XIX

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 (UCLA Research Reactor)	Docket No. 50-142 (Proposed Renewal of Facility License)

DECLARATION OF PROFESSOR MICHIO KAKU

I, Dr. Michio Kaku, do hereby declare as follows:

1. I am Associate Professor of Theoretical Physics at the City College of the City University of New York. A statement of my professional qualifications is attached hereto.

2. I have performed an assessment of credible accidents for the UCLA Argonaut-type research reactor. This assessment included a site visit, review of the relevant scientific literature and of the following documents of record in the above-captioned proceeding: the contentions admitted to the proceeding by the Atomic Safety and Licensing Board, the accident analysis included in the 1980 application for license renewal and the 1982 amendments thereto, the original Hazards Analysis for the UCLA reactor, the reports on Argonaut-type reactors by Hawley et al and by Cort (NUREG/CR-2079 and -2198 respectively), the Neogy memorandum ("Transient Analysis of the UCLA Argonaut"), and relevant portions of the NRC Staff's Safety Evaluation Report. Certain other documents reviewed are identified below.

3. It is my conclusion, based upon the afore-mentioned review and assessment, that there are a number of credible accident scenarios for the UCLA reactor, each of which could result in substantial fission product release to the environment.

4. The categories of accidents examined include: severe power excursion, fire, explosion, core-crushing and/or flooding, out-of-reactor criticality incident, chemical reactions such as metal-water reaction or severe cladding corrosion, and fuel-handling incidents. Attention has been paid to multiple or common mode failure sequences that could result in accidents involving more than one of the above categories (e.g., seismically-induced event involving both core disruption and fire). Initiating events examined include both mechanical and human elements, the latter considering both accidental and intentional acts.

5. As I understand the Atomic Safety and Licensing Board has requested that technical terms and concepts be explained so as to aid the general public in understanding the issues involved, this statement begins with a brief explanation. Its brevity necessitates considerable simplification of rather complex concepts and should be viewed as such.

Brief Explanation of Terms and Concepts

6. The chief hazard associated with nuclear reactors arises from the extremely toxic nature of the products of nuclear fission. These fission products are intensely radioactive and can pose very considerable danger to the public if ever exposed. In order to prevent that, most reactors engage in multiple, redundant barriers to fission product release in an effort to ensure that, if ever an accident were to occur, little if any of the radioactive material would reach the environment. These barriers include the fuel cladding, the pressure vessel, the containment structure, and a series of engineered features to enhance the effectiveness of the containment, such as containment sprays, ice systems, radioactivity removal systems, filters and the like. Much of the debate over nuclear safety has focused on the effectiveness of these features.

7. In addition to multiple barriers to fission product release, most reactors have substantial exclusion zones and low-population zones surrounding them so as to permit significant dispersion of the radioactive material, if released, before it reaches members of the public. Concentrations drop by several orders of magnitude within the first quarter mile or so, so keeping the

nearest person at least that distance away, and providing that densely populated areas be considerably further away provides a measure of protection.

8. Some research reactors have containment structures and exclusion zones, though not as large or complex as power reactors. The UCLA reactor has neither.

9. In the case of the UCLA Argonaut-type research reactor, the primary barrier to fission product release is the cladding on the fuel, aluminum fifteen thousandths of an inch thick. (Application, page V/1-4). If the cladding were damaged, fission products would be released with essentially no other barrier preventing them from reaching the public. The small size of the fission product inventory relative to that of large power reactors is essentially compensated for by the lack of exclusion zone and high population density immediately adjacent to the reactor room and the lack of multiple barriers or other means of mitigating fission product release should it occur. The amount of fission product release would depend on the degree of damage and the temperature of the fuel. At or near the melting point, fission product release would be very substantial.

10. The UCLA fuel meat and cladding (as well as the control blades) are among the lowest-melting such materials used in reactors of which I am aware. It thus becomes very important to ensure that no accident can occur which can elevate the temperature of these materials to close to their melting point.

11. The primary reactor materials-- particularly the graphite, uranium metal, and magnesium-- are all potentially combustible. Should they catch fire, fission products, including large quantities of particulate material, would be driven out of the fuel and into the environment. Thus any condition which might result in fire would present a serious threat.

12. Certain events can also initiate violent chemical reactions or explosions which can disassemble the core and release significant portions of the fission product inventory. Steam explosions, metal-water reactions, or placement of explosive materials in an irradiation port within the core or merely nearby the reactor can induce such disassembly, perhaps initiating an incendiary reaction as well. Furthermore, numerous chemicals

attack aluminum; corrosion of the cladding can thus penetrate the primary barrier to release of the radioactive materials contained inside.

13. Mechanical damage to the fuel, as initiated in an earthquake which crushes the core or through a fuel-handling accident, can damage the cladding and release some of the gaseous fission products inside.

14. Finally, there is a kind of accident peculiar to reactors which is essentially a reactor runaway, where the power escalates by many orders of magnitude in extremely short periods of time. This can result in melting of the fuel, violent steam and chemical explosion, and destruction of the reactor core. A less severe version of this accident, called a "criticality accident", can result when uranium or plutonium is accidentally brought into the right configuration to go "critical" outside the reactor, resulting in an intense radiation burst. Such accidents have occurred at the rate of about one per year in the U.S. nuclear industry.

15. Because there aren't multiple barriers to fission product release, and because of the dense population with no exclusion zone, UCLA must, to be licensed, demonstrate that no credible accident could occur at its facility that would result in release of more than a very small fraction of one percent of the reactor's radioactive inventory. Release of even a very small fraction of the core inventory would be devastating given the site characteristics.

16. UCLA has attempted to make that showing by asserting that the maximum credible accident at its facility is mechanical damage to <u>one</u> of the 264 fuel plates of the reactor, <u>after</u> three weeks of "cooling down" has permitted the radioactivity in that plate to decay quite considerably, and assuming only 2.7% of the remaining Kr-85, Xe-133, I-131 and I-132 are released, all other isotopes remaining in the fuel. (Amended Application, p. III/8-12). UCLA has thus taken as its maximum credible accident the release of 0.0344 curies total. (<u>id</u>., Table III/8-2, column 1; 0.0024 + 0.017 + 0.013 + 0.0020 = 0.0344 Ci). This is less than one ten-millionth of the core inventory. (Computer run, portions attached, performed by UCLA indicate inventory goes to 344,000 curies after a single 8-hour run; maximum core inventory is considerably higher).

17. Thus, the question to be examined is whether there are any credible accident scenarios that result in fission product releases substantially in excess of 1×10^{-7} of the core inventory. A release of 10% of the inventory from the fuel--as shall be shown, a not-unreasonable estimate for several classes of accident at this facility--would produce consequences at least a million times greater than those assumed by UCLA for its asserted maximum credible accident. (It should be noted that, in addition to assuming an extraordinarily small fission product release for its design basis accident, UCLA has used a number of extremely unrealistic assumptions for estimating dispersion. For example, although Technical Specification 3.4.1.2. mandates that the ventilation system be shut down automatically in an emergency involving excessive radiation, the UCLA analysis assumes continued operation of the ventilation system and thus substantial dilution of effluent which will not exist in an emergency. These inappropriate calculational methods, when corrected, would increase consequences an additional order of magnitude or two.)

