PANEL i

6/14/83

INTRODUCTION

CONCEPTS OF INHERENT SAFETY AS APPLIED TO THE UCLA ARGONAUT REACTOR

1. The review reported in the testimony which follows focuses on the question: Is the UCLA nuclear reactor inherently safe?

2. A detailed examination of whether the Argonaut-type reactor on the UCLA campus meets that standard is best begun by a brief discussion of what the basic concepts of inherent safety are.*

3. Nero (A Guidebook to Nuclear Reactors, UC Press, 1979) points out that reactor safety features may be divided into two categories: intrinsic safety features, "those that are inherent in the physical nature of the reactor concept being considered," and engineered safety features, "systems that are added to the basic reactor concept." Nero gives the following examples for the two kinds of safety features:

> An example of intrinsic mechanisms is that, in a light-water reactor, overheating of the coolant caused by an abnormal rise in the reaction rate tends to reduce the water density and thereby shut down the chain reaction due to insufficient moderation of neutrons. Alternatively, the emergency shutdown-control-rod system may be regarded as an engineered safety feature. (p. 13)

Thus, engineered safety relates to devices which, if they operate correctly, can prevent hazard; inherent safety relates to intrinsic design features of the reactor which operate automatically, due to the laws of physics, rather than relying on correct response by devices or human operators.

4. The history and underlying philosophy of the development of research and training reactors was an attempt to create reactors that were indeed inherently or intrinsically safe, i.e.

* We understand that 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. The brevity of this explanation necessitates considerable simplification of rather complex concepts.

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that did not need to rely on proper operation of engineered safety features or proper maintenance and operation by human beings. This was because research reactors, unlike power reactors, were to be used for training students, who must be expected to learn by making mistakes. A safe training reactor, thus, must be of an intrinsicly "forgiving" design, so that the worst possible errors could not possibly cause harm, either to the student or to the public.

5. A clear indication of the roots of this philosophy of the necessity of inherent safety for research reactors can be found by reviewing the origins of the development of the TRIGA reactor, probably both the most successful reactor design and the one that most closely meets the inherent safety standard. The TRIGA was designed in 1956 by Edward Teller, Theodore Taylor, Freeman Dyson and others. As Dyson has described it (Disturbing the Universe, Harper & Row, 1979),* their task was to "design a reactor so safe that it could be given to a bunch of high school students to play with, without any fear that they would get hurt."

6. Meeting that standard is difficult because of the unique nature of nuclear reactors. As will be described in detail during the panel on power excursions, nuclear reactors are different than virtually any other machine in that they can undergo astronomical rises in power in extraordinarily short periods of time, from zero to billions of watts in a very small portion of a second. Such an event can lead to melting of the fuel and explosive disassembly of the reactor core. Preventing such occurrences is thus the goal of the designers of reactors. As Dyson puts it:

> The first rule in operating a reactor is that you do not suddenly yank the control rods out of a shut-down reactor. The result of suddenly pulling out the control rods would in most cases be a catastrophic accident, including as one of its minor consequences the death of the idiot who pulled the rods.

The TRIGA design team knew that, for training reactors to be put on university campuses for student instructional purposes, the reactor must be "idiot-proof." Other kinds of reactors could rely on engineered safety, which Dyson defines as meaning that a catastrophic accident is "theoretically possible but is prevented by the way the control system is designed."

7. For the designers of the TRIGA, Dyson says, "engineered safety was not good enough." The reactor must be designed with inherent safety,

> meaning that its safety must be guaranteed by the laws of nature and not merely by the details of its engineering.

* excerpts attached as Exhibit C-i-1

> It must be safe even in the hands of an idiot clever enough to by-pass the entire control system and blow out the control rods with dynamite.

Dyson said that stated more precisely, the ground rule for a safe research reactor was "that if it was started from its shutdown condition and all its control rods instantaneously removed, it would settle down to a steady level of operation without melting any of its fuel." This was the standard for measuring the safety of research reactors with regards one of their principal hazards--potential for power excursion. The same standard--protection by the laws of physics, without need for proper operation of devices or people, either of which can fail--was applied to other hazards (e.g. fire) as well. The inherent design was to be failsafe and foolproof.

THE ARGONAUT REACTOR

8. The original Argonaut, though not nearly as inherently safe as the TRIGA, was far more safe than the Argonaut-type reactors that were a few years later sold commercially by AMF and American Radiator and Standard Sanitary Corporation, before they went out of the reactor business. And as shall be described in detail later, the modifications to the UCLA Argonaut-type reactor over the years have significantly reduced safety margins further.

9. The original Argonaut did not have the inherent safety feature that is the primary characteristic of the TRIGA--exceedingly prompt, almost instantaneous self-shutdown features, due to part of the moderator being built right into the fuel. But it used low enriched oxide fuel, had an exclusion area around it, was physically restricted to very low power operation (normal operating power was 100 watts, one thousandth that of UCLA's; because of the inherent negative temperature coefficient, sustained operation above about 1 kw would result in shutdown), had a series of shutdown and interlock systems not found in the commercial model, as well as other safety features. The low power restriction and exclusion zone features alone serve to reduce the consequences of an accident at the original Argonaut many orders of magnitude below the consequences of an accident at a contemporary Argonaut-type reactor. These matters will be discussed in more detail in other panels.

10. AMF's Argonaut-type reactor--the "Educator"--reduced the safety margins but still maintained some important ones, particularly the excess reactivity restriction (to be explained later) of less than that necessary for prompt criticality. The reactor's maximum design power was 10 kw, and the requirement for an exclusion zone was removed as were other of the safety features.

11. A few years after the UCLA reactor was built, safety margins were further reduced. Power (and thus fission product inventory available for release in an accident) were increased ten-fold, and excess reactivity limits four-fold, to precisely the level the Hazards Analysis for the reactor calculated would render what inherent shutdown mechanisms the reactor had ineffective in preventing fuel melting. Other modifications, described in other panels, occurred as well, further reducing safety margins.

12. Before the UCLA reactor lost these safety margins, the University's Regional Advisory Committee On Radiological Safety a comparison of the Argonaut and TRIGA reactors in terms of safety. Their conclusion: "[in]almost every respect, the safety margin seemed to favor the TRIGA reactor, in spite of its higher possible operating level." Exhibit C-i-2. It should be remembered that <u>after</u> this assessment, the Argonaut power went up by a factor of ten and the excess reactivity level far over the prompt critical threshhold that had previously been its primary inherent safety feature. (see also Exhibit C-i-3).

<u>BRIEF EXPLANATION OF RELATED TERMS AND CONCEPTS</u>

13. 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 <u>multiple, redundant barriers</u> 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.

14. In addition to multiple barriers to fission product release, most reactors have substantial <u>exclusion zones</u> and <u>low-population zones</u> 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.

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

16. 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.

17. The UCLA fuel meat and cladding (as well as the control blades) are among the lowest-melting such materials used in reactors.

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.

18. 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.

19. 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.

20. 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.

21. 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.

22. 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.

23. 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).

24. 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.)

25. 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 inherent self-limiting features preclude release of more than a small fraction of a percent of the core inventory. The answer, as shall be demonstrated, is in the negative.

26. 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.)

27. The history of non-power reactor accidents involving severe fuel damage is far more extensive that 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 (as can be seen from the films of their final destructive excursions) 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 inbusiness 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.

28. The panels that follow will address how such changes have affected the potential for serious accident. The categories of accidents examined include: severe power excursion, fire, 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 resulting in both core disruption and fire).

INTRODUCTION

Exhibit List

Exhibit Number	Description
C-1-1	Freeman Dyson, Disturbing the Universe, excerpts
C-1-2	Report and Minutes of the Regional Advisory Committee on Radiological Safety, UCLA, 5/12/60
C-1-3	"Advantages of TRIGA Fuel for Research Reactors", General Atomics, October 1981
C-1-4	Fission product computer estimate

Exhibit C-i-1 page 1 of 3

FREEMAN DYSON

DISTURBING THE UNIVERSE

HARPER & ROW, PUBLISHERS

NEW YORK
Cambridge London
Hagerstown Mexico City
Philadelphia São Paulo
San Francisco Sydney
1817

PUBLISHED AS PART
UNDATION PROGRAM

staff. Freddy rented a little red schoolhouse that had been abandoned as obsolete by the San Diego public school system. He proposed to move into the schoolhouse and begin designing reactors there in June.

Freddy had been at Los Alamos with Edward Teller in 1951 and had made some of the crucial calculations leading to the invention of the hydrogen bomb. He invited Teller to join him in the schoolhouse for the summer of 1956. Teller accepted with enthusiasm. He knew that he and Freddy could work well together, and he shared Freddy's strong desire to get away from bombs for a while and do something constructive with nuclear energy.

Freddy also invited thirty or forty other people to spend the summer in the schoolhouse, most of them people who had been involved with nuclear energy in one way or another, as physicists, chemists or engineers. Robert Charpie, even younger than Freddy, had been the other American in the group of scientific secretaries of the Geneva meeting. Ted Taylor came directly from Los Alamos, where he had been the pioneer of a new art form, the design of small efficient bombs that could be squeezed into tight spaces. For some reason, although I had never had anything to do with nuclear energy and was not even an American citizen, I was also on Freddy's list. Probably this was a result of my encounter with Teller the previous summer. Freddy promised me a chance to work with Teller. I accepted the invitation gladly. I had no idea whether I would be successful as a reactor designer, but at least I would give it a try. For nineteen years I had been waiting for this opportunity to make Eddington's dream come true.

Freddy de Hoffmann was my first encounter with the world of Big Business. I had never before met anybody with the authority to make decisions so quickly and with so little fuss. I found it remarkable that this authority was given to somebody so young. Freddy handled his power lightly. He was good-humored, and willing to listen and learn. He always seemed to have time to spare.

We assembled in June in the schoolhouse, and Freddy told us his plan of work. Every morning there would be three hours of lectures. The people who were already expert in some area of reactor technology would lecture and the others would learn. So at the end of the summer we would all be experts. Meanwhile we would spend the afternoons divided into working groups to invent new kinds of reactors. Our primary job was to find out whether there was any specific type of reactor that looked promising as a commercial venture for General Atomic to build and sell.

The lectures were excellent. They were especially good for me, coming into the reactor business from a position of total ignorance. But even the established experts learned a lot from each other. The physicists who knew everything that was to be known about the physics of reactors learned about the details of the chemistry and engineering. The chemists and engineers learned about the physics. Within a few weeks we were all able to understand each other's problems.

The afternoon sessions quickly crystallized into three working groups, with the titles "Safe Reactor," "Test Reactor" and "Ship Reactor." These were considered to be the three main areas where an immediate market for civilian reactors might exist. In retrospect it seems strange that electricity-producing power reactors were not on our list. Freddy knew that General Atomic must ultimately get into the power reactor business, but he wanted the company to begin with something smaller and simpler to gain experience. The ship reactor was intended to be a nuclear engine for a merchant ship, and the test reactor was intended to be a small reactor with a very high neutron flux which could be used for the testing of component parts of power reactors. Both these reactors would be competing directly with existing reactors that had already been developed for the Navy and the Atomic Energy Commission. Both of them were designed during the summer and then abandoned when Freddy concluded that they had no commercial future. The safe reactor was the only product of our little red schoolhouse which actually got built.

The safe reactor was Teller's idea, and he took charge of it from the beginning. He saw clearly that the problem of safety would be decisive for the long-range future of civilian reactors. If reactors were unsafe, nobody in the long run would want to use them. He told Freddy that the best way for General Atomic to break quickly into the reactor market was to build a reactor that was demonstrably safer than anybody else's. He defined the task of the safe reactor group in the following way: The group was to design a reactor so safe that it could be given to a bunch of high school children to play with, without any fear that they would get hurt. This objective seemed to me to make a great deal of sense. I joined the safe reactor group and

Page 2 of 3

spent the next two months with Teller fighting our way through to a satisfactory solution of his problem.

Working with Teller was as exciting as I had imagined it would be. Almost every day he came to the schoolhouse with some harebrained new idea. Some of his ideas were brilliant, some were practical, and a few were brilliant and practical. I used his ideas as starting points for a more systematic analysis of the problem. His intuition and my mathematics fitted together in the design of the safe reactor just as Dick Feynman's intuition and my mathematics had fitted together in the understanding of the electron. I fought with Teller as I had fought with Feynman, demolishing his wilder schemes and squeezing his intuitions down into equations. Out of our fierce disagreements the shape of the safe reactor gradually emerged. Of course I was not alone with Teller as I had been with Feynman. The safe reactor group was a team of ten people. Teller and I did most of the shouting, while the chemists and engineers in the group did most of the real work.

Reactors are controlled by long metal rods containing substances such as boron and cadmium, which absorb neutrons strongly. When you want to make the reactor run faster, you pull the control rods a little way out of the reactor core. When you want to shut the reactor down, you push the control rods all the way in. The first rule in operating a reactor is that you do not suddenly yank the control rods out of a shut-down reactor. The result of suddenly pulling out the control rods would in most cases be a catastrophic accident, including as one of its minor consequences the death of the idiot who pulled the rods. All large reactors are therefore built with automatic control systems which make it impossible to pull the rods out suddenly. These reactors possess "engineered safety," which means that a catastrophic accident is theoretically possible but is prevented by the way the control system is designed. For Teller, engineered safety was not good enough. He asked us to design a reactor with "inherent safety," meaning that its safety must be guaranteed by the laws of nature and not merely by the details of its engineering. It must be safe even in the hands of an idiot clever enough to by-pass the entire control system and blow out the control rods with dynamite. Stated more precisely, Teller's ground rule for the safe reactor was that if it was started from its shut-down condition and all its control rods instantaneously removed, it would settle down to a steady level of operation without melting any of its fuel.

