,; and victim number 3, Richard C. Legg, U.S. Navy, shift supervisor, age 26, who was impaled to the ceiling above the reactor.

The neutron fields the workers were exposed to would have killed them as would the high gamma radiation fields, had the explosion not killed them. The experts who would be investigating the accident did not know that the reactor damage that occurred could be solely due to the reactor. Consultants were hired to see if an explosive device had been placed in the reactor. [57] So, while the crew would likely have thought that the reactor would go critical if the rods were withdrawn too far, they likely did not expect an explosive accident to have occurred while the reactor was initially shutdown.

Would a crazed worker have decided to die a slow agonizing death and kill his fellow workers by exposure to high neutron fields of an unshielded reactor gone critical? I seriously doubt it. And remember, even those investigating the accident did not guess that the vessel had jumped 9 ft until they discovered this months after the accident.

A worker at Los Alamos who died in 1958 from exposure to an accidental criticality suffered an agonizing 35 hours. He was autopsied by the same man, Clarence Lushbaugh, who would be brought to examine the men who died in the SL-1 accident. That victim, Cecil Kelley had previously been exposed to plutonium, and this was an opportunity to see if his plutonium uptake matched their predictions made by the Los Alamos laboratory. Lushbaugh would later be sued, along with the University of California that ran the Los Alamos Medical Center, for performing numerous autopsies without obtaining permission from the families, in order to determine plutonium uptakes. Autopsies were performed not just of former laboratory employees but of citizens living in the Los Alamos area. [58]

SL-1 Mystery Solved

[56] ibid. *Idaho Falls: The Untold Story of America's First Nuclear Accident.* The radiologic survey of the victims on page 128 misidentifies the men in the figure title, but shows the minimal contamination to the second victim's hands.
[57] ibid. IDO-19311. Appendix C.
[58] ibid. *Atomic America*, p. 164.

As Todd Tucker points out in his book, *Atomic America*, the army's reactor program completely lacked the demanding oversight of all aspects of design, construction, and operation that Admiral H. G. Rickover had provided to the naval submarine nuclear program. The many unsafe conditions at SL-1 recognized by people experienced with nuclear operations, then and later, were not discussed publicly because it would be seen as counter to promotion of nuclear power industry. It could also limit the longevity of one's career in the nuclear industry.

Examination of the SL-1 core after the accident would reveal faulty welds, excessive corrosion and buckled boron strips. It would be noted in the final report [59] that all materials used in a reactor should undergo rigid quality control and materials testing in both selection and fabrication—and materials should be periodically examined to confirm their suitability. The edict that poor material condition and sticking control rods played no role in the accident stuck—and with flimsy basis. And the detailed examination of the damaged core did not attempt to discuss burs, swelling, or discontinuities that might have produced sticking or reduced clearances in the vicinity where the final manual lift took place. At this elevated height during maintenance, 2 to 3 inches above the normal scram position, as the blade follower was being retracted up into the shroud, no discussion of the potential for sticking is discussed. However, photos and drawings of the blade and shroud do suggest that there were ways for reduced clearance or snagging to have caused the rod to stick.

When you consider the faulty welds and flaking boron, the increasingly sticking rods during the last month of operation, the ambitious testing at relatively high reactor powers in order to test the new condenser which would have accelerated the degradation of materials, it becomes clear that management decisions caused the sticking control rod which, in my mind, undoubtedly caused the accident.

I believe that deliberately ambiguous statements placed blame for the accident on the crewman who lifted the control rod in order to shield the contractor and AEC. These young men, I believe, were doing their best to get on with the long list of tasks to do that fateful evening. They had lacked the seniority and broader perspective on safety shortcomings to have demanded changes at the facility.

The complexity of the accident and the long months of investigation would play a role in the speculation of what caused the accident. But carefully crafted statements had two objectives (1) divert blame from the AEC and its contractors and (2) do nothing to undermine public faith in the nuclear industry.

The careful omissions and downplaying of a multitude of serious design, operational, and oversight problems emphasized in official reports about the SL-1 reactor are revealing. Its way past time to acknowledge that the rod stuck and it was jerked free by a tired worker put in harm's way by a multitude of poor design and safety management decisions that the crew had no control of.

[59] ibid. IDO-19313 p. 127.

You can read my report about the consequences of the SL-1 accident on the Environmental Defense Institute website, *The SL-1 Accident Consequences*, at http://environmental-defense-institute.org/publications/SL-1Consequences.pdf

Report by Tami Thatcher, former nuclear safety analyst at the Idaho National Laboratory and nuclear safety consultant. She provided safety analysis and probabilistic risk assessment for the Advanced Test Reactor at the INL. The ATR is used to exposure materials to a specified high neutron environment for materials testing. Unlike a commercial nuclear power reactor, at the ATR, core configurations change significantly and often, generally every few weeks. Although she was not a core safety analyst, her perspective has partly been shaped by her association with the reactor operations and engineering organization that monitored core reactivity predictions for normal and off-normal conditions in order to ensure adequate fuel cooling during unplanned reactivity insertions.