18. The issue, then, is not whether the inventory of the UCLA reactor is so small as to be inconsequential in case of accident. It appears established that release of a substantial fraction of that inventory would produce unacceptable consequences, given the lack of other barriers and the site characteristics. The issue appears to be whether any credible accident scenario exists which could result in release of more than a small fraction of a percent of the core inventory. The answer, as shall be demonstrated below, is in the affirmative.

19. One additional prefatory comment is in order. UCLA, in its amended application, at page III/8-1, appears to assert that only power reactors can have "Maximum Credible Accidents," and that by virtue of being a non-power reactor, the UCLA Argonaut-type reactor is automatically immune from any accident which could result in release of a substantial fraction of its inventory. That is simply not true. In fact, if one examines the history of reactor accidents that have involved fission product release, the vast majority of those to date have occurred in non-power reactors. Of U.S. commercial power reactors, the Fermi reactor and the Three Mile Island Unit 2 reactor are the primary instances. (It is interesting to note that the official estimate of iodine release to the environment is within the same order of

magnitude at TMI as the Hawley study assumes for a fuel-handling incident at UCLA involving one cold fuel element; the inhalation doses, moreover, would be considerably greater at UCLA because of the lack of exclusion zone.)

20. The history of non-power reactor accidents involving severe fuel damage is far more extensive than that for power reactors to date, largely because of the lack of engineered safety features, the lack of standardized design, the experimental nature of the program involving such reactors, and the far smaller degree of safety research focused upon them. The list of non-power reactors suffering major accidents involving substantial fuel damage includes the SL-1, the SRE, the NRX and NRU, the HTRE-3, EBR-1, and Windscale, to name a few. Furthermore, the SPERT and BORAX tests clearly demonstrated that such reactors can suffer substantial fuel melting and even complete core disassembly. It is interesting to note that all of the above reactors are of the same vintage as the UCLA Argonaut-type reactor. The lack of a reactor vendor still in business to provide updating, spare parts, and expertise exacerbates the problems regarding the UCLA reactor. The lack of a system for identifying problems with similar reactors and passing on lessons learned and requiring appropriate backfitting makes the facility, in addition to being relatively primitive from a safety standpoint due to its design prior to major advances and insights in the field, something of an operating relic. Most of the changes that have been made have been in the non-safe direction: ten-fold increase in power and fission-product inventory; very substantial increase in excess reactivity, to a dangerous level; pneumatic tube system for rapidly inserting and withdrawing reactivity; and others. We shall see below how such changes have increased the already considerable potential for serious accident.

Power Excursion

19. This discussion will begin with a brief explanation of criticality and reactivity; as indicated above, brevity will necessitate considerable oversimplification.

20. A reactor is said to be critical when the number of neutrons in one generation is equal to the number in the next. In such a situation, k, the criticality or effective multiplication factor, is said to be equal to unity, or 1. When the chain reaction is increasing with time, so that the number of neutrons in one generation is larger than the number in the previous, the reactor is said to be supercritical. A supercritical reactor is thus defined by $k > 1$. $k < 1$ means the reactor is subcritical.

21. A nuclear chain reaction runs on two different kinds of neutrons: prompt and delayed. By far the majority, approximately 99.35% for UCLA, are prompt, being produced virtually instantaneously at the moment of fission. A small fraction, approximately 0.65% are delayed neutrons, i.e. neutrons produced by decay of the fission fragments created by the nuclear fission process. These delayed neutrons are produced in periods of time ranging from microseconds to hours. If it were not for the delayed neutrons, mechanical control of a reactor would be impossible.

22. Neutron generation times are measured in milliseconds; if the only neutrons upon which the chain reaction is based were prompt neutrons, there would simply not be enough time for either human or mechanical intervention to prevent a runaway condition. Growth in reactor power (a function of growth in the neutron population) is essentially exponential. Even a small rate of growth from one generation of prompt neutrons to the next could cause reactor power to increase to such a level that fuel melting could occur long before human or mechanical intervention (e.g., insertion of control rods) could be completed in order to prevent such a runaway condition.

23. Such a runaway condition, where the neutron population grows uncontrollably, is called a power excursion. If the power excursion is severe enough (i.e., if power gets high enough before a shutdown mechanism can be activated), fuel melting can possibly occur as well as a steam explosion or explosive metal-water reaction. It is very important, therefore, that a reactor not be able to run away at a rate faster than its shutdown mechanisms can act; i.e., that it not become "prompt supercritical," or supercritical on prompt neutrons alone.

24. When a reactor is running on delayed neutrons--i.e., when the reaction needs both the 99.35% of neutrons that are prompt and the .65% or so that are delayed in origin--the delayed neutrons provide a margin of safety that permits intervention of control rods or other shutdown features in time to prevent an increase in power that is so rapid that melting can occur. The delay required for generation of these neutrons provides time for electronic indicators to report an abnormal growth in neutron population, inform a control operator who can take appropriate actions or activate an autocontrol which can mechanically do likewise.

25. However, when a reactor is running on prompt neutrons alone, it has lost that protection. An increase in neutron population can occur so suddenly, and continue to increase exponentially so rapidly, that intervention by human response or engineered safety feature is not possible. Thus, this situation is strongly to be avoided.

26. When a reactor is supercritical, i.e. $k > 1$, meaning that each generation of neutrons is larger than the previous, the power rise is exponential. The exponential period (T) is that amount of time it takes the power to increase by a factor e, or approximately 2.718. Thus, in five exponential periods, the power would rise by e^5 or about 150 times, for example. The ability of reactor power to rise astronomically on a very short period is thus evident, and explains why supercriticality on prompt neutrons can be so dangerous.

27. The effect of the delayed neutrons is to elongate the exponential period T quite substantially, giving time for human or engineered features to come into play before the exponential imperative brings a dangerous power level. But as a reactor approaches prompt supercriticality, the exponential period becomes exceedingly short, making possible massive power rises in very small fractions of a second, given by the following general equation

$$\frac{P}{P_o} = e^{\frac{t}{T}}$$

where P_o is the initial power and P is the power after the lapse of time t. For very short reactor periods T, then, very large power rises can occur in very short time intervals t. And when a reactor is supercritical on prompt neutrons, the period T becomes exceedingly short.

28. Thus, it would be quite incorrect to assert that prompt critical is just another point on the curve. Near prompt critical, the exponential period jumps from a manageable range measured in seconds or hours to periods measured in milliseconds, making engineered safety features such as control blades and dump valves potentially useless should the excursion go unchecked. (We will discuss the effect on any remaining intrinsic safety features shortly.)