One of the first steps toward the design of the safe reactor was to introduce an idea called the "warm neutron principle," which says that warm neutrons are less easily captured than cold neutrons and are less effective in causing uranium atoms to fission. The neutrons in a water-cooled reactor are slowed down by collisions with hydrogen atoms and end up with roughly the same temperature as the hydrogen in whatever place they happen to be. In an ordinary water-cooled reactor, after the postulated idiot has blown out the control rods, the fuel will be growing rapidly hot but the water will still be cold, with the result that the neutrons remain cold and their effectiveness in causing fission is undiminished, and therefore the fuel continues to grow hotter until it finally melts or vaporizes. But suppose instead that the reactor was designed with only half of the hydrogen in the cooling water and the other half of the hydrogen mixed into the solid structure of the fuel rods. In this case, when the idiot yanks out the control rods, the fuel will grow hot and with it the hydrogen in the fuel rods, while the hydrogen in the water remains cold. The result is then that the neutrons inside the fuel rods are warmer than the neutrons in the water. The warm neutrons cause less fission and escape more easily into the water to be cooled and captured, and the reactor automatically stabilizes itself within a few thousandths of a second, much faster than any mechanical safety switch could hope to operate. So the reactor carrying half of its hydrogen in its fuel rods is inherently safe.

There were many practical difficulties to be overcome before these ideas could be embodied in functioning hardware. The greatest contribution to overcoming the practical difficulties was made by Massoud Simnad, an Iranian metallurgist who discovered how to make fuel rods containing high concentrations of hydrogen. He made the rods out of an alloy of uranium hydride with zirconium hydride. He found the right proportions of these ingredients to mix together and the right way to cook them. When the fuel rods emerged from Massoud's oven, they looked like black, hard, shiny metal, as tough and as corrosion-resistant as good stainless steel.

After we had understood the physics of the safe reactor and the chemistry of its fuel rods, many questions still remained to be answered. Who would want to buy such a reactor? What would they use it for? How powerful should it be? How much should it cost? Teller

Report of the Regional [illegible] Safety, [illegible]

Subject - Triga Reactor

Meeting held Thursday, May 12, 1960, Physics Building, UCLA, 3:00 p.m.

Present: Clara Szego-Roberts, Thomas E. Hicks, Charles A. West, Raymond L. Libby for (L. R. Bennett), Kenneth R. MacKenzie (chairman).

Invited guests: John C. Evraets, Gerald H. McDonell

Agenda

The meeting was called to review the safety aspects of a proposed Triga Mark II reactor installation on the UCLA campus. The details of this proposal are contained in a document dated May 1, 1960 entitled "A proposal and request for assistance for a MULTI-PURPOSE REACTOR FOR BASIC AND APPLIED RESEARCH AND TRAINING IN NUCLEAR REACTOR TECHNIQUES" to the National Science Foundation from the Regents of the University of California.

Committee Views and Recommendations

The minutes are summarized as follows: The proposed location of the Triga reactor was determined as just east and adjacent to the essentially completed Argonaut reactor, which is housed as part of Engineering unit III. Both reactors would be on the same level, but the Triga would be well below ground on two sides. Both reactors would be operated by the same personnel and supervision.

This committee had previously (May, 1958) examined and approved the Argonaut installation. A comparison was made between the two reactors and in no respect was the Triga found to be more hazardous than the Argonaut. In fact, in almost every respect, the safety margin seemed to favor the Triga reactor, in spite of its higher possible operating level.

A problem distinct from health and safety is the effect on low level counting in research areas. In this respect, background radiation and active gaseous effluent were considered and found to be well below the amount which would bother low level counting in adjacent buildings. Since low level counting requires a background considerably lower than health considerations, it is automatic that there is no danger to personnel on the surrounding campus.

The committee therefore finds that the proposed Triga installation is perfectly safe from an overall campus standpoint, and for personnel directly concerned it appears safer than the present Argonaut reactor.

Minutes of the Meeting

Dr. Hicks described the proposed location and how it would be integrated with the present reactor and the future building program.

Dr. McDonell provided the committee with descriptive literature on the Mark II Triga, pointing out that the Mark II, which is an above ground design, is much more flexible and useful than the Mark I and in addition has some improved

safety features in the best exchanges. He also provided the committee with copies of a "Hazards Report" prepared by General Atomics. Dr. McDonell verbally gave the committee information about a safety investigation at the University of Illinois where a similar Triga reactor has been installed. The study showed that the reactor was safe to everyone's satisfaction. A written report of these investigations will be available in the near future.

Dr. McDonell provided the committee with copies of a letter from H. R. Zeitlin of General Atomics to the La Jolla members, describing the safety test program on its prototype reactor. Pertinent parts of this letter are quoted below:

> "Enclosed is a description of the TRIGA Mark II reactor of interest to the University. Also enclosed is a report of the first phase of an experimental program conducted last year by GA on its prototype reactor. In this program the inherent safety of the TRIGA resulting from the use of a unique fuel-moderator element developed by GA has been experimentally verified. In this program we purposely rapidly ejected a control rod employing a special mechanism to do this in an attempt to simulate an accident. The result, as predicted theoretically, demonstrated conclusively that no hazards to personnel or equipment were produced. Similar experiments conducted at the Idaho National Reactor Testing Station on another type of reactor did, in fact, result in hazardous conditions. Our experiments were conducted with personnel in the immediate vicinity of the reactor at all times, whereas, those on the other type of reactor were conducted with personnel approximately one half mile distant from the reactor building.
>
> "Since the completion of the first phase of this program last June, we have subjected the reactor to more intense "pulses", again without producing any environmental hazards.
>
> "A summary of this additional experience as well as a list of the TRIGA reactors in operation or being installed, are also enclosed. Of the domestic installations the reactor at the University of Arizona is in daily operation in a former drafting classroom in the main engineering building on the University campus. The reactor at the Veterans Administration Hospital in Omaha is located in the basement of an eleven story hospital and is indeed a unique installation with regard to safety. The reactor at Cornell as well as that at the University of Illinois and Kansas State are both being built on the respective campuses."

Dr. Clara Szego-Roberts raised the question about the active argon 41 produced in the air around samples placed in the reactor, and asked how this would affect low level counting in adjacent areas. From the data in the General Atomics hazards report, the committee determined that a reasonable estimate of the maximum value that such background interference could attain, would be less than .1% of natural cosmic ray background at a distance of 150 feet from the reactor, and hence would be almost impossible to detect.

There was no reason to request that the core of the reactor be sealed to prevent escape of argon 41 (this was suggested for the Argonaut reactor) since the Triga core is inherently sealed by nature of the basic design.

Dr. Hicks injected further evidence of low expected background by stating that low level natural carbon 14 counting and continuous monitoring of natural radioactivity and fallout is currently proceeding some 150 feet from the reactor site and is expected to continue with no interference after the reactor is in operation.

Dr. McDonell suggested that safety experts from General Atomics be immediately invited to visit the campus and inspect the overall plans. The committee approved the suggestion and recommended the visit.

The chairman advised the committee that members from other campuses who could not attend the meeting had been advised that the safety of the Triga reactor was to be the subject of discussion. The members at La Jolla were the only ones in any way familiar with the Triga. The chairman determined that in general they regarded the Triga as very safe, and asked that if they found any information to the contrary, that it be forwarded in time for this meeting. No such information has been received.

K. R. MacKenzie

K. R. MacKenzie Chairman
Regional Advisory Committee on
Radiological Safety
Southern Section

KRM:bp

GENERAL ATOMIC

CON-11 (Rev. 1)

ADVANTAGES OF TRIGA FUEL FOR RESEARCH REACTORS

October 1981

ADVANTAGES OF TRIGA FUEL FOR RESEARCH REACTORS

All TRIGA fuel is made by GA with a uranium enrichment of just under 20% and thus is classified as Low Enrichment Uranium (LEU) fuel.

The following discussion of advantages applies to TRIGA fuel clusters which can be inserted in existing grid plates to convert plate-type fueled reactors, and some pin-type fueled reactors, to TRIGA. TRIGA's advantages have motivated the owners of twelve plate-type fueled reactors to convert to the use of TRIGA fuel. TRIGA LEU fuel is available now for use in existing reactors operating at steady state power levels to 50 MW.

1. UNIQUE SAFETY

All of GA's research and test reactors are fueled with UZrH. This unique fuel provides the highest degree of safety available in any type of nuclear reactor. In these days of increasing public concern with perceived hazards of nuclear facilities, these safety advantages alone should justify use of UZrH fuel.

A. The UZrH fuel has a prompt negative temperature coefficient of reactivity, vs. a delayed coefficient in aluminum-clad plate-type fuel. This allows UZrH cores to safely withstand accidental reactivity insertions that have completely destroyed plate-fueled cores.

B. UZrH is chemically stable. It can be safely quenched at 1200°C in water, while exothermic metal-water reactions take place with aluminum at 650°C.

C. High-temperature strength and ductility of TRIGA's Incoloy-800 fuel cladding provide a yield strength greater than 10,000 psi at 900°C. The aluminum cladding on plate-type fuels melts at about 650°C.

D. The UZrH fuel material has very superior fission product retention. The aluminum-clad plate-type fuels melt at 650°C, releasing 100% of the volatile fission products. Whereas, at this same temperature UZrH retains about 99.9% of these fission products even with the cladding removed.

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//           EXEC    PGM=ORIGEN,REGION=390K
//STEPLIB   CC   DISP=SHR,CSN=JGI013.MCK.CRIGEN.LC2
*** EXEC   FORTCLG,COMPILE=FORTH,RC=250K,RL=300K,RG=390K,PG='(100,100)',
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***          LIBB='SYS1.FORTLIB',LIBA='JGI013.MCK.CRIGEN.LIB'
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***KED.SYSIN   DD   DISP=SHR,DSN=JGI013.MCK.ORIGEN.OVLY
***
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//              UNIT=SYSDA,SPACE=(TRK,(200,200))
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//CC.FT07F002  DD DISP=SHR,DSN=JGI013.MCK.URIGEN.FILC2
//CC.FT07F003  DD DISP=SHR,DSN=JGI013.MCK.CRIGEN.FILC4
//CC.FT07F004  DD DISP=SHR,DSN=JGI013.MCK.CRIGEN.FILC5
//CC.FT07F005  DD DISP=SHR,DSN=JGI013.MCK.CRIGEN.FILC6
//CC.FT07F006  DD DISP=SHR,DSN=JGI013.MCK.ORIGEN.FILC7
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IEF237I  826    ALLOCATED  TO  STEPLIB
IEF237I  714    ALLOCATED  TO  FT05F001
IEF237I  716    ALLOCATED  TO  FT06F001
IEF237I  826    ALLOCATED  TO  FT07F001
IEF237I  826    ALLOCATED  TO  FT07F002
IEF237I  826    ALLOCATED  TO  FT07F003
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

Exhibit C-1-4 page 1 of 2

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	6.58E+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
EU159	0.0	9.03E-01	8.67E-01	8.80E-02	0.0	0.0	0.0
GD159	0.0	2.43E-01	2.43E-01	2.47E-01	1.02E-01	4.00E-04	2.35E-13
TB159	0.0	0.0	0.0	0.0	0.0	0.0	0.0
EU160	0.0	2.53E-01	1.89E-01	1.32E-08	0.0	0.0	0.0
GD160	0.0	0.0	0.0	0.0	0.0	0.0	0.0
TB160	0.0	8.02E-05	8.02E-05	8.02E-05	7.94E-05	7.50E-05	6.01E-05
DY160	0.0	0.0	0.0	0.0	0.0	0.0	0.0
GD161	0.0	6.43E-02	5.26E-02	7.72E-07	0.0	0.0	0.0
TB161	0.0	3.64E-03	3.64E-03	3.65E-03	3.31E-03	1.81E-03	1.80E-04
DY161	0.0	0.0	0.0	0.0	0.0	0.0	0.0
GD162	0.0	3.29E-02	3.07E-02	5.85E-04	0.0	0.0	0.0
TB162M	0.0	3.29E-02	3.28E-02	1.78E-03	0.0	0.0	0.0
TB162	0.0	0.0	0.0	0.0	0.0	0.0	0.0
DY162	0.0	0.0	0.0	0.0	0.0	0.0	0.0
TB163M	0.0	1.52E-02	1.37E-02	3.91E-05	0.0	0.0	0.0
TB163	0.0	4.08E-06	4.07E-06	3.66E-06	3.15E-07	6.76E-14	0.0
DY163	0.0	0.0	0.0	0.0	0.0	0.0	0.0
TB164	0.0	1.19E-03	1.18E-03	1.14E-03	5.72E-04	7.47E-06	4.45E-13
DY164	0.0	0.0	0.0	0.0	0.0	0.0	0.0
DY165M	0.0	9.35E-05	5.20E-05	3.33E-19	0.0	0.0	0.0
DY165	0.0	2.92E-05	2.91E-05	2.16E-05	2.27E-08	4.42E-27	0.0
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

PANEL I

REACTOR RUNAWAY OR SUPERCRITICAL NUCLEAR EXCURSION ACCIDENTS

"An outstanding characteristic of nuclear reactors is their potential ability to achieve extremely high power levels in a short time if adequate control of the machine is lost. A typical nuclear runaway accident may start and be over in times appreciably less than a second. In this respect they are different from any other large-scale machines..."