29. A reactor can become supercritical on prompt neutrons-- or go "prompt critical"-- if sufficient "excess reactivity" is inserted in the reactor core (e.g. through addition of extra fuel or moderator or through removal of neutron absorbing materials such as control blades or samples that have been inserted into the core for experimental irradiation). The effect is the same whether positive reactivity is added (by dropping, for example, a sample of uranium-235 into an irradiation port) or removing a negative worth sample by pulling it out of the core-- the reactor "sees" the same thing either way, a flood of extra neutrons, which cause more fissions, which produce more neutrons in the form of the expanding chain reaction.

30. If sufficient excess reactivity is added (or negative reactivity removed) so that the delayed neutrons are no longer needed to get the reactor critical, the reactor is then critical on prompt neutrons alone and the exponential period, or e-folding time, becomes very short. Power is thus increasing on such a short period that there would potentially be no time to stop the reaction by mechanical means such as operator response or scramming of control rods automatically.

31. The capacity of a reactor to go supercritical on prompt neutrons is measured in terms of the available excess reactivity. For a reactor to be just critical requires, as we indicated above, $k = 1$. How much reactivity is available to push the reaction beyond just critical is the excess reactivity. Because the delayed neutrons represent approximately 0.65% of the neutrons in the UCLA reactor, if one adds excess reactivity of 0.65% or more, the reactor will be supercritical on prompt neutrons alone. The delayed neutron fraction is called β (beta) and that amount of excess reactivity is sometimes measured in units called dollars, with β = \$1. If the percent notation is used, the units are in percent of $\Delta k/k$.

32. In short, if a reactor has available more than \$1 or 0.65% $\Delta k/k$ excess reactivity, the resulting power rise (or "power excursion") might possibly occur so rapidly that the reactor's normal engineered safety features such as control blades would be unable to prevent the rise from attaining levels sufficient to melt the fuel and release the contained fission products, should that excess reactivity be for some reason inserted into the reactor.

33. If such a power excursion were to occur, power would keep increasing until the excess reactivity is somehow removed. This can occur in several ways-- for example, with low enriched fuel, the neutron capture rate will undergo substantial "Doppler broadening" as temperature in the fuel rises, reducing the number of neutrons available. This effect is virtually nonexistent in highly enriched fuel, and is a strong argument for converting the HEU at UCLA to LEU, for safety as well as non-proliferation reasons.

34. Other factors which can terminate a power excursion are heat transfer to the moderator, which reduces the effectiveness of the moderator (for certain moderators), or void formation in the moderator (creation of steam, for example, in a water moderator), or expulsion of the moderator (as from a steam explosion). Thermal reactors like most power and research reactors require a moderator to function-- some substance that slows down the neutrons to increase the probability of their causing fissioning in adjoining U-235 atoms. Without the moderator, or with less of it available, the reaction can't keep going. The final shutdown mechanism is disassembly of the core. (A destructive power excursion, in fact, is sometimes referred to as an "RDA" or Rapid Disassembly Accident.) Essentially the energy rise is so rapid and so large that the core explodes. This happened in the final SPERT and BORAX tests and in the tragic SL-1 accident.*

35. Until one of these factors comes into play, however, the power will continue to rise, the fuel temperature will likewise continue to rise, and substantial release of fission products and energy is possible.

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* In an extreme case, e.g. the atomic bomb, negative reactivity is introduced because of the rapid expansion of the plasma, governed by the equation of state, which self-terminates the chain reaction.

A sense of the core destruction and fuel melting possible from such an excursion can be obtained from the attached photos of the BORAX I final power excursion.

36. The goal of a research reactor designer, particularly one whose reactor might be operated by students, was to design a reactor with a very high degree of inherent safety. A reactor with inherent safety is one in which features involving the very nature of the reactor itself can limit a power excursion without the necessity of appropriate response by the reactor operator or appropriate function of the reactor's engineered safety features. To have a high degree of inherent safety, the reactor needs very large and very prompt negative temperature coefficients, so that when the power rises, and accordingly temperature, the temperature rise automatically shuts the reactor down before damage can occur. Inherent safety features are the very last line of defense in a reactor which can go prompt critical, the only defense in fact, other than administrative controls (which can't be counted on at a training reactor).

37. Degree of inherent protection, i.e. the magnitude and promptness of self-shutdown mechanisms, vary widely, reactor to reactor. In some reactors, these reactivity coefficients are occasionally positive, creating potentially dangerous situations where the reaction feeds on itself rather than providing a measure of self-control.

38. Some reactors' inherent shutdown mechanisms are vastly more effective and reliable than others. The TRIGA reactor, for example, has part of its moderator built right into the fuel; thus there is virtually no time delay in the negative temperature coefficient (which is very large in the TRIGA) taking hold, because there is no delay in transferring the heat to the moderator.

39. Other reactors, such as the Argonaut, have far less prompt and effective self-limiting inherent features. In reactors of the BORAX and SPERT variety, essentially the only inherent mechanism that can limit an excursion is transfer of the rising heat from the fuel meat to the cladding to the water moderator and formation of steam and expulsion of the remaining water. This reduces moderation, increases neutron leakage, and eventually stops the reaction.

40. In such reactors, this last remaining shutdown feature is far slower than that of the TRIGA reactor, in which there is essentially no time delay necessary for heat transfer. That delay, for excursions of short exponential period, will prevent self-shutdown before the reactor power has reached a dangerous level resulting in fuel melting and possible explosion. And it is very difficult to estimate for different reactor designs what the limiting period is, i.e. at how short a period the formation of voids ceases to be effective in preventing fuel damage. The effect depends upon a wide variety of variables (plate thickness and conductivity, surface area-to-moderator volume, coolant channel thickness, size and sign of void and temperature coefficients, and so on).

41. The UCLA Argonaut-type reactor is left with this one inherent shutdown feature, lacking the Doppler effect of low-enriched fuel or the very large, very prompt negative coefficients of the TRIGA. And compared with the BORAX and SPERT reactors, upon which the primary reactivity tests have been performed, UCLA's sole remaining shutdown mechanism (temperature effects in the water coolant/moderator) is far less effective in limiting a power excursion than those characteristics for BORAX and SPERT; The void and temperature coefficients are considerably smaller for UCLA; furthermore, UCLA has several <u>positive</u> coefficients. When water is removed from the UCLA core, the initial effect is positive, since there is more than the optimum water in the core. Only after some of the water has been removed, increasing reactivity, does the coefficient become negative, causing reactivity to drop. Besides delaying shutdown, the positive contribution can make a more severe excursion. Furthermore, the earthquake vibration tests revealed power oscillations caused by the fact that within the coolant channels the reactor is undermoderated and thus any increase in plate spacing amounts to a positive reactivity insertion. (The misleading nature of the reference in the application to the vibration tests is especially serious, not only because it obscures this potential positive reactivity effect but because the information, if not so obscured, makes clear that the assertion elsewhere in the application that the reactor is optimally moderated so that any seismically-induced or other rearrangement of the core or fuel would decrease reactivity is simply not correct.)

42. Finally, the graphite also has a positive coefficient. Unlike the SPERT and BORAX reactors, the UCLA Argonaut has two moderators, water and graphite. Graphite also serves as a reflector. Thus the reduced effectiveness of voids in the water moderator is made clear, because the graphite remains largely unaffected by the temperature rise (i.e., it can't be expelled from the core in the fashion the water can). What effect there is, is positive.

43. Given the relatively ineffective nature of the sole shut-down mechanism in the UCLA Argonaut-- production of voids in one of the reactor's moderators-- the reactor has far less inherent safety than other reactors of the TRIGA or SPERT variety.

44. The original Hazards Analysis for the UCLA reactor, the one that formed the basis for granting the original license and the basis for twenty-two years of operation thereafter, examined in some detail the amount of excess reactivity that should be permitted at that reactor, consistent with student operation and urban siting and lack of containment features. It should be recalled, as pointed out above, that all of the traditional safety features (exclusion zone, containment, radioactivity removal systems for emergency, low population zone, emergency core cooling system, and the like) are lacking at UCLA. There is only one barrier to fission product release-- the fuel cladding. And there is only one method that might be able to limit the consequences of a reactivity insertion greater than $1.00, for which control blades and dump valves would be ineffective, and that is a relatively weak and slow voiding effect about which there are numerous uncertainties as to how large a reactivity insertion can be compensated prior to fuel melting occurring. One thin, low-melting barrier to fission product release; and one uncertain self-shutdown mechanism to prevent penetration of that barrier.

45. The Hazards Analysis wisely concluded that the fission product inventory should be kept low, by limiting operation to 10 kw, so as to reduce consequences if the fission product barrier were breached; and to limit excess reactivity available to less than that necessary for prompt criticality, for which engineered safety features would still be effective. It demonstrated that this was, in its view, a sufficient margin of safety by estimating that the fuel would reach the melting point of aluminum with an excursion of roughly

2.3% $\Delta k/k$, based on rough extrapolations from the BORAX data, corrected for a few of the differences between the reactors (and assuming linear corrections were possible). Given the uncertainties in the calculations (i.e., which meant that melting might occur far below 2.3%), restricting the reactor to less than that necessary for prompt criticality was determined necessary.

46. The Intervenor is correct in pointing out that use of the lower void coefficients, delayed neutron lifetime, and consideration of eutectic melting would substantially lower the estimate of the point at which melting might be expected. More importantly, consideration of error bars at each step in the calculation would reduce considerably further the estimated reactivity insertion that could be tolerated without melting, i.e. that could be successfully terminated by steam formation. The Hazards Analysis makes clear the fuel meat could exceed its melting point (below that of aluminum) with a 2.3% insertion; corrected for void coefficients, delayed neutron lifetime, and error bars, insertions considerably below \$3.00 can be expected to cause melting.

47. The Hawley study attempts to address the same issue, except fewer corrections are made for the difference in characteristics between the UCLA reactor and the SPERT reactor, the original data source. Hawley concludes temperatures about 54° below the melting point of the fuel could be attained; given the NRC Staff assumption of a 75°C starting temperature for the fuel instead of 60°C, for the same energy release (SER p. 4-3 and 14-4), thus producing only a 39°C margin of safety. Given the <u>extremely crude approximations</u> used, the numerous factors not considered (e.g. lower void coefficient) that would markedly push up the estimated energy release and temperature, and the lack of error bars, just a few of these corrections could push the temperature above the melting temperature of the fuel.

48. An example of the non-conservative nature of the Hawley analysis can be demonstrated by comparing his results with those of the other analyses he cites at pages 4-7. Hawley assumes an excursion of period 7.2 msec at UCLA will only release 12 MW-s of energy, getting within a few degrees of melting, from a 2.6% insertion (supposedly the equivalent of 2.3% on a cold day). Yet the 1961 ATL analysis is reported as indicating a far smaller insertion, 1.5%, will produce an energy release of 24 MW-s, double that assumed by Hawley. The GNEC report assumes the same period, around 7 msec, producing 32 MW-s, plus about 4 MW-s to raise the temperature to the saturation point of water

(i.e., about 36 total). UCLA's 1960 Hazards Analysis assumes a smaller insertion, 2.3%, producing the longer period of 9.1 msec, producing 28.4 MW-s plus about 4 MW-s to bring the temperature to saturation. The Jason reactors are referred to as estimating 10 MW-s, nearly that estimated by Hawley for UCLA for a 2.6% insertion, occurring from only 0.5% insertion at Jason. Hawley notes that the variations "are not resolved" in the available documentation. (p.7). With such wide variation, and lack of documentation to explain the variation, it is most non-conservative of Hawley to utilize a 12 MW-s estimate of energy release whereas other estimates several times higher exist, noting that less than 1 MW-s additional energy release would eliminate Hawley's 39° margin of safety (even ignoring the lack of error bars, which would obliterate margins of safety far larger).

49. Hawley's chief error is in equation (2) on page 17, where he assumes the total energy release can be precisely determined by doubling the ratio of the reciprocal period to a reactivity coefficient. He assumes that he can apply, without any modification, the reactivity coefficient found in the SPERT I-D tests (which was substantially different than the reactivity coefficient found in the SPERT I-A tests, the BORAX tests, or from the SL-1 accident) directly to the UCLA case. He commits this error by ignoring the second portion of the power excursion mechanism-- the imposition of shutdown features.

50. As we showed above, power rise in a power excursion is exponential, essentially increasing by a factor of 2.718 every few milliseconds. The amount of energy released is thus a function of essentially two features: the exponential period (the e-folding time) and the length of the excursion before shutdown (i.e., the number of e-folding periods). From the equation given in paragraph 27, we see immediately that very small changes in either t (the time that elapses before shutdown mechanisms take hold and terminate the power rise) or T (the exponential period, or e-folding time; the time it takes the power to increase by 2.718) can have very large effects on the power reached. Hawley essentially ignores the fact that any linear delay in the shutdown mechanism can cause a <u>nonlinear</u> (i.e., exponential) increase in the total power.

51. Because of the reported longer neutron lifetime at UCLA, the same reactivity insertion will produce a longer exponential period T than it would at SPERT. Hawley takes into account this difference between UCLA and SPERT (which helps UCLA), but ignores the differences between the reactors which will mean a longer total excursion because of slower or smaller shutdown mechanisms. Thus, T may be longer, which Hawley considers, but t may also be longer, which he does not. Since the power rise is exponential, ignoring even a few millisecond delay in shutdown mechanism can be devastating.

52. Assume an exponential period T of 7 msec and a time interval of rising power before the shutdown mechanism acts of .07 sec (e.g., the time it takes the heat to transfer from fuel to clad to coolant and cause voiding of the moderator). The power would thus rise by $e^{.07/.007}$, or e^{10}, a very large number (about a 22,000-fold rise in power). If initial power was 100 kw, seven hundredths of a second later the power would be over 2000 MW. If the shutdown mechanism at UCLA is even a few percent slower or less effective than that of SPERT, (e.g., because of the 50% smaller void coefficient, the thicker plate dimensions, a little bit of added corrosion on the clad, or the positive effect of the initial coolant drop or the graphite temperature coefficient), the difference in peak power can be very substantial.