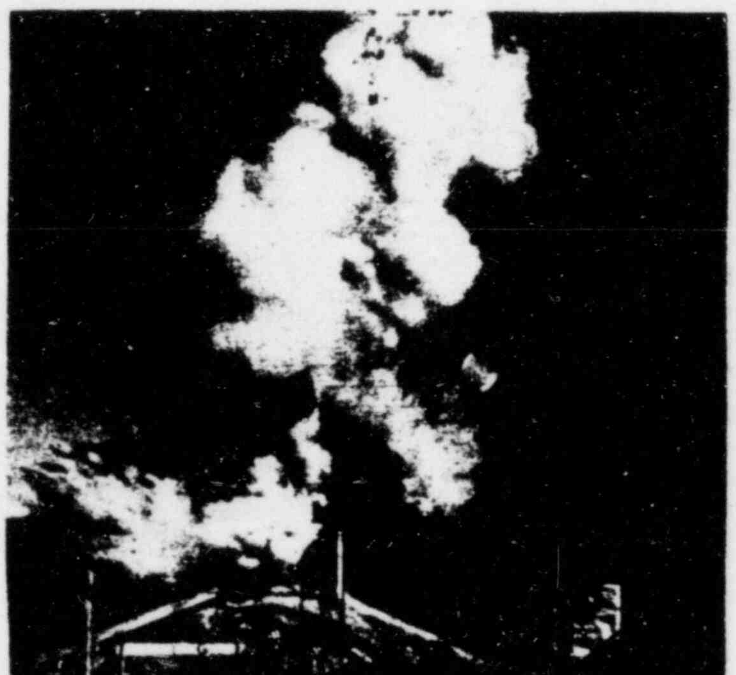
--"The Safety of Nuclear Reactors"
1955 Geneva Conference Paper by
McCullough, Mills, and Teller[1]

INTRODUCTION

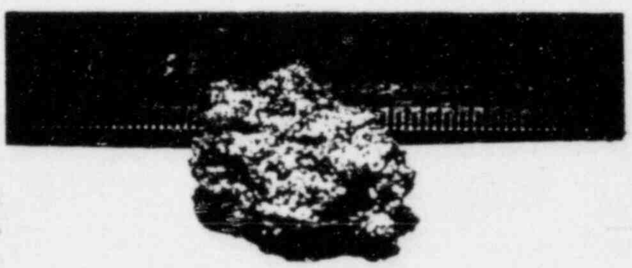
What is a Power Excursion Accident?

1. A supercritical power excursion or "reactor runaway" is an accident unique to nuclear reactors in which power can rise out of control, from zero to billions of watts, in very much less than a second. This can result in melting, and even vaporization, of the fuel and explosive disassembly of the reactor.

2. In such an accident, the power rises exponentially in periods measured in milliseconds, fuel temperatures shoot past the melting temperature, and steam explosions and often metal-water chemical explosive reactions can destroy the core. Fuel fragments may be sent high into the air, as in the photo below, and may be scattered hundreds of feet around where the core had been.



Disassembly of the BORAX reactor[2]



Pellet of spongy aluminum-uranium mixture

Fragments of fuel plates

BORAX reactor fuel fragments[2]

3. Because of the speed (periods measured in milliseconds) with which these accidents can occur, there can be no time for operator response or automatic engineered features like control rod insertion to prevent the runaway. Protection must rest on inherent design features which, nearly instantaneously, must dampen, stop, and reverse the rapid power rise before the fuel is damaged. The effectiveness of these inherent design features, and the accuracy with which they can be predicted, are extremely important in safety analysis of reactors. Despite detailed safety analysis prior to operation of some reactors, there have been some very unfortunate surprises.

A Case Study: The SL-1 Reactor Accident

4. One of the most grisly power excursion accidents occurred on January 3, 1961, at a small reactor built at the National Reactor Testing Station (NRTS) at Idaho Falls, Idaho. The reactor, known as the Stationary Low-Power Reactor Number One (SL-1 for short), had been built for experimental and training purposes. It had many features similar to the UCLA reactor: fuel that was highly enriched, made of uranium-aluminum alloy and clad in aluminum, formed into flat plates bolted together as bundles, water-cooled and -moderated. (There were differences as well-- the effects of similarities and differences between the UCLA Argonaut-type reactor and the SL-1, the BORAX, and the SPERT reactors will be discussed in more detail below.)

5. At the time of the accident at the SL-1, it had been shut down for routine maintenance, and three men were completing certain preparations for nuclear startup. Apparently, in the process of attaching control rods to drive motors, one of the men raised the central control rod too far and too fast, triggering a supercritical power excursion in the reactor core. In a fraction of a second the power reached a magnitude of an estimated several billion watts, melting and perhaps even vaporizing a large part of the core. The water in the core region was vaporized, creating a devastating steam explosion. The remaining water in the reactor vessel was hurled upward at high velocity, striking the underside of the reactor's lid and lifting the whole nine-ton vessel upward, shearing cooling pipes in the process and crushing the men who had been on top of the reactor vessel against the ceiling of the building before the vessel dropped back into place. One of the men remained impaled on the ceiling by a piece of control rod rammed through his body. It all happened in a second or so. (Even had the worker, as he pulled the control rod out, realized he had pulled it out too far, there would not have been time to push it back in before the reactor exploded, so fast is the exponential power rise in a runaway reactor.)

6. It was a terrible accident, made even more grisly because the intensely radioactive fission products scattered inside the building by the accident hampered the work of recovering the bodies. Staying in the building for mere seconds resulted in a year's allowable dose of radiation for the rescue workers. And it took several days to remove the body that was impaled on the ceiling, requiring use of a remotely operated crane and closed-circuit television because of the intense radiation fields. The bodies were so badly contaminated, the heads and hands of the victims had to be severed and buried with other radioactive wastes at the NRTS.

The Lessons of the SL-1 Accident

7. The accident at SL-1, and other similar accidents, point to the inherent (and often unanticipated) dangers in certain reactor designs, and the extreme importance of assuring that safety analyses have not overlooked design flaws that could lead to or contribute to serious accidents such as the one at SL-1.

8. This is especially true when considering a reactor such as UCLA's, located without a containment structure or exclusion zone in the midst of a densely populated area, where the consequences of a serious accident could be far greater than the accidents which have occurred to date. Virtually all of these occurred, fortunately, in remote locations with considerable distance between them and any populated centers.

9. Therefore, review of potential accidents must ensure, to a very high degree of certainty and with very large safety margins, that the UCLA reactor is inherently safe from destructive power excursions--i.e., that intrinsic self-limiting features would prevent a power rise of sufficient magnitude to do to the UCLA reactor what the incident at SL-1 did to that reactor.

Varying Effectiveness of Inherent Safety Features

10. Certain research reactors, particularly the TRIGA, seem to meet, to a significant degree, the standard of inherent safety as to power excursion potential. In the case of the TRIGA, this is because part of the moderator is built right into the fuel, so that there is no time delay in the shutdown mechanism. (This will be explained in more detail later.) With the TRIGA, if an excursion occurs and power starts increasing, fuel temperatures increase as well, which promptly cuts off the power rise because the shutdown mechanism is built into the fuel itself. There is no time delay. (Reactors that use low enriched fuel have a similar prompt component to the shutdown mechanism, Doppler broadening involving neutron capture in the Uranium-238 which is likewise part of the fuel itself. This too will be discussed in more detail later.)

11. The shutdown mechanisms in the case of the UCLA Argonaut-type reactor, on the other hand, are far slower in operation.* The heat that is generated in the fuel meat by the power rise must be transferred through the meat to the cladding, through the cladding to the water coolant, and then that heat must vaporize sufficient water to produce enough voids (steam bubbles) to reduce moderation and stop the reaction. For a range of power excursions, this shutdown mechanism is too slow to prevent the runaway from reaching power levels (and thus temperatures) sufficient to melt the fuel. And,

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* Furthermore, a number of safety features with which the Argonaut was originally designed have been altered or have eroded over the years, thus further reducing safety margins at UCLA. For example, the UCLA reactor was originally designed with an excess reactivity limit less than that necessary to go prompt critical; a few years later the excess reactivity limit was increased substantially, to precisely the level that the calculations in the UCLA Hazards Analysis indicate could cause fuel melting. More on this below.

as indicated earlier, such power excursions can occur far faster than either human operators or automatic engineered safety features can respond.

Unpredictability of Destructive Threshholds

12. For reactors that don't have the inherent safety of extremely prompt shutdown features, such as those of the TRIGA, there is considerable uncertainty where the danger point lies. A wide variety of factors (such as fuel meat and cladding thickness, void and temperature coefficients, among others) will permit one reactor to tolerate safely an excursion of a particular magnitude while causing another to be totally destroyed. Furthermore, predictability for even the same reactor is not terribly good.

13. To obtain a better understanding of the mechanisms of self-shutdown, a series of tests were performed on certain reactor designs at the NRTS in the 1950s and 1960s. Several of these reactors were tested to destruction-- i.e., permitted to run away at rates sufficient to cause fuel melting and core explosion. One such was the BORAX; another was the SPERT. These tests taught us much about the mechanisms of power excursions in certain kinds of reactors, although they revealed large uncertainties as well.

14. Perhaps the most significant conclusion of the SPERT destructive tests was the unpredictability of destructive threshhold during power excursions, even with a reactor that had been as thoroughly studied as SPERT had been. While fuel melting had been expected in the final test (because melting had been observed in previous tests with somewhat smaller reactivity insertions), the violent explosion which demolished the reactor came as a surprise. Although the BORAX and SL-1 reactors had suffered similar explosions, there had been no prior indication at SPERT that going to a period slightly smaller than that of previous tests represented crossing a threshhold for SPERT which made possible the violent pressure pulse which would demolish the core.*

15. Even after an extensive series of actual tests with the SPERT reactors, there is much about the behavior of those reactors during power excursions that remained poorly understood and difficult to predict. This is considerably more the case with regards the potential behavior of reactors substantially different from those tested-- for example, the UCLA Argonaut.

Dangers of Extrapolation from One Reactor Type to Another

16. The uncertainties are vastly greater when comparing an excursion of one magnitude in a particular reactor, not against an excursion of a different magnitude in the same reactor, but against a completely different reactor,

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* As T.J. Thompson put it, describing the SPERT I destructive test, "The sudden onset of total core destruction for only a factor of two increase in total energy deposited was a surprise. It emphasizes the need to carry on such extrapolation tests at remote sites such as NRTS." (Technology of Nuclear Reactor Safety, "Accidents and Destructive Tests," p. 685).

with different design features and nuclear characteristics. It is those uncertainties, the extremely large error margins that must be considered in applying data from the BORAX and SPERT tests to a reactor such as the UCLA Argonaut, whose fundamental design and key variables are quite different, which represent one of the key matters to be addressed.

17. Several analyses, relying heavily on the SPERT I tests, have been performed purporting to predict the potential behavior of the UCLA Argonaut-type research reactor during power excursions that might be initiated by insertion of that reactor's available excess reactivity. Because of the large amount of excess reactivity requested (far more than that capable of producing supercriticality on prompt neutrons alone), and because of the highly populated site and lack of a containment structure, we have paid special attention to those portions of the documents which attempt to analyze the capacity of the UCLA Argonaut reactor to undergo a destructive power excursion, one that could result in release of fission products to the environment.

18. The three key attempts to apply BORAX and SPERT experience to the UCLA Argonaut case have been examined. In each case, correction of just a few erroneous or non-conservative assumptions or values demonstrates the potential for a seriously destructive power excursion at the UCLA reactor, assuming the correctness of the basic method applied in the original analyses.

SPERT-ID Destroyed by $3.50 Insertion

19. For example, as will be discussed in more detail below, UCLA has asserted that its Argonaut-type reactor can safely tolerate a far larger excess reactivity insertion (a measure of the initiating event for a power excursion, often measured in "dollars") than its original design limit. UCLA appears to rest most of its case in that regard on the assertion that the BORAX I and SPERT I tests "proved" that that level of excess reactivity is safe in the Argonaut. As UCLA put it in its 1980 Application (p. V/3-6), with similar statements in the 1982 version:

> "SPERT and BORAX tests showed that plate type fuel elements survived step reactivity insertions of $3.54."

20. That simply isn't correct. In fact, the SPERT-ID reactor core was completely destroyed by a $3.50 insertion, which resulted in explosive disassembly of the core and extensive melting of the fuel, as can be seen from the attached photos from that SPERT excursion.[3]

21. However, even were it correct that SPERT survived a $3.54 insertion, which it did not, that would in no way mean that the UCLA Argonaut, a different reactor, would likewise survive such an excursion. Extrapolations from SPERT or BORAX experience to the UCLA Argonaut require numerous corrections for different reactor characteristics, each correction increasing the magnitude of the margins of error that must be assumed.