53. Taking the example given above, and assuming a very modest difference of 10% in speed of shutdown, representing a few milliseconds, one additional e-folding period would occur at UCLA before shutdown than at SPERT, from which Hawley obtained his 12 MW-s estimate. This could mean, thus, peak power 2.7 times higher, just because of a delay of a few thousandths of a second in transferring heat to the coolant, voiding the coolant, or the reactivity worth of voiding the coolant. In other words, a few percent less prompt or less effective shutdown mechanism does not mean a few percent higher peak power, but because of the exponential nature of the rise, would mean several times higher peak power.

54. All indications are that the shutdown mechanisms for UCLA could be substantially slower and smaller in effect than those of the SPERT or BORAX reactors with which they are being compared. The 1960 Hazards Analysis made clear that just correcting for a few of the differences between UCLA and BORAX, the minimum period UCLA was expected to be able to tolerate was considerably longer than that estimated for BORAX. The void coefficient is smaller, which is quite important, and simple effects like the 50-100% increase in thermal resistivity in the Al-U fuel meat caused by irradiation (p. 192, Reactor Handbook, 2nd edition, Volume I, Materials, 1960, edited by Tipton) can substantially elongate the time interval for the heat generated in the excursion to be transferred to the moderator for eventual shutdown. Given the exponential nature of the rise, and the exponential period measured in milliseconds, delays of a millisecond or two in transferring the heat, and differences of a few percent in the effectiveness of the voids once formed in the coolant, mean melting can occur substantially below the reactivity insertions assumed by Hawley or the original Hazards Analysis. My professional opinion is that, based on the analyses done to date, insertion of either \$3.00 or \$3.54 must be considered a credible cause of fuel melting.

Chemical Reactions

55. It has been asserted that the only chemical reaction of significance to be considered for the UCLA reactor is a reaction with aluminum, and that aluminum would have to be in the form of metal filings for such a reaction to occur. That is not the case.

56. Each of the destructive power excursions with aluminum-clad, plate type fuel (SPERT, BORAX, and SL-1) has apparently resulted in significant metal-water reaction. Much of the core disassembly in those three cases can be traced to a combination of steam explosion and metal-water reaction. The aluminum was in the form of fuel cladding; most assertedly, not in the form of metal filings.

57. Other chemical reactions will be discussed in the section on reactor fire, below. Suffice it to say at this point that a destructive power excursion could not only result in fuel melting, but explosive disassembly of the core due to steam explosion and metal-water reaction.

Fire

58. The original UCLA Hazards Analysis was apparently copied virtually verbatim from materials provided by AMF and by the University of Florida, and makes a number of very serious errors. This is one of the major dangers in copying analyses from others, rather than doing the analysis yourself, particularly when the copied analysis is from a vendor who has an interest in minimizing potential hazards. One would think a sophisticated university would have the wherewithal to have perform its own analyses, and avoided some of the errors made by the authors of the analyses that were copied.

59. Perhaps the most egregious error is the assertion on page 62 of the Hazards Analysis that there is no possibility of fire because none of the materials of reactor construction is flammable. That is dangerously untrue.

60. The graphite will burn; the uranium metal will burn; the magnesium will burn; even the aluminum under some circumstances will burn (as we have seen in the Falklands). The graphite, uranium and magnesium all have relatively low ignition points (i.e., temperatures that could quite credibly occur at some point in the reactor's operating lifetime, through accident, equipment malfunction, building fire, or the like).

61. A new assertion of the University is that there is not sufficient air present to sustain combustion for an extended period of time. It cites an air flow rate during normal operation for an Argon extract line as evidence. In so doing, the University completely misses the point. Even were the University correct in its estimate of flow rate during normal conditions, in which an extract line fan produces forced circulation, such a measurement is irrelevant with regards the air flow possible in case of a fire. The fire would produce convection currents, drawing air in and exhausting depleted air. After all, the measured air flow rate from a gas oven or heater, when off, will be close to zero; it is the fire which produces air flow through convection.

62. The reactor is simply a pile of combustible materials, with additional airslots for control blades, irradiation ports, and the like. The Windscale fire demonstrated clearly that such piles can burn.

63. And as the Windscale fire demonstrated, graphite reactors which operate at low temperatures can store substantial Wigner energy, which can cause or exacerbate severely a range of accidents. Even were a power excursion to occur which could only raise the temperature of the fuel to something below the melting or ignition temperature, the eventual transfer of some of that heat to the graphite could trigger the release of the stored energy therein, and push the total temperatures "over the top." The same is true for other accident sequences, such as Cort's hypothetical core-crushing incident, which would be sufficient to raise the graphite temperature to the Wigner release point, adding enough energy to the incident to bring about melting or ignition.

64. Lastly, fire suppression response, particularly if ill-prepared as in the UCLA case, can vastly worsen the situation. Metal-water reactions between the aluminum, uranium, and magnesium can be explosive and liberate considerable energy if water were poured on those substances when on fire. Likewise with burning graphite. And addition of water to the reactor could cause a power excursion. Hawley indicates (p.25-27) flooding the core air spaces could produce as much as $6\% \Delta k/k$ insertion; even a third of that amount could cause a devastating power excursion.

Seismic Disruption

65. The site is in a highly seismically active region, near major faults, and none of the special construction features usually associated with reactor construction in seismically active areas were apparently included. Thus, the Staff SER and the Cort study rightly presume an earthquake could collapse the building above and crush the reactor core. The question is how much damage would be done to the fuel.

66. The Hawley study assumes a fuel handling accident to cause exposure of 10,500 cm^2 of surface area of fuel meat, if the accident involved only one of the reactor's 24 fuel elements. (p. 47). It states additionally (p. 26) that the consequences of a core-crushing accident "would be some multiple of the consequence of the fuel-handling accident." The damage to a single bundle in a severe core-crushing accident induced by collapse of the building above onto the core in a severe earthquake would be substantially greater than the damage induced in a fuel-handling accident to a single bundle. Furthermore, one must presume most, if not all of the fuel bundles in a core-crushing accident would be similarly affected. At minimum, then, the consequences would be twenty times as great for a core-crushing incident as the assumed fuel-handling incident. (I note that the Staff attempts to make some comparison to guillitone-type breaks in the fuel. The fuel is unlikely to shatter in clean, guillitone-type cuts. The jagged exposed surfaces will have substantially more surface area exposed; the fission product release rate from jagged surfaces as opposed to clean cuts would not be substantially different. The Hawley estimate of likely exposed area in a fuel-handling accident is not too high; it is quite unrealistic to assert that the damage that a severe core-crushing accident could produce would be the same or less than the damage that could result from a fuel-handling accident to a single bundle.).