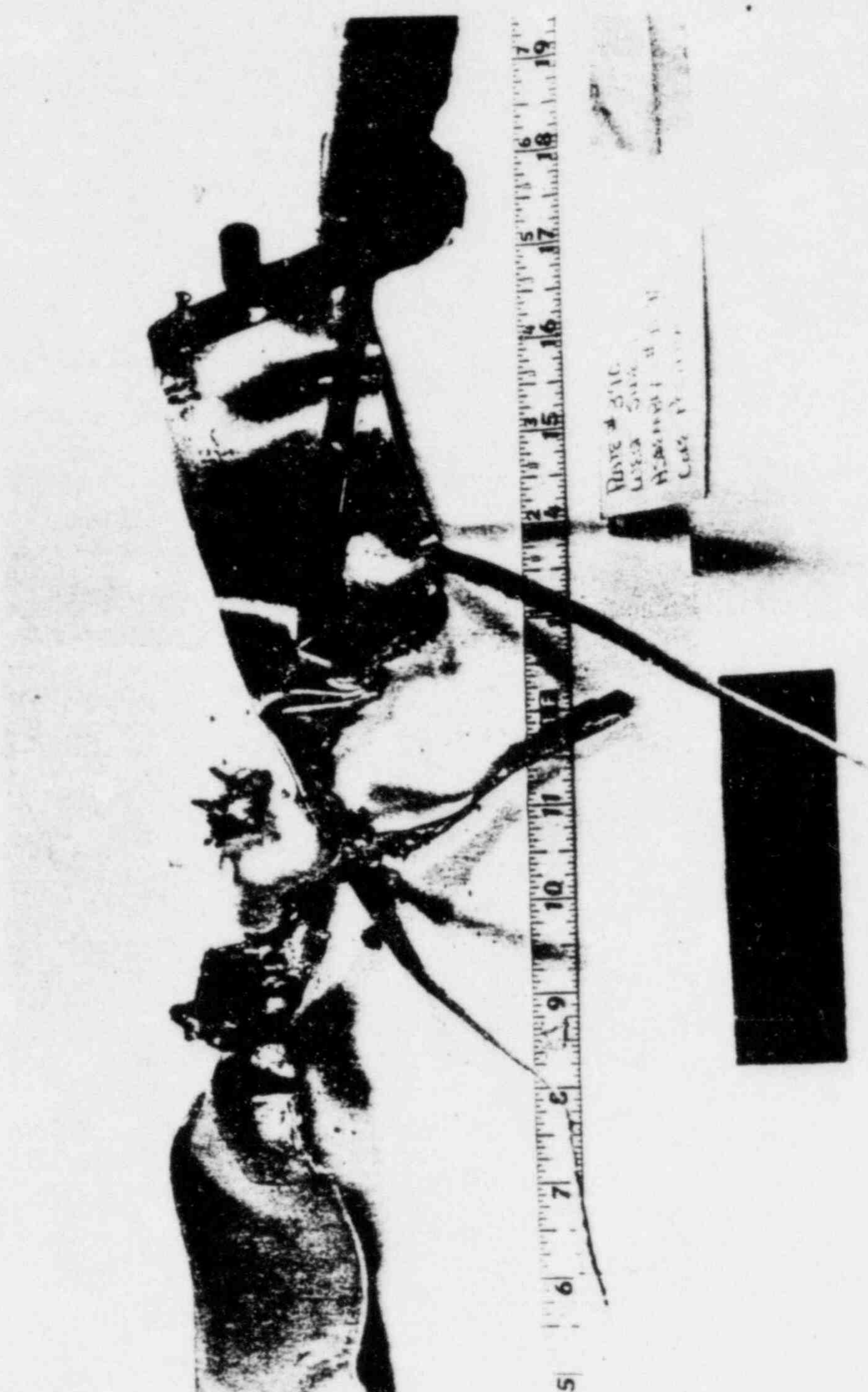
Summary of Conclusions

22. Based on our review, it is concluded that the amount of excess reactivity requested by UCLA is too high, the safety margins too small, and the potential for a destructive power excursion unacceptable, especially given the local population density, lack of containment structure and exclusion zone, and student operation, among other factors. The analyses done to date do not, we believe, demonstrate that such an accident is not credible. In fact, because of errors made in each, the analyses indicate, when the errors are corrected, that such an accident is indeed credible. Questionable methodological assumptions employed by the analysts suggest that a definitive answer as to the maximum "safe" reactivity insertion for the UCLA reactor, or even an answer merely providing reasonable assurance of its being right, would require further research.

23. Because of the substantial differences between the Argonaut and the reactor types previously investigated, that research would likely necessitate SPERT-type tests on actual Argonaut cores. In the absence of such definitive research, very substantial margins of safety are essential. Furthermore, the changes made since the UCLA facility began operation have, in our opinion, resulted in a gradual but significant erosion of important safety margins, making a potentially serious accident at the UCLA facility both more likely and of greater potential consequence.

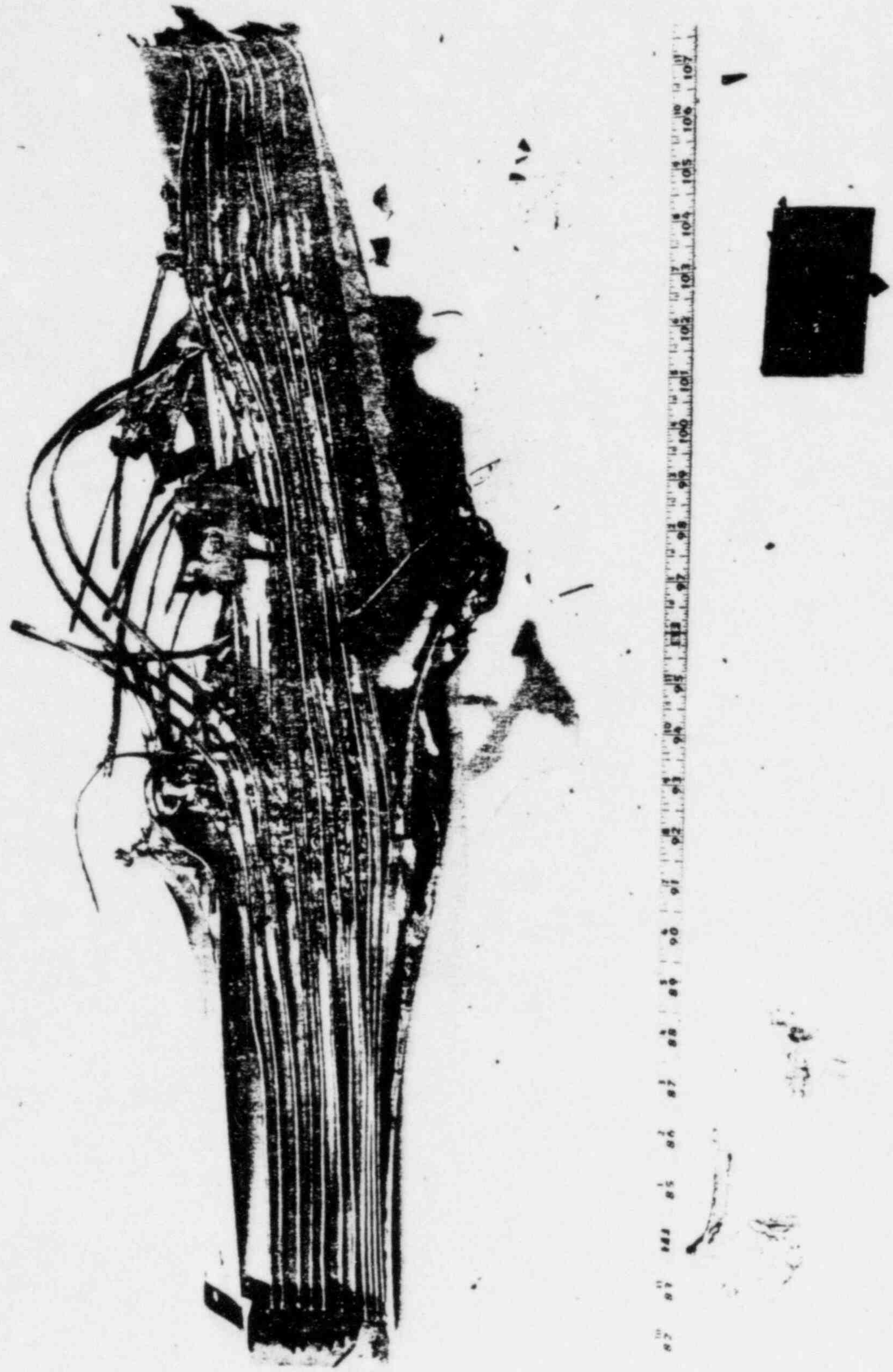
24. In what follows, we discuss the basic nuclear concepts related to power excursions, review the analyses done to date, discuss some of the ill-advised alterations to the UCLA Argonaut over the years, and indicate some of the many potential mechanisms for initiating such an event.

25. We conclude that a series of unfortunate design features and post-design modifications conspire to make the UCLA Argonaut, with its current characteristics, not inherently protected against destructive power excursions of the kind that destroyed the SL-1. Adequate margins of safety are lacking.



Fuel plate end fragments from assembly D5.

(from SPERT-I destruct test)[3]



Typical plates from peripheral assembly-edge view, position G6.
(from SPERT-I destruct test)[3]

BRIEF EXPLANATION OF BASIC NUCLEAR PHYSICS CONCEPTS RELATED TO POWER EXCURSIONS

26. This section constitutes a brief explanation of some of the basic concepts of nuclear physics related to power excursion phenomena. Brevity will necessitate considerable simplification.

Criticality and Delayed Neutrons

27. A reactor such as the UCLA Argonaut-type reactor operates by fissioning atoms of uranium-235 which, when hit by neutrons of particular energies, will split apart. In the process of fissioning, uranium-235 atoms release additional neutrons which can then cause other uranium-235 atoms to fission, creating a chain reaction, which can be self-sustaining.

28. 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. 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.

29. 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.

30. 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.

Supercriticality and Prompt Neutrons

31. Such a runaway condition, where the neutron population grows uncontrolably, is called a prompt supercritical 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 respond; i.e., that it not become "prompt supercritical," or supercritical on prompt neutrons alone.

32. When a reactor is running on delayed neutrons-- i.e., when the reaction needs both the 99.35% of neutrons that are prompt <u>and</u> the 0.65% 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.

33. 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.

Exponential Increase in Power in Milliseconds

34. 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 <u>period</u> (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.

35. 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 approahes 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.

36. 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. (Any remaining intrinsic safety features will be discussed shortly.)

Excess Reactivity

37. 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.

38. 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.

"Dollars" and $\%\Delta k/k$

39. 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 $\beta = \$1$. If the percent notation is used, the units are in percent of $\Delta k/k$.

40. In short, if a reactor such as UCLA's has available more than \$1 or $0.65\%\Delta k/k$ excess reactivity, and that excess reactivity were for some reason inserted into the core, the reactor's normal engineered safety features such as control blades might be unable to shut the reactor down before power levels sufficient to melt the fuel were attained and the contained fission products released. It is for this reason that the original design of the UCLA Argonaut prohibited the reactor from ever having available more than $0.6\%\Delta k/k$ excess reactivity.

Inherent Shutdown Features Affecting Termination of Super-prompt Power Excursions

41. If a reactor were to go supercritical on prompt neutrons, power would keep increasing essentially exponentially until the excess reactivity were somehow removed. As indicated above, engineered safety features are too slow-acting to be of use in stopping very fast excursions. The only remaining means of shutdown, then, are inherent features-- fast acting, non-mechanical automatic responses not dependent upon operator action, but rather intrinsic design and the laws of physics.

"Doppler" Effect

42. For example, with reactors utilizing low enriched fuel, the neutron capture rate will undergo "Doppler broadening" as temperature in the fuel rises. With increased neutron capture, the number of neutrons available for fissioning is reduced, dampening the power rise. Unfortunately, this effect is virtually nonexistent in highly enriched fuel such as that used currently at UCLA, and is a strong argument for converting the fuel at UCLA from highly enriched uranium to low enriched uranium, for safety in addition to non-proliferation reasons.

43. Use of a low-enriched fuel would add some safety margin to the facility, because of the increased Doppler effect. At SPERT, it was found that a low-enriched, uranium oxide core was able to withstand larger reactivity insertions than the highly enriched uranium-aluminum plates. A reduction in enrichment of the fuel can thus decrease the possibility or consequences of a destructive power excursion or criticality accident. Low enriched fuel has a contribution to limiting an excursion, an abating effect, that is the cancelling out of reactivity because of the increased capturing of neutrons by uranium-238. Highly enriched fuel, of course, is highly enriched in uranium-235, with very little uranium-238 present, so the Doppler effect is essentially not present in such fuel.

Moderator Heating, Void Formation

44. Other factors which can help 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.

Final Shutdown Mechanism: Disassembly of the Core

45. 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.*

46. 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. The core destruction and fuel melting that can ocur from such an excursion is indicated by the photos, films, and videotapes of the BORAX I and

*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.

SPERT-ID final power excursions.

Inherent Safety: Very Large and Very Prompt Negative Coefficients of Reactivity

47. 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 the temperature accordingly, 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).

48. Degree of inherent protection, i.e. the magnitude and promptness of self-shutdown mechanisms, varies 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.

49. These positive reactivity coefficients create positive feedback loops, so that as the power rises, so does the temperature which, rather than force the power back down, pushes it even higher. This can obviously be potentially quite dangerous. Often a reactor will have several reactivity responses, some of which will be negative and some positive. Even if the positive coefficient does not predominate, it can severely limit the effectiveness of the negative coefficient, permitting a far longer power rise before the negative coefficient succeeds in producing shutdown, and thus far greater energy release and higher temperature for the fuel. The UCLA reactor appears to have several such positive reactivity coefficients, as will be discussed later.

Some Reactors' Inherent Shutdown Features More Effective Than Others': Case in Point, the TRIGA

50. 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.

51. The main feature of TRIGA research reactors and the TRIGA fuel-moderator elements (even when used in reactors originally designed for flat plate fuel) is a very strong and very prompt negative temperature coefficient, much more prompt than that of other research reactors, which effectively controls large prompt positive reactivity insertions. Any sudden increase in power heats both the fuel and the moderator simultaneously,

causing the moderator to become less effective immediately and to return the reactor automatically and instantaneously to normal operating levels. Such control is intrinsic to the TRIGA reactor fuel and does not rely on mechanical or electrical control devices. This most important property is due to the fact that the fuel elements are constructed of a solid homogenous alloy of uranium fuel and zirconium hydride moderator, making them "fuel-moderator" elements.

52. There is, of course, a level of excess reactivity above which that safety feature of not being able to damage any of the fuel with an accidental excursion is no longer true, but that level would be much, much higher for a TRIGA reactor or a reactor with TRIGA fuel than for the UCLA Argonaut with its current flat plate MTR-type fuel. At comparable levels of excess reactivity, the TRIGA fuel would definitely have significant inherent safety advantages.

53. In the TRIGA fuel, when the ratio of captures (in the water and other materials) to fissions (in the fuel) goes up, the reactivity goes down. That effect is produced by a change in temperature in the fuel itself, relative to the cooling water, and thus requires no heat conduction. It happens instantaneously because the heat is liberated by the fission reaction right in the fuel. In the UCLA Argonaut-type reactor, the heat has to be transferred to the water, which takes a while, to make it expand. It is the expansion of that water plus some other effects that have to do with the water having to heat up that makes the reactivity go down. Because of this time-delay involved with the transfer of heat from the fuel meat to the water in the UCLA reactor, the shutdown mechanism is slower. This allows for greater energy release before shutdown for an excursion of the same exponential period, and a greater opportunity for fuel melting to occur before the excursion terminates than is true with the TRIGA reactor or reactors converted to TRIGA fuel.

54. The shutdown mechanism in the UCLA Argonaut, which requires transfer of the heat to the water to cancel the reactivity, can produce effects in the water like boiling or a sudden expansion of the coolant which can, in effect, do some damage even if fuel melting does not occur. The likelihood of changes in the fuel arrangement or other core rearrangement or damage is less for the TRIGA than for the UCLA Argonaut, for the same reasons that the TRIGA is considerably more protected against excursions leading to melting than is the Argonaut.

The UCLA Argonaut: Less Effective Shutdown Features

55. Other reactors, such as the Argonaut, have inherent self-limiting features far less prompt and effective than research reactors like the TRIGA. The original Argonaut reactor, and some of the university Argonaut-type reactors that followed it, used 20% enriched fuel, providing some prompt Doppler contribution to reactor shutdown in an excursion. But for the UCLA Argonaut, utilizing as it does highly enriched plate-type fuel, with little U-238 and no moderator in the fuel meat itself, essentially the only inherent self-shutdown mechanism short of full core disassembly that can limit an excursion is: transfer of the rising heat from the fuel meat to the cladding to the water moderator and then formation of steam and expulsion of the remaining water. This void formation/water expulsion reduces moderation, increases neutron leakage, and eventually stops the reaction.

56. In such a reactor, this last remaining shutdown feature is far slower than that of the TRIGA reactor, in which there is practically no time delay necessary for heat transfer. The delay involved in transferring heat from the fuel meat to the water, for excursions of short exponential period, can prevent self-shutdown occurring before the reactor has reached a dangerous level, resulting in fuel melting and possible explosion. It is very difficult to estimate with a high degree of certainty for different reactor designs what the precise limiting period would be, i.e., at how short a period the formation of voids would cease to occur in time 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).

57. The UCLA Argonaut-type reactor is now left with this one, delayed inherent shutdown feature (void formation in the water), lacking the Doppler effect of low-enriched fuel of the original Argonaut, the very large, very prompt negative coefficients of the TRIGA, and the protection provided by its own original design limitation to less excess reactivity than that necessary for prompt criticality. Furthermore, compared with the BORAX and SPERT reactors, upon which the primary reactivity tests have been performed and against which it has been compared in safety analyses, UCLA's sole remaining shutdown mechanism of temperature effects in the water coolant/moderator is far less effective in limiting a power excursion than those effects were for BORAX and SPERT.

58. Reported void and temperature coefficients for the water moderator are considerably smaller for UCLA than for BORAX or SPERT. A principal design difference between the UCLA Argonaut and the BORAX and SPERT reactors is that the moderator and reflector for the latter was solely water, whereas the UCLA Argonaut has, in addition to a water moderator, large volumes of graphite as moderator and reflector. While the sole moderator at BORAX and SPERT can be readily expelled through steam formation, only part of the moderator can be expelled at UCLA. The solid graphite will remain, reducing the effectiveness of a shutdown mechanism depending upon void formation.

Positive Feedback Features

59. In addition to relatively less effective negative feedback features, the Argonaut appears to have certain unfortunate positive feedback features.

60. For example, the graphite has a positive temperature coefficient of reactivity. As it is heated, instead of dampening a power rise as is the case with water moderators and reflectors, the power rise is enhanced. The positive reactivity coefficient for graphite has been known for years, at least in fact since the Manhattan Project era, and it is something of a surprise that UCLA didn't know of it and had to learn of its existence from another reactor.[4] This is a good example of the importance of reviewing reactor experience from similar reactors for potential generic problems.

61. Another example: If the reactor water is dumped and the reactor is not scrammed, power rises at first and only thereafter decreases. This is due to over-moderation of reflected neutrons in a portion of the core region. As water level drops, the over moderation decreases, and reactivity increases, until ideal moderation is reached; if the water level drops further, the reactivity trend reverses and finally starts to drop.[5] (In other areas of the core, the reactor is severely undermoderated so that if fuel bundle spacing, or plate spacing, are altered, or other forms of core disruption take place, including flooding, positive reactivity effects may occur.)

62. Given the relatively ineffective nature of the sole shutdown mechanism in the UCLA Argonaut-- production of voids in one of the reactor's two moderators-- and the additional problem of several potential positive reactivity effects, the reactor is far less inherently safe than other reactors of the TRIGA type or of the SPERT/BORAX variety.

ORIGINAL INHERENT DESIGN FEATURES OF THE UCLA ARGONAUT, AND HOW THEY HAVE BEEN ALTERED OVER THE YEARS

> "A reactor which is to be used for student instruction must be designed so that safety is insured without exercising greater restraint on the activities of students than is normally advisable in a university laboratory. This necessitates: (1) that the total available excess reactivity be limited to something less than that needed for prompt criticality, (2) that the reactor have a high degree of demonstrated inherent safety, and (3) that it be limited to low-power operation."
>
> --original UCLA Reactor Hazards Analysis, p. 19

Original Safety Premises

63. 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 that virtually 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, made of low melting aluminum. 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.

64. 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 (even so, it estimated thyroid doses as high as 1800 rem for a release of only 10% of the radioiodines); and it concluded also that excess reactivity should be limited to less than that necessary for prompt criticality, for which engineered safety features could 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 operation by students, who can be expected to make mistakes, and given the uncertainties in the calculations (which meant that melting might occur far below 2.3%), restricting the reactor to 10 kw_{th}* and to less than that necessary for prompt criticality was determined necessary.

65. The original Hazards Analysis for this reactor begins a discussion of the general safety premises applied to its design with the following statement:

> "The inherent safety of the reactor is based on four points. First, the amount of excess reactivity in the reactor is limited to about 0.6%. Second, the reactor has negative thermal and void coefficients. In addition, the reactor is provided with sufficient interlocks and safety trips to make a hazardous incident extremely improbable. Third, the amount of contained fission products will be relatively small since the reactor is to be limited to a maximum power of 10 kw. Fourth there is no credible way in which the fission products can be made to escape." (p.59)

Intrinsic Safety Features of the Reactor Have Been Substantially Mitigated or Removed

66. In the years since the original design and analysis were completed, each of the above four bases for the supposed inherent safety of the reactor has been substantially mitigated. First, the licensed limit of excess reactivity is no longer restricted to 0.6%, less than that necessary to go prompt critical, but is now nearly four-fold larger, at precisely the level the calculations in the Hazards Analysis indicate could cause melting. Second, the reactor has been found to have unexpected positive coefficients and feedback mechanisms, as discussed in the previous section. In addition, the reactor's staff has over the years found ways to disconnect interlocks and safety trips, and the value of the latter

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* The power limitation was important, the Hazards Analysis indicated, because it limits the consequences of an accident, should one occur, by limiting the radioactivity available for release to the environment: "[T]he amount of contained fission products will be relatively small since it is limited to a maximum steady state power of ten kilowatts." ibid. The increase in power to one hundred kilowatts thus brought with it a concomitant increase in fission product inventory and in possible consequences should an accident result in release of that inventory.

has been brought into serious question by lack of accurate calibration. Third, the amount of contained fission products is no longer small relative to twenty years ago since the power limit has increased tenfold. And fourth, there are a number of credible ways in which fission products can be made to escape, including power excursions made possible by the increase in excess reactivity available and other factors.

67. In addition to the quadrupling of excess reactivity, far beyond the prompt critical limit prescribed by the Hazards Analysis and up to the precise level which its calculations indicate could cause fuel melting; and in addition to the tenfold increase in reactor power, a number of other developments over the years at the UCLA reactor have considerably reduced the safety margins presumed initially. These include discovery of smaller-than-expected negative reactivity coefficients (in addition to the unexpected discovery of several positive reactivity effects); apparent lack of deflectors, as designed, to prevent repeated criticality; enlargement of irradiation ports, making possible insertion of larger samples; and the addition of a pneumatic tube "rabbit" system, which makes possible new mechanisms for rapid insertion and removal of reactivity. (Other oversights in the original design review, such as errors about combustibility of the reactor constituent materials and Wigner energy storage, are discussed elsewhere; they too can impact upon effects of reactivity accidents at this reactor.)

68. The original design for the UCLA reactor called for substantial inherent safety features as well as large margins of safety: 10 kw_{th} power limitation, large prompt negative temperature and void coefficients, and excess reactivity below that necessary for prompt critical. As for the restriction on excess reactivity to below 0.6%Δk/k, the Hazards Analysis said:

> "it is possible to operate the reactor with an amount of excess reactivity which is well below that required for prompt criticality. Under these condition, the reactor meets the safety requirements of a training reactor and can tolerate considerable operational error without damage." (p. 19)

If the reactor ever met those safety requirements, it no longer does.

THE DANGER OF EXTRAPOLATING, WITHOUT VERY LARGE ERROR BARS OR SAFETY MARGINS, FROM SPERT AND BORAX TESTS TO THE UCLA ARGONAUT

69. UCLA argues that none of the alterations or problems that may have occurred during the reactor's operating history to date are of consequence because the reactor is protected by inherent design against significant fission product release. In particular, UCLA argues in its license renewal request that its reactor can safely tolerate a far larger excess reactivity insertion than the reactor's original design limit. UCLA appears to rely heavily on an assertion that the BORAX I and SPERT I tests conducted at the NRTS in Idaho in the 1950s and 1960s "proved" that the requested level

of excess reactivity is safe in the UCLA Argonaut. As UCLA put it in its 1980 Application:

> "SPERT and BORAX tests showed that plate type fuel elements survived step reactivity insertions of \$3.54." (p. V/3-6)

70. As indicated in the introduction, that simply is not the case. The SPERT I reactor core was completely destroyed by a \$3.50 insertion. In fact, non-explosive melting of fuel was observed with even smaller reactivity insertions.

71. It is also worth recalling that the grisly SL-1 reactor accident, which occurred at the Idaho Testing Station not far from SPERT, was initiated by about the same reactivity insertion (in the SL-1 case, 2.4% $\pm$ 0.3% $\Delta k/k$ *). This resulted in an energy release several times greater than that which destroyed SPERT I, sufficient, in fact, to not merely melt the fuel but vaporize parts of it. The resulting steam explosion was so intense that the whole nine ton reactor vessel was lifted nine feet in the air.