67. I note also that Hawley asserts that only gaseous fission products one recoil-distance from the exposed surface could be released if the cladding were broken or breached. This assertion is inadequately demonstrated. The concept of recoil-distance for fission fragments originated in determining whether the kinetic energy of fission fragments, primarily while fissioning was occurring, would be enough to penetrate relatively thin cladding. The concept of recoil-distance was used to indicate that if the cladding were substantially thicker than the recoil-distance, fission fragments during fissioning would not penetrate intact cladding. The step between that finding and Hawley's assertion that gaseous fission products more than one recoil-distance from the surface of unclad irradiated fuel meat would not be released is not clearly shown.

68. Earthquake-induced structural damage is often accompanied by fire. In this case, the structural damage could essentially expose the core interior to more air than might be available were the core intact, making propagation of fire even easier.

69. The history of tie-bolt failures for the fuel and the unreported finding from the vibration tests of reactivity oscillations due to the severely undermoderated current configuration, particularly with regards coolant channels that are half the optimum width, creates potential for reactivity surges due to bowing or other plate spacing changes induced by the seismic shock. An earthquake could also readily cause a large negative worth sample to be removed from the core region rapidly, without time for intervention of the control blades to compensate, resulting in a power excursion. (One particularly worrisome scenario would be a large negative worth sample in an irradiation port, the sample being in liquid form in a container which is squeezed or shattered by the compressive forces in the earthquake, rapidly expelling the contents from the core region. In addition to effecting a positive reactivity insertion, if the liquid were a solvent, the reactivity-induced temperature rise could ignite the material).

70. The history of control blade difficulties also presents problems. The vibration tests indicate that seismically-induced core-shifting can pin the drive mechanisms. The results were delayed from the vibration tests, but those tests simulated very small accelerations compared to what can be expected in a realistic earthquake. Pinning of control blades, jamming of the dump valve can make reactor shutdown impossible, perhaps with operation at a power level far in excess of the design level which can result in enough decay heat, once the water is boiled off, to result in fuel melting. (Note that Cort concluded that melting could occur simply from seismically-induced blockage of cooling for a 500 kw Argonaut).

Cladding Damage

71. As indicated at the outset, the primary barrier to fission product release at the UCLA reactor is .015 inch thick aluminum cladding. Severe corrosion of that cladding could produce substantial fission product release, including release of soluble, non-gaseous fission products.

72. Because of the low utilization of the reactor, fuel originally installed in the core in 1960 could remain there until the end of the proposed license period, the year 2000, due to the small burnup rate. Forty years, much of which is spent in water, could produce very substantial corrosion of the thin cladding. Failure to properly calibrate or maintain the resistivity monitor for the primary coolant, and an inadequate secondary coolant monitor, could prevent discovery of substantial release until after it had occurred. Post-accident monitoring would be of little usefulness in terms of preventing the release to begin with.

73. The University now claims that the primary coolant leak that developed after the 1971 earthquake and required major maintenance and a long shutdown were not due, as originally stated, to the earthquake but rather to corrosion of the aluminum primary coolant piping. If the far thicker aluminum piping was so substantially corroded in ten years of operation by exposure to the primary coolant, then the far thinner aluminum fuel cladding could well be at substantial risk over the forty year period being considered.

74. I note also that numerous materials attack aluminum. The primary coolant make-up system is based on the third floor above the reactor. Accidental insertion of chemicals detrimental to aluminum, or experimental addition of some such material, or an attempt to remove material clogging parts of the coolant piping through addition of a flushing compound, or deliberate sabotage could all result in severe and rapid degradation of the cladding and release of fission products. Some materials react explosively with aluminum, which could be even more devastating.

Sabotage

75. For obvious reasons, this section will be rather sketchy. Under more restricted conditions, more details could be provided.

76. In addition to the problem of intentional insertion of materials into primary coolant make-up water that could attack the cladding rapidly, insertion of explosive or incendiary devices within the reactor core (through insertion in one of the several irradiation ports which penetrate to the center of the core), could produce results far greater than those of most accident sequences. An explosion would fragment the fuel to a far greater extent than would an earthquake; the likelihood of fire amidst the debris, particularly if incendiary devices were utilized, would vastly increase the fission product release.

Criticality Accident

77. In addition to in-reactor reactivity accidents, possession of the requested amounts of HEU pose potentials for out-of-reactor criticality accidents. Particularly in connection with a facility that additionally has two subcritical facilities, with fuel (in addition to the HEU in the reactor and in storage) and a large quantity (many barrels) of heavy water and graphite for experimental uses. Experiments have already been performed at the facility with fuel bundles in water pools outside the reactor; extensive experiments were performed changing spacing for fuel bundles and determining reactivity effects. More creative experiments can be assumed as the basic reactor physics experiments that can be done with such a reactor have been exhausted. With much-criticized administrative controls, experiments by students without proper prior review or supervision could be very dangerous. For these reasons, strict and detailed procedures to prevent criticality accidents are important. Training in criticality accident prevention should be included in the basic training; this should include extensive review of the history of criticality accidents, a list that would not be nearly so long had other facilities taken the risk more seriously and taken appropriate precautions.

Miscellaneous Points

78. Mr. Ostrander, in his September 1, 1982, declaration, at page 10, asserts that the reason why BORAX data suggest a so much larger power excursion for the same period than does SPERT (and why he believes it appropriate to ignore the more conservative BORAX data) is because of different active core height producing hydrostatic pressure and inertia forces which impede boiling more in the BORAX case. This is an interesting hypothesis; unfortunately, the validity of it is not demonstrated by Mr. Ostrander in his declaration, which primarily speculates on the possible effect.

79. However, assuming for the moment that Mr. Ostrander is correct, such an effect may well be very unfavorable for UCLA. Because among the many differences between SPERT/BORAX and UCLA, a clear one is that the former were open tank reactors at atmospheric pressure. There was nothing to impede the expulsion of the moderator out of the core. In the UCLA case, the moderator is in a closed system; in order for the coolant to be expelled, a pressure pulse must be generated in the core region, transmitted through the coolant through relatively narrow piping and several junctions to a rupture disk, where sufficient pressure must be generated to cause it to break, and the coolant then to drain out. All of this can take considerable time. Under normal conditions, it takes approximately 20 seconds for 20% of the core water to drain out of the dump valve; under pressure it would be faster, but the central question is whether this rather complicated sequence of events for water to be removed from the core would result in a delay over the SPERT/BORAX shutdown mechanism of simple expulsion out of the reactor tank open top. As indicated earlier, a delay of even milliseconds can mean substantially higher power released.

80. It has also been asserted that the graphite positive coefficient is slow-acting. As Mr. Ostrander correctly points out (p. 12), there is a prompt component to the graphite heating effect, due to the prompt gamma and neutron radiation. In fact, even the normal graphite heating effect is largely due to these prompt effects, because the graphite temperature rises soon after reactor operation begins to a temperature above that of the coolant, and continues to rise steadily as operation continues,

thus demonstrating that the heat transferrence mechanism even during normal operation is primarily prompt radiation, not slower conduction. Mr. Ostrander estimates a $2^{o}F$ rise in the graphite during a 12 MW-s excursion, which he terms insubstantial. As indicated above, such an energy release estimate is quite low; a more realistic estimate would be 3-4 times higher. An $8^{o}F$ graphite temperature rise, still seemingly quite small, would result in a positive reactivity insertion of about .05% $\Delta k/k$, given the reported coefficient of + .006% $\Delta k/k$ / ^{o}F. At UCLA's current level of requested reactivity ($3.00, or 1.95%), that would mean the actual insertion would be about 2%, a difference enough to wipe out a substantial portion of the $39\text{-}54^{o}$ temperature margin of safety assumed by Hawley.

81. The more significant aspect of the graphite coefficient is that it means that the <u>available</u> excess reactivity can <u>increase</u> by nearly $1 from normal simply by a temperature rise of $100^{o}F$ (normal for a 2 hour run, as reported by UCLA), assuming the .006% coefficient, and remain elevated after shutdown because of the long cool-down period for the graphite.

82. I understand there is some question about whether large negative worth samples can be inserted in the reactor core. They clearly can be in the irradiation ports, as was indicated to me during my site visit. The pneumatic tube system, likewise, is capable of carrying some highly absorbing materials. In addition, large positive worth samples could be inserted in the core. If such insertion were not possible, UCLA would have had no need to increase its excess reactivity level from .6% to 2.3%; if such insertion is impossible or not anticipated, then there should be no reason not to reduce the level back to the design value.

<u>Accident Consequences</u>

83. I have made some preliminary estimates of fission product release fraction possible for the various accident scenarios discussed above. Power excursion could result in 20% release of the gaseous material and substantial release (though much less than 20%) of the particulate material entrained in the steam. Seismically-induced mechanical damage to the fuel would result in at least 24 times the release assumed by Hawley for fuel-handling accidents. Sabotage involving explosion and incendiary-induced fire could release 80% of the full inventory. Fire could release close to 100% of the gaseous material (those

species volatile at the temperatures attained in the fire) and roughly 40% of the particulate matter, dispersed by the driving force of the fire. These are rough estimates. The Applicant should be required to analyze the above-described accidents and comprehensively estimate release fraction.

84. I would say, however, that the maximum credible accident at the UCLA reactor could result in at least release of 25% of the radioiodine. The total release could be far greater. In particular, several accident sequences at this facility have a potential greater than for most reactors to drive out with a substantial driving force particulate matter that is normally assumed to remain within the reactor or reactor structure.

85. I would add that several of the accident sequences could readily involve contamination of ground water supplies, if there were water wells used for drinking purposes in the vicinity. Many of the materials are water soluble; driven into the environment they would eventually reach ground water.

Conclusion

86. There are numerous credible accident scenarios involving release of a substantial fraction of the UCLA reactor core inventory. These result in consequences far in excess of those postulated by UCLA for a fuel handling accident. The reliance on a sole fission product barrier-- the fuel itself-- and the high population density with no exclusion zone make the accident consequences unacceptably high.

I declare under penalty of perjury that the foregoing is true and correct to the best of my knowledge and belief.

Michio Kaku, Ph.D.

Dated at New York City, NY, this 1 day of Jan, 1983

Statement of Professional Qualifications

PROFESSOR MICHIO KAKU

My name is Michio Kaku. I am Associate Professor of Theoretical Physics at the City University of New York (City College campus).

I did my undergraduate work in physics at Harvard University, from which I graduated in 1968 summa cum laude and phi beta kappa.

I did my graduate work in nuclear physics at the Lawrence Radiation Laboratory of the University of California at Berkeley, from which I received my Ph.D. in 1972.

From 1972-73 I was on the faculty of Princeton University as a lecturer.

From 1973 through the present I have been on the faculty at CCNY.

I have published approximately thirty-five research papers in theoretical physics in such professional journals as Nuclear Physics, Physical Review, and Physics Letters, and have contributed to five books on nuclear and theoretical physics.

My special interests include reactivity calculations for breeder and other reactors, neutron transport theory in reactor physics, the history of accidents at non-power reactors, and metal-water reactions in reactor accident sequences, as well as aspects of theoretical physics such as hadron-hadron interactions, general relativity, supergravity, and unified field theories of the strong and weak interactions.

I am co-editor of the book Nuclear Power: Both Sides, published by Norton in 1982.

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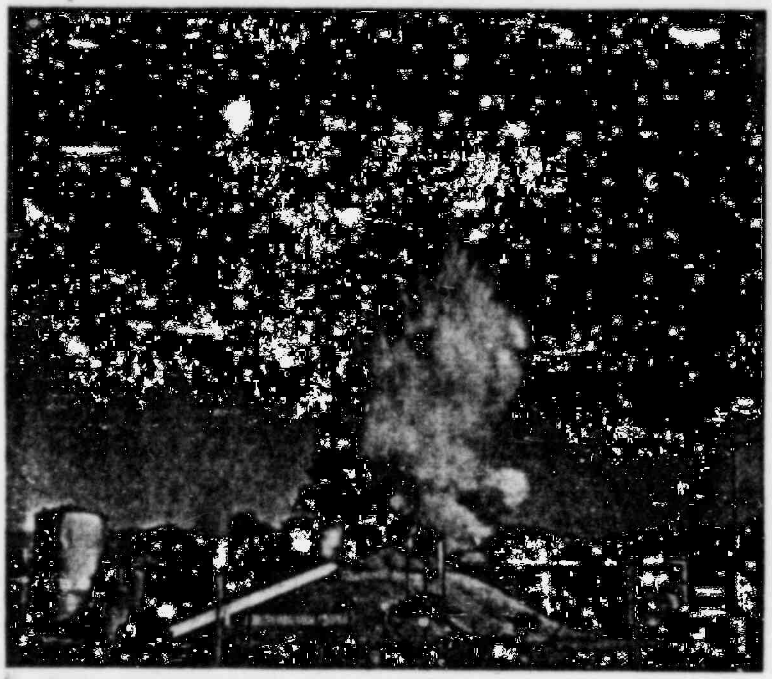
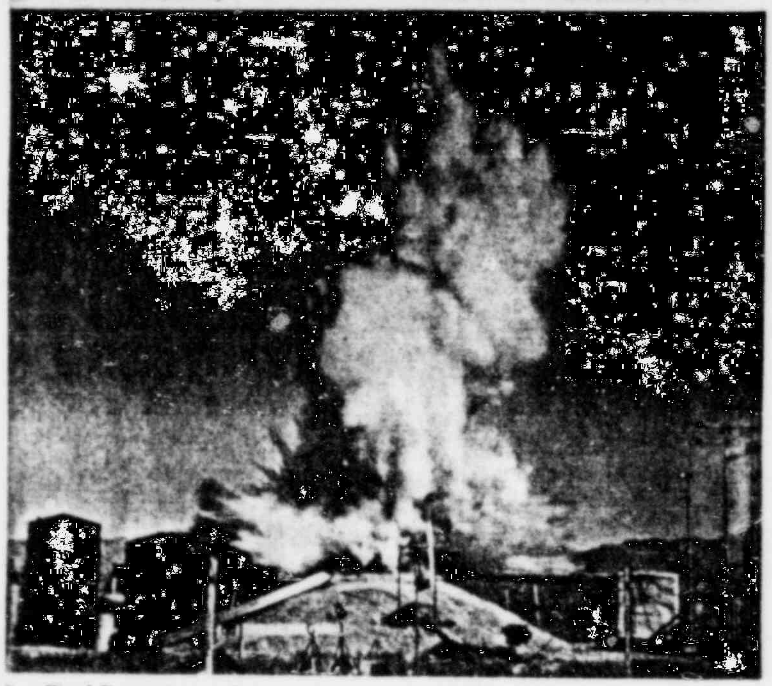
Volume 1