72. Even were it true that plate type fuel elements survived step insertions of \$3.54 at SPERT-- which they most certainly did not (as is plainly demonstrated in the photos of melted plates from the \$3.50 excursion found on pages 7 and 8)-- that would by itself say nothing about whether plate fype fuel would survive the same insertion in the UCLA Argonaut, a different reactor design with significantly different operating characteristics. There is no magical relationship, as the UCLA statement cited in paragraph 69 above implies, between reactivity insertion and fuel plate response, <u>independent of the reactor</u> in which the excursion is occurring. A reactivity insertion of \$3.50 will melt one core, while leaving abother virtually untouched, depending upon a whole litany of varying characteristics-- plate thickness, coolant channel width, void coefficient, moderator temperature coefficients, the presence of a non-expellable moderator such as graphite, the metal-water ratio in the core, plate surface area, degree of burnup and corrosion, prompt neutron lifetime, fuel enrichment and uranium weight %, starting moderator temperature, and many other factors.

73. Even had SPERT not been destroyed by a \$3.50 insertion, the UCLA statement quoted above could not be true, because it implies that SPERT and BORAX tests proved that plate fuel could not be damaged by reactivity insertions of \$3.54, no matter in what reactor and under what conditions it was placed. And if anything was learned through the SPERT tests, it was that seemingly minor variations, even within the same reactor (e.g., degree of subcooling), could significantly affect the total energy release and thus, whether fuel melting occurred. Differences between different reactor types were even more pronounced, affecting the very nature of the shutdown mechanism that terminates, and thus limits, the excursion itself. The SPERT and BORAX tests could not, by any stretch of the imagination, "show" that a certain general kind

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* \$3.54 would be the equivalent of between approximately 2.3% and 2.7% $\Delta k/k$, depending upon the value used for the delayed neutron fraction. (The SL-1 was a low-power experimental and training reactor utilizing highly enriched aluminum-uranium flat plate fuel, cooled and moderated by water, similar to BORAX and SPERT.)

of reactor fuel (e.g., flat plate) could survive a \$3.50 insertion in any imaginable reactor.

74. The important question, then, is not what reactivity insertion destroyed SPERT or BORAX or SL-1, or even what insertion could be expected to be the minimum necessary to induce melting in those reactors, but rather, what level is a safe level for the UCLA Argonaut, with sufficient margins of safety consonant with student operation in a densely populated location. After all, SPERT, BORAX and SL-1 were all destroyed in the Idaho desert far from any populated center. And the UCLA Argonaut-type reactor is a substantially different reactor than the three Idaho reactors mentioned above.

75. The differences are significant. Plate and meat thicknesses are different, as are coolant channel widths. The SPERT tests used essentially fission-product free cores, with fresh cladding. UCLA's fuel has been irradiated for two decades, can be irradiated for another two decades if relicensed, and has been sitting intermittently in water, allowing for some degree of corrosion, for many years. Each of those factors might affect the heat transfer time to the water *, potentially elongating the transient and increasing the energy release, factors not analyzed in the existing reports.

76. Furthermore, SPERT and BORAX were entirely water-moderated and -reflected, as was SL-1. UCLA's reactor is moderated by both water and graphite, and reflected by graphite. This lengthens the neutron lifetime, producing a longer period for any given reactivity insertion, but it also significantly reduces the value of the shutdown feature caused by expulsion of the water portion of the moderator. In the UCLA case, part of the moderator and reflector, i.e. the graphite, cannot be expelled from the core during the normal course of an excursion, thus reducing the effectiveness of moderator voids in limiting the peak power reached. And further, the reported void coefficient is smaller for UCLA than SPERT or BORAX, as is the temperature coefficient for the water portion of the moderator. The positive coefficient for the graphite further weakens the size of the shutdown mechanism for UCLA, and the positive reactivity effects noted when water level initially drops in the core and when fuel plate spacing (and/or bundle spacing) is altered, as by oscillation, are other important differences.

77. These differences can be very significant in determining the energy release from any particular excursion and whether fuel melting will result. Even different reactors of the same general type produced widely different energy releases for the same period, as is shown in the plot of energy versus reactor period on the next page, taken from Thompson and Beckerley's

* A potentially significant factor not considered in the analyses to date is the reduction of thermal conductivity in the fuel due to irradiation. A relatively small degree of burnup can result in reduction of thermal conductivity to half its initial value. (see J.A.L. Robertson, Irradiation Effects in Nuclear Fuels, p. 261; Tipton (ed.), Reactor Handbook, vol. 1, Materials, p. 192; Report TID-7515, part 2, p.13)

E-1	E-2
2.4	2.4
S.V	S.V
2.3	1.4
175	125
1.2	2.7
2000	2000
4.8	1.0
7.6	2.9
64	50
39.1	39.4
130	215

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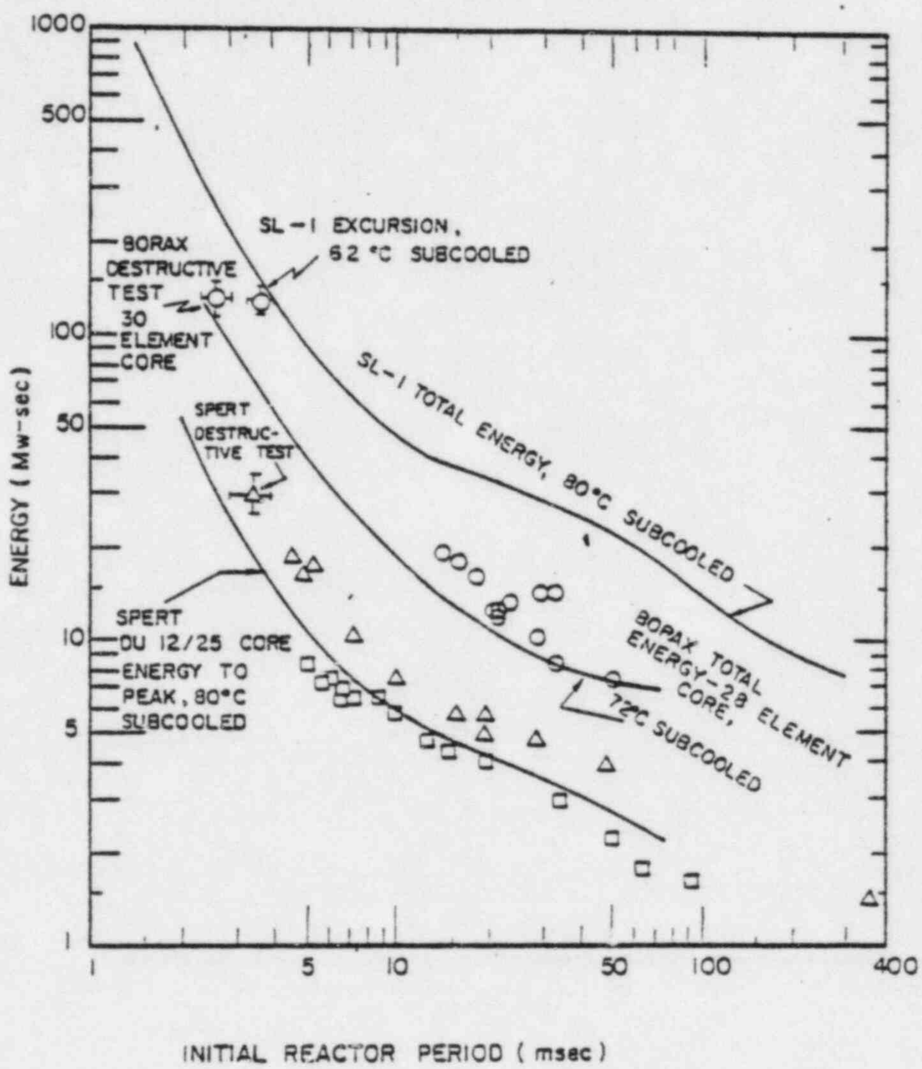

FIG. 3-30 Predicted and measured nuclear energy release vs. period for BORAX-I, SPERT-I, and SL-I. Total energy unless otherwise specified. Circles are BORAX data from reference [19], squares SPERT-I OU 12/25 data from [65a] and triangles SPERT-I destructive test data from [65].

a single slug. The water level in the tank was about 2.5 ft (76.2 cm) below the top of the vessel and the slug, therefore, had this distance to acquire kinetic energy. This slug hit the bottom of the of the plate area in the central 16 elements reached the vaporization temperature and this caused more steam production and violent destruction of this region. About 20% of the entire core shows melting proceeding to the clad surfaces. General Electric estimates that the total nuclear transient energy was 133 ± 10 Mw-sec and that no more than an additional 33 Mw-sec of energy (best estimate 24 ± 10 Mw-sec) was released in chemical reactions between the molten or vaporized metal and water.

The formation of the steam void terminated the nuclear transient, but it also created a high pressure region. The pressure wave front which developed no doubt spread out in all directions, striking the vessel side walls next to the core first and bulging them, then striking the bottom head and giving a net downward force on the vessel,* and finally accelerating upwards the entire mass of water above the core. It appears likely that the water moved upwards more or less as

*Apparently no one has looked into this downward force and one can only conjecture as to whether this downward force was sufficient to sever the pipe connections to the tank. It is difficult to judge the resistance to such a shock provided by the vessel supports.

Technology of Reactor Safety Thompson + Beckerly 1964

Technology of Nuclear Reactor Safety, p. 675. As is shown there, BORAX produced substantially more energy than SPERT, and SL-1 more than either, given the same initial reactor period. (This is an important reason why estimating the energy release for an excursion of a particular period at UCLA directly from the release for SPERT at the same period is so non-conservative-- the same period produced far higher excursions in other reactors.) Seemingly minute differences in metal-water ratios, temperatures and void coefficients, etc., had marked effects on total energy released.

78. This is understandable when one realizes that the process of a power excursion is essentially exponential. The nature of the exponential rise is that very minor decreases in exponential period (the "e-folding time") or increases in total time of the excursion (by delay in the shutdown mechanism) can cause the power to increase by large amounts. Thus a delay of a few milliseconds in the transfer of heat from the fuel meat to the clad and then to the coolant (caused, for example, by thicker fuel plate or lowered thermal conductivity because of corrosion or irradiation) can mean the difference between an excursion terminated safely and one resulting in melted fuel and substantial fission product release. Thus, minor errors in calculation or extrapolation can have potentially disastrous results.

79. In the absence of actual SPERT-type excursion tests with an Argonaut-type reactor, it is understandable perhaps that hazards analysts would attempt to extrapolate from the excursion tests that have been performed, albeit on reactors of different type. Thus UCLA's own 1960 Hazards Analysis, the Hawley et al review, and the Neogy memorandum all rely on the power excursion tests performed at the NRTS in Idaho. UCLA relies largely on the BORAX tests in its original analysis; Hawley et al on the SPERT ID series of tests; and Neogy on the SPERT IA series. (Surprisingly, none even touches on the SL-1 accident.) All are based on the fundamental assumption that one can extrapolate with extremely high precision from the SPERT or BORAX tests to the UCLA Argonaut.

80. We take substantial issue with such an ssumption. First of all, the SPERT tests were not intended to be used in such a fashion. SPERT was an attempt to understand the mechanisms of shutdown in power excursions, not to produce an absolute number that could be plugged into reactor analyses for significantly different kinds of reactors. In particular, it was never intended that a hazards analyst would simply look at the exponential period at which some melting was expected to begin at SPERT and say that therefore substantially different reactors could safely handle precisely the same period. The SPERT tests simply do not permit such extrapolation to different reactors without an extremely detailed accounting for differences between the reactors, which is very difficult to do, and very significant error bars to take into account the significant uncertainties in such extrapolation.

81. If the SPERT core was destroyed with a \$3.50 insertion, it would have been of considerable concern if a reactor operator used that fact as basis for a \$3.40, or \$3.00 limitation for another reactor, particularly of a different type and in an urban environment. The SPERT tests were never intended to be so used-- the uncertainties are just too large. To say, as the Hawley et al review essentially does, that the SPERT ID core indicating melting beginning around a 7 msec period meant that the UCLA Argonaut could tolerate a 7.2 msec period excursion without any melting

or release of fission products goes far beyond the purpose of the SPERT tests and the statistical significance of the data.

82. The primary value of the SPERT tests was a significant advance in the qualitative understanding of reactor behavior during power excursions and, in particular, the various components of shutdown mechanisms in differing cores-- radiolytic gas production, water-moderator expusion, fuel plate expansion, Doppler effect, density changes, "warm neutron" effects, and the final shutdown mechanism, rapid disassembly of the reactor core. It provided qualitative understandings about the nature of the phenomena, not direct quantitative data universally applicable, particularly to other reactor types.

83. With the above prefatory comments about the difficulties inherent in such extrapolations, an analysis foolows of the three attempts that have been made to extrapolate the BORAX and SPERT data to the UCLA case.

THE 1960 UCLA HAZARDS ANALYSIS AND 1980 SAFETY ANALYSIS REPORT

Melting Estimated to Occur Around 2.3% $\Delta k/k$ (The Current Licensed Limit)

84. The UCLA Argonaut-type reactor was designed for a maximum power of 10 kw_{th} and maximum excess reactivity of about 0.6% $\Delta k/k$. As indicated above, these limitations were considered prudent in light of student operators, lack of containment and dense population immediately next to the facility. The Analysis supporting the proposed license argued in particular that the 0.6% reactivity limitation was prudent because it was below that necessary for prompt criticality, above which level engineered safety features such as scram systems tend to be too slow to compensate for the rapid power growth. To demonstrate that not only was 0.6% safe, but that a sufficient safety margin existed for a training reactor, the Hazards Analysis attempted to estimate, quite roughly, the level at which melting could be expected. This was done, the Analysis indicates, to show the magnitude of the safety margin and to provide further support for the 0.6% limitation.

85. To make this showing, the Hazards Analysis relied on BORAX data. Obtaining a proportionality from those tests for temperature rise per Mw-sec of energy release, the analyst determined that it would take approximately 41 Mw-seconds of energy release to raise the temperature of the fuel plate from the temperature of boiling water to the melting point of aluminum (not of the fuel meat, which melts at a 36°F lower temperature). Using a chart obtained from the BORAX tests, it was estimated that an excursion of reciprocal period 150 sec^{-1} would give an energy release of 41 Mw-seconds plus the energy necessary to raise the plate temperature to the boiling point of water; i.e., a reciprocal period of 150 sec^{-1} would produce enough energy to raise the plate temperature to the melting point of aluminum, at the center of the hottest plate.

86. As stated in the Hazards Analysis:

"It is useful to estimate the value of excess reactivity which, if suddenly inserted and not removed by the control system, would raise the maximum temperature in the hottest fuel plate to the

melting point...

"The first step in the procedure is the <u>estimation</u> of the exponential period corresponding to the excess reactivity which <u>would</u> <u>have</u> characterized a power excursion of <u>similar</u> effect in BORAX I." (p. III/A-3, emphasis added)

87. The Analysis then attempted to correct for the different void coefficients, coolant channel width, figure of merit for fuel performance, and peak to average power ratio, concluding that the limiting excursion for UCLA is 9.1 milliseconds. Correcting for the different prompt neutron lifetimes, it was stated that that period corresponds to an insertion of 2.3% $\Delta k/k$. (It is interesting to note that the Hazards Analysis estimated that the UCLA reactor could tolerate a considerably smaller power excursion in terms of energy release than could BORAX, because of the different characteristics of the reactor-- 41 Mw-sec, plus the energy to bring the water to saturation, as the limit for BORAX, and 28 Mw-sec for UCLA. Conversely, BORAX was stated to reach its limit with a 6.7 msec period, UCLA with a 9.1. This shows the problems with assuming that if SPERT, for example, could tolerate a 7 msec period, so too would UCLA.

88. As the original Hazards Analysis calculations make clear, 2.3% $\Delta k/k$ would be sufficient to cause fuel melting at UCLA, if the assumptions employed are correct. * We have made clear above our objections to such

* There is some confusing language in the text of the Analysis on this point. The calculations make perfectly clear that, if the Analysis is correct, a 2.3% reactivity insertion will bring the hottest part of the fuel meat to the melting point of aluminum. Yet it is stated at one point that the reactor will tolerate a power excursion of at least that magnitude without melting occurring at the hottest part of the fuel. This is primarily a semantic difference, asserting that a certain estimated point is the end of the safety zone instead of saying it is the beginning of the danger zone.

Some of the confusion can be traced to the fact that UCLA copied its 1980 Safety Analysis Report (by which time the reactivity limit had been raised to 2.3%) from its 1960 Hazards Analysis (at which time the limit was 0.6%), which in turn was copied from a 1959 AMF analysis, which in turn was copied from a 1958 analysis for the University of Florida reactor. (see attachments). A comparison of the analyses indicates that while the language was copied virtually verbatim, there was a significant difference between the University of Florida reactor, upon which the original analysis was based, and the UCLA reactor. The fuel at the former was 20% enriched, 90% enriched for UCLA; the U of F fuel was 46 w/o U-Al, whereas UCLA's is right at the eutectic point, 13.4 w/o. (See page 1 of U of F and UCLA's "Estimation of Effects of Assumed Large Reactivity Additions.") The uranium-aluminum alloy in the U of F fuel meat melts considerably above the melting point of aluminum, unlike the alloy in UCLA's, which melts below the critical temperature of aluminum.

Furthermore, the U of F fuel had a Doppler contribution to shutdown, since it was LEU, whereas UCLA practically does not. The 1960 UCLA

extrapolations from one reactor type to another in the absence of empirical evidence from tests like we conducted at SPERT or very significant error bars at each point in the calculation. As we read the Hazards Analysis, this was recognized by its author, who recognized the approximations he was making required substantial margins for error. These margins were provided by the fact that the analyst was not trying to show that 2.3%, or 2.2%, or some similar number was safe, but rather that 0.6% was prudent and had a sufficient margin of safety for a training reactor. He did this by estimating, through some rather crude extrapolations, that danger might be found in the 2.3% range, and therefore limited the facility to 0.6% so there would be a margin of safety for errors in calculation or operational errors that might slightly exceed the license limits. As we read that analysis, it shows melting at around 2.3% $\Delta k/k$, and supports a 0.6% limitation. It cannot be used to justify a limit at or close to 2.3%. In fact, as will be discussed in the next section, correcting for some non-conservative values in the Hazards Analysis calculations indicates melting substantially below that level.

<u>The Hazards Analysis, When Corrected, Indicates Risks Below 2.3% $\Delta k/k$</u>

89. The Hazards Analysis makes clear that the fuel meat could reach the melting point of aluminum with a 2.3% insertion. Corrected for more conservative void coefficients and delayed neutron fraction, plus consideration of eutectic melting, indicates danger with considerably smaller insertions. Furthermore, proper consideration of error bars at each step in the calculation, as well as consideration of UCLA's positive feedback features, would reduce considerably further the estimated reactivity insertion that could be tolerated without melting, i.e. that could be successfully terminated by steam formation.

90. The Hazards Analysis uses a void coefficient for UCLA of -0.18%k/% coolant void, whereas the current application cites a value of -0.164% (p. III/6-6). If UCLA's reactor has a smaller void coefficient than initially thought, its capacity to tolerate certain excess reactivity insertions is substantially reduced, and fuel melting could thus occur at substantially less than a 2.3% $\Delta k/k$ reactivity insertion. Uncertainties in the precise void coefficient (which can vary by region of the core and other variables) add substantial reason for added margins of safety.

91. Using the Hazards Analysis calculations and merely substituting the more conservative-- although perhaps not sufficiently conservative-- void coefficient for the value used initially, before measurements had been made, results in the excursion that could cause melting being 3 Mws <u>smaller</u> than that assumed in the Hazards Analysis:

Analysis merely removed the sections of the U of F analysis dealing with the Doppler effect and other fuel characteristics, failing to correct for the differences, and keeping in language contradicted by the calculations. And thereafter UCLA used the Analysis, which had concluded 0.6% was safe and 2.3% dangerous, to support modification of reactor limits to 2.3% $\Delta k/k$. This is but one example of the problems that can occur when copying analyses performed by others or for other reactors.

using 1960 estimate for void coefficient	using 1980 estimate for void coefficient
$\frac{C_{Borax}}{C_{UCLA}} = \frac{0.24}{0.18} = 1.33$	$\frac{C_{Borax}}{C_{UCLA}} = \frac{0.24}{0.164} = 1.46$
$\frac{41 \text{ MW sec}}{1.33} = 31 \text{ MW sec}$	$\frac{41 \text{ MW sec}}{1.46} = 28 \text{ MW sec}$
31 MW sec x .82 x 1.12 = <u>28.4 MW sec</u>	28 MW sec x .82 x 1.12 = <u>25.7 MW sec</u>

92. In addition, as has been pointed out above, the Hazards Analysis calculations appear to neglect eutectic melting. The calculations were based on the melting point of aluminum, whereas the UCLA fuel meat is in the eutectic range and melts at 20°C lower temperature than aluminum. (Hawley, p.18). Thus, a smaller excursion than estimated in the Hazards Analysis would bring the fuel to melting. Using the figure of 24.4°F/MW-sec supplied in the Hazards Analysis, about 1 MW-sec less energy would be required than previously estimated, producing a commensurate reduction in the amount of excess reactivity necessary to produce fuel melting.

93. The Hazards Analysis used a non-conservative delayed neutron fraction (β) of 0.0074, whereas the Application now cites a figure of 0.0065. β is important in the conversion from period to excess reactivity through the "inhour equation." Use of the form of the inhour equation cited in the Hawley review (p. 16) shows that use of the smaller β results in a shorter exponential period for the same reactivity insertion, and thus more energy release and higher fuel temperature. * Conversely, use of the smaller β means a smaller reactivity insertion will produce the same result (i.e. fuel melting) than estimated in the Hazards Analysis employing the larger figure.

94. If the Hazards Analysis concludes that a 2.3% $\Delta k/k$ insertion will bring the hottest parts of the fuel to the melting point of aluminum-- and it clearly does-- then use of the smaller figures for void coefficient and β, as well as consideration of the eutectic melting point of the meat (below that of aluminum), would indicate fuel melting occurring with a substantially smaller reactivity insertion.

95. There are a number of other factors which should further substantially reduce the Hazards Analysis estimate of the excess reactivity necessary to induce melting -- the effect of fuel irradiation or cladding corrosion (which can reduce thermal conductivity and thus delay shutdown), as well

* The version of the inhour equation cited by Hawley is

$$T = \frac{\ell}{[\Delta k/k\,(1-\beta_{eff}) - \beta_{eff}\ ]}$$

as initial moderator temperature, to name just a few. Although the Analysis conservatively assumed the 2.3% $\Delta k/k$ insertion to occur in a subcooled reactor, the Hawley review at p. 15 rightly points out that excess reactivity is normally measured at normal operating temperatures of the reactor and that negative temperature coefficients for the water would make, for example, 2.3% at operating temperature actually much more at lower-than-normal temperature. Conversely, if 2.3% is dangerous on a cold day, far less than that amount must be installed if measurement is under warm moderator conditions.

96. And, as discussed in more detail later, the positive coefficient for the graphite can likewise mean that 2.3% $\Delta k/k$ measured when the graphite is cool can result in more than 2.3% $\Delta k/k$ being available after its temperature has risen. That factor, plus positive feedback effects in an excursion (such as the positive coefficient for the graphite, the positive void coefficient in a portion of the water moderator, and the positive effects from changes in plate and bundle spacing that might accompany the initial stages of the excursion) further dramatically reduce the "safe" level. Proper inclusion of adequate error bars for the various steps in the calculation, pushes the level even further down.

97. Thus, given the basic assumptions employed in the Hazards Analysis, and the numerical values utilized, the Analysis' calculations predict fuel melting with insertions in the range of 2.3%. When a few of the numerical values are changed to reflect more appropriate values (e.g., β, void coefficient, and eutectic melting point), substantially less than 2.3% $\Delta k/k$ would appear to be sufficient to induce melting-- if the methodological assumptions employed are correct. If other factors are included, even smaller levels are tolerable.

98. There are problems, as indicated at the outset, with extrapolating from one reactor to a different one-- to three significant figures-- without error bars. This assumes that there exists a complete knowledge of all the differences between the reactors and how those differences precisely affect behavior. As has been shown, a number of differences were not considered, and to assume that what differences are considered can be corrected for using simple linear relationships is inappropriate for the level of precision assumed. For example, the Hazards Analysis assumes a linear relationship between void coefficients and total energy release, which is unlikely to be correct, given the exponential nature of energy release in a power excursion.

99. The Hazards Analysis merely declares that the 0.6% k/k limit has a reasonable safety margin to compensate for the potential errors in extrapolating from the BORAX data. It is filled with terms describing the calculations clearly as estimates and extrapolations, based on unverified assumptions:

> <u>On the assumption that this minimum value is the true value</u>, a rise of water temperature from near 0°C to 80°C <u>would reduce</u> reactivity by 0.6% k_{eff}.
>
> III/A-2 emphasis added

> The characteristics of the UCTR which determine its behavior during power transients resulting from large reactivity additions are quite similar to, but not identical with, those of the BORAX I reactor.
>
> III/A-1 emphasis added

> Experiments of the BORAX and SPERT types have not been made with reactors having widely different neutron lifetimes. The general evidence of the experiments, however, supports the supposition that...
>
> III/A-3 emphasis added

> In comparing the behavior of different fuel plates, it must be recognized that the total energy release of the power excursion can no longer be considered as a definitive variable...
>
> III/A-4 emphasis added

> that the maximum fuel-plate temperature rise is, to within experimental error, proportional to the maximum energy release of the power excursion. The proportionality was determined to be constant 24.4°F per MW-sec.
>
> III/A-3 emphasis added

> The relative importance of the two moderators, graphite and water, in determining the effective neutron temperature introduce uncertainties in the theoretical computation of this computation.
>
> III/A-2, emphasis added

100. The text is replete with phrases about estimation, assumption, uncertainties, suppositions, and so on. The Hazards Analysis was designed to merely estimate how large a safety margin the reactor would have at the then-licensed limit of 0.6% $\Delta k/k$.

101. Substantial error bars, or margins of safety, are required in such analysis, which is why the Hazards Analysis concluded that excess reactivity at this facility should be limited to about 0.6% $\Delta k/k$. The Hazards Analysis demonstrates that currently requested levels of excess reactivity provide no margin of safety and could lead to fuel melting in the UCLA reactor.