REACTOR SAFEGUARDS



7. Final Borax I experiment. (Argonne National Laboratory photograph.)



8. Final Borax I experiment. (Argonne National Laboratory photograph.)

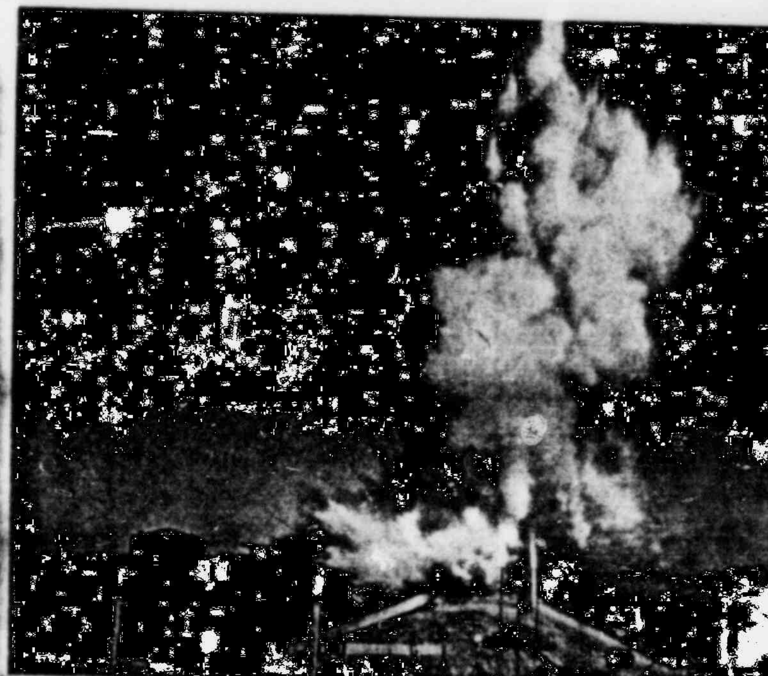


FIG. 9. Final Borax I experiment. (Argonne National Laboratory photogr

FIG. 10. Final Borax I experiment. (Argonne National Laboratory photogr

Fig. 14. Remains of fuel element from final Borax I experiment. (Argonne National Laboratory photograph.)

Fig. 15. Remains of fuel plate from final Borax I experiment (Argonne National Laboratory photograph.)

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IEF237I  826    ALLOCATED TO FT07F004
IEF237I  826    ALLOCATED TO FT07F005
IEF237I  826    ALLOCATED TO FT07F006
```

Core

20 yr at 2.5 Kw + 8 hr at 100 kw

Discharge - 1 min - 1 hr - 1 Day - 1 wk

UCLA RI REACTOR 20 YEAR FUEL CYCLE 93 % ENRICHED URANIUM

POWER= 0.00MW, BURNUP= 18.MWD, FLUX= 2.10E+10N/CM**2-SEC

NUCLIDE RADIOACTIVITY, CURIES
BASIS = TOTAL HEAVY METAL INVENTORY OF CORE

	CHARGE	DISCHARGE	0. D	0. D	1. D	7. D	30. D
SM153	0.0	1.92E+01	1.92E+01	1.92E+01	1.37E+01	1.63E+00	4.76E-04
EU153	0.0	0.0	0.0	0.0	0.0	0.0	0.0
GD153	0.0	7.30E-06	7.30E-06	7.30E-06	7.28E-06	7.16E-06	6.70E-06
PM154	0.0	6.49E+01	4.83E+01	3.39E-06	0.0	0.0	0.0
SM154	0.0	0.0	0.0	0.0	0.0	0.0	0.0
EU154	0.0	9.89E-03	9.89E-03	9.89E-03	9.89E-03	9.89E-03	9.86E-03
GD154	0.0	0.0	0.0	0.0	0.0	0.0	0.0
SM155	0.0	2.78E+01	2.70E+01	4.50E+00	3.97E-18	0.0	0.0
EU155	0.0	6.92E-01	6.92E-01	6.93E-01	6.92E-01	6.88E-01	6.72E-01
GD155	0.0	0.0	0.0	0.0	0.0	0.0	0.0
SM156	0.0	5.41E+00	5.40E+00	5.02E+00	9.22E-01	2.25E-05	4.74E-23
EU156	0.0	3.57E-01	3.57E-01	3.66E-01	4.55E-01	3.63E-01	1.25E-01
GD156	0.0	0.0	0.0	0.0	0.0	0.0	0.0
SM157	0.0	6.53E+00	1.50E+00	0.0	0.0	0.0	0.0
EU157	0.0	2.12E+00	2.12E+00	2.02E+00	7.09E-01	1.01E-03	1.18E-14
GD157	0.0	0.0	0.0	0.0	0.0	0.0	0.0
EU158	0.0	1.69E+00	1.66E+00	6.79E-01	6.37E-10	0.0	0.0
GD158	0.0	0.0	0.0	0.0	0.0	0.0	0.0
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TB159	0.0	0.0	0.0	0.0	0.0	0.0	0.0
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DY160	0.0	0.0	0.0	0.0	0.0	0.0	0.0
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DY161	0.0	0.0	0.0	0.0	0.0	0.0	0.0
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TB163M	0.0	1.52E-02	1.37E-02	3.81E-05	0.0	0.0	0.0
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HO165	0.0	0.0	0.0	0.0	0.0	0.0	0.0
DY166	0.0	9.39E-08	9.39E-08	9.31E-08	7.66E-08	2.25E-08	2.06E-10
HO166M	0.0	3.94E-13	3.94E-13	3.94E-13	3.94E-13	3.94E-13	3.94E-13
HO166	0.0	8.84E-08	8.84E-08	8.86E-08	8.64E-08	3.29E-08	3.07E-10
ER166	0.0	0.0	0.0	0.0	0.0	0.0	0.0
ER167	0.0	0.0	0.0	0.0	0.0	0.0	0.0
TOTAL	0.0	3.44E+05	2.29E+05	6.73E+04	9.43E+03	2.49E+03	1.27E+03