THE HAWLEY et al REVIEW

A Very Small Margin of Safety

102. The Hawley study attempts to address the same issue as the Hazards Analysis, except that fewer corrections are made for the difference in characteristics between the UCLA reactor and the SPERT reactor, the original data source. Hawley concludes that 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), there would be only a 39°C margin of safety. Given the extremely crude approximations used, the numerous factors not considered (e.g. lower void coefficient) that would markedly increase 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.

103. The section of the Hawley, et al, report dealing with excess reactivity issues appears to consist almost exclusively of a brief literature review and some extrapolations from the SPERT I tests. Whereas the 1960 Hazards Analysis took into account a number of differences between the UCLA Argonaut and the BORAX reactor, from which it was extrapolating its data, the Hawley review does not account for several of the UCLA-SPERT differences, particularly UCLA's smaller void coefficient, which would tend, if not otherwise compensated, to suggest that an excursion of the same period in SPERT and the Argonaut would produce greater energy release at UCLA. The Hawley report's primary consideration of differences between the two reactors consists of correcting for the longer neutron lifetime at UCLA, a factor which is helpful to UCLA.

104. The Hawley approach was extremely simple-- calculate the period produced by an insertion of available excess reactivity, estimate the energy release an excursion of similar period would have produced at SPERT ID, and then scale temperature linearly to the peak temperature estimate during the SPERT ID destruct test.

105. And yet, even without taking into account factors such as void coefficient differences, which would tend to produce higher temperatures, the analysis estimates peak fuel temperatures only about 50° below the melting temperature. No error bars whatsoever are provided for the extrapolation steps nor for the final conclusion. (There appears to be a subtraction error in that Hawley et al assert on page 19 of their report that a hot spot of 586°C would be 74°C below the melting point of the fuel meat, which they cite on the previous page as being 640°C.)

106. 50° is not an adequate margin of safety, particularly when so many of the differences between SPERT and the UCLA Argonaut were not taken into account. Furthermore, significant effects may appear just below the melting point, such as volumetric expansion of the fuel resulting in cladding failure, or considerably increased diffusion of fission products through the hot metal. It was noted at SPERT, for example, that some of the fuel plates were very substantially softened and warped, even though not truly melted, and that they would stay in that softened form for several days thereafter, behaving something like a wet noodle. This was prior to the final destruct test.

107. So even if Hawley et al were correct in their estimate of peak temperatures 50 or so degrees below the melting temperature, there would still be concerns. However, questionable assumptions used by Hawley et al suggest far greater temperatures could be achieved in the UCLA Argonaut than those estimated.

Questionable Assumptions

108. Perhaps the most questionable assumption is that a 7.2 msec period would produce a 12 MW-sec energy release in the UCLA Argonaut. Given the linear scaling assumption of temperature to energy release employed by Hawley (p. 19: 1500°C per 30.7 MW-sec, or about 49°C/MW-sec), a 13 MW-sec energy release would cause melting, if Hawley's assumptions are accepted. That is not much of a margin of safety if his 12 MW-sec estimate is correct.

109. 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-sec of energy, getting within a few degrees of melting, from a 2.6% insertion (supposedly the equivalent of 2.3% on a cold day). However, it is noted that the 1960 Hazards Analysis estimates a considerably longer period than the one assumed by Hawley (9.1 instead of 7.2 msec) will produce an energy release of 28.4 MW-sec, plus the energy necessary to raise the fuel to the boiling point of water. How a longer period is estimated to produce $2\frac{1}{2}$ times the energy assumed in the Hawley report is not explained. The 1961 ATL analysis is reported as indicating a far smaller insertion than that assumed by Hawley, 1.5% $\Delta k/k$, will produce an energy release of 24 MW-sec, double that assumed by Hawley. The GNEC report assumes the same period, around 7 msec, producing 32 MW-sec, plus about 4 MW-sec to raise the temperature to the saturation point of water (i.e., about 36 MW-sec total). The Jason reactors are referred to as estimating 10 MW-sec, 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-sec estimate of energy release whereas other estimates several times higher exist, noting that less than 1 MW-sec 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).

110. Thus, were Hawley right that a 13 MW-sec excursion could cause fuel melting, and were the estimates of energy release from any of these other studies correct, fuel temperatures would be considerably over the melting temperature, not, as Hawley asserts, just under. The ATL estimate of 24 MW-sec for a 1.5% insertion would be over the melting threshhold. The 1960 Hazards Analysis estimate of 28.4+ MW-sec would be over. So would the GNEC estimate. So would the Jason estimate.

111. Empirical data from actual excursions also underscore the non-conservative nature of the Hawley assumptions. For example, a 7 msec period in the SPERT IA core is reported to have released 23 MW-sec of energy, nearly twice that assumed by Hawley based on SPERT ID data (Schroeder, 1957). The plot of period versus energy release (Thompson and Beckerley, 1964), mentioned earlier, likewise shows how the choice of 12 MW-sec for a 7.2 msec period is quite non-conservative. SL-1 extrapolations, for example, would suggest an energy release five times greater than that assumed by Hawley. When one recalls that an energy release of 13 MW-sec would cause melting, if Hawley's other assumptions are correct, then data suggesting releases of 23, 28+, and even 60 MW-sec of energy from a 7.2 msec period excursion, not the 12 MW-sec assumed by Hawley, indicate an unacceptable probability of a destructive power excursion, one that could release significant amounts of fission products.

112. Hawley's chief error is in equation (2) on page 17, where he assumes that the total energy release from an excursion in the UCLA reactor can be precisely determined by doubling the ratio of the reciprocal period to a reactivity coefficient found from excursions during the SPERT ID series of tests. He assumes that he can apply, without any modification, that reactivity coefficient (which was substantially different than the reactivity coefficient found in the SPERT IA tests, the BORAX tests, or from the SL-1 accident) directly to the UCLA case.

113. Hawley commits this error by focusing on those factors affecting the first part of a power excursion, the power rise, but ignoring the second part of the excursion, its termination by self-shutdown features. He assumes that total energy release for an excursion at the UCLA reactor is entirely controlled by the exponential period of the rise. However, total energy release (which determines the severity of the incident) is tightly controlled, not just by how fast the power rises, but by how quickly the power rise is terminated. A slow power rise in one reactor may cause far more damage than a fast rise in another, <u>if the shutdown mechanisms in the former are slow as well</u>.

114. As discussed earlier, 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 below (the same as in 35 above), 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:

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

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.

115. 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 mechanisms can be devastating.

116. Assume an exponential period T of 7 msec and a time interval of rising power before the shutdown mechanism acts of .07 sec (i.e., 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 raised to the .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.

117. 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-sec 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.

118. 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% reduction in thermal conductivity in the Al-U fuel meat caused by irradiation 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. Based on the analyses done to date, insertion of either \$3.00 or \$3.54 must be considered a credible cause of fuel melting.

119. It should be noted once again, however, that the methodology of very simplified extrapolation from SPERT or BORAX data to the UCLA Argonaut case, as done in the Hawley report, seems most inappropriate given the differences in the reactors and the difficulties in correcting for those differences. The SL-1 accident, which took the lives of the only people nearby at the time, was "non-credible" in Hawley's terms, yet it happened. It released several <u>times</u> more energy than Hawley's extrapolations from SPERT ID would predict, even though it was much more similar to SPERT than is the UCLA Argonaut. * The Hawley extrapolations cannot be relied upon to prevent an SL-1 type accident at UCLA, one that would occur not in a remote federal testing station but in the midst of tens of thousands of people.

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* 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, its validity has not been demonstrated.

However, assuming for the moment that it 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

THE NEOGY MEMORANDUM

120. The Memorandum provides very little information on the methodology employed, primarily reciting the conclusion reached. The following points can be readily made from what information is provided: the choice of a relatively slow ramp insertion is most unrealistic, the use of clad temperature instead of peak meat temperature is non-conservative, the utilization of a computer code designed to model Loss of Coolant Accidents (LOCAs) and other transients for BWRs and PWRs for analysis of reactivity accidents in small research reactors seems of unproven validity, and the use of an adjusted "lambda" seems little more than a "fudge factor."

121. Neogy is said to have "qualified" the RETRAN program for assessing Argonaut research reactor power excursions, a purpose apparently not intended in the original program, by comparing the predicted power trace with an actual measured power trace from a single SPERT IA excursion. The two did not match, so a fudge factor "lambda" was added, to adjust the predicted estimates to the actual data. The comparison of predicted versus actual data from SPERT was apparently only done for the one 15.8 msec SPERT transient, where adjustment with "lambda" was found to be necessary. No checking of the program, once modified by "lambda," against other SPERT IA transients is reported, let alone against SPERT ID, BORAX I, or SL-1 transients.

122. Certain non-conservative assumptions appear to have been used in addition. For example, values such as UCLA's void coefficient, prompt neutron lifetime and delayed neutron fraction are all larger than values reported elsewhere.

123. Furthermore, the very premise of the analysis-- the assumption of a relatively slow ramp insertion-- is unreasonable. A person manually pulling a control rod, as in the SL-1 case, or withdrawing a neutron-absorbing sample from an irradiation port, could insert reactivity very much faster than the ramp rate assumed in the Neogy memorandum. The assumption, then, that the $3.00 insertion would produce an excursion of relatively long period (15.8 msec) is inappropriate. The energy release and temperature estimates that follow therefrom are thus substantially too low. Correction of these assumptions, and consideration of the positive feedback features, void the conclusion that melting will not occur.

124. Again one must emphasize that extrapolations from SPERT to the UCLA Argonaut are fraught with peril. But if one is to make such extrapolations, they should be done with a significant element of conservatism. The analyses done to date have lacked sufficient conservatism and have made a number of other errors. Rather than indicating that the UCLA facility

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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. Thus, if Mr Ostrander were correct in his explanation of the SPERT/BORAX differences, the situation for UCLA might be even less favorable than either.

is inherently safe with its present or proposed excess reactivity loading, each suggests, upon careful reading, the opposite.

125. There are really only two ways to find out for sure whether fuel melting can occur with the assumed excess reactivity insertions. One is to do a SPERT-type series of excursion tests at a remote location with an Argonaut core very similar to UCLA's. The other way is for an accidental power excursion to occur at UCLA itself. To relicense the UCLA Argonaut as is would be to risk the latter form of uncontrolled research.

RELATED OBSERVATIONS

126. There are numerous mechanisms for initiating a power excursion at the UCLA Argonaut-type reactor. There are basically two categories or ways of initiating the event: insertion of positive reactivity and removal of negative reactivity.

127. The original Hazards Analysis recognized one such scenario:

> One procedure to achieve maximum excess reactivity in the reactor would be to insert into the reactor a sample with sufficient absorption to prevent start-up. When the controls were fully withdrawn and criticality was not achieved, the maximum reactivity would be added if the sample were removed without reinserting the control blades. (p.60)

128. Thus, if a large negative worth sample were inserted for irradiation (either in the enlarged irradiation ports or through the added-on pneumatic "rabbit" system) and the sample was removed without the reactor operator remembering to first reinsert the control blades, a power excursion could result. Having to rely upon the reactor operator to follow correct procedure, particularly with student operators learning at the controls, is precisely the opposite of the basic premise of an educational reactor--inherent safety, a "forgiving" fail-safe machine, such that the worst mistake possible cannot cause injury. (A potential precursor of such an accident is suggested in the attached November 16, 1981, notice of violation from UCLA.)

129. Substances of large reactivity worth, negative or positive, can be inserted in the reactor, through the pneumatic tube system, the irradiation ports, or through other means. There are a number of substances that are highly absorbing and would represent significant negative reactivity worth. If large negative worth samples were neither possible nor anticipated to be needed, then UCLA would have had no need to increase its excess reactivity level from 0.6% 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, at which other Argonaut reactors operate and at which this one did for some time.

130. Just as removal of a large negative sample from the core region, without a compensating prior insertion of control blades, can result in the equivalent of a large positive reactivity insertion, initiating a power excursion, so too can insertion of material of positive reactivity. Fissionable materials and perhaps some good moderating materials could have substantial positive worth. Rapid removal of the negative materials or rapid insertion of the positive materials would have the same effect-- a potentially large

reactivity insertion. For example, UCLA at one point requested 250 grams of U-235 for irradiation in the reactor's thermal column. If such material were instead placed in an irradiation port, a very sizeable positive insertion could result. That such material (or perhaps an unexpectedly good moderator) could be inserted in an irradiation port--as a prank, by mistake, or as an inadequately reviewed experiment--could certainly occur, particularly if there had been a history of weak administrative controls at the facility.

131. There are numerous other mechanisms for accidentally initiating a power excursion. For example, the facility has had repeated problems with control blades becoming stuck. The method of trying to free them is to try to torque them free with a hand wrench applied to the drive mechanism, which is located outside the reactor shield and readily accessible (See photos to be filed at a later date in compliance with protective order). While normal insertion rate of reactivity with the control blades should be limited by the motor (if proper speed motors are used), that would not be the case were the blades to be manipulated manually, as in an effort to free them or otherwise to do maintenance on them. (It should be noted that the SL-1 accident occurred during such maintenance to the control rod drive mechanism and that a history of sticking control rods, necessitating torquing with a hand wrench, had preceeded the accident.)

132. Other mechanisms of insertion involve water level variations. Should the water level in the core drop for one reason or another, and the reactor be kept critical by further withdrawal of control blades, a sudden rush of water (particularly, cold water) into the core could result in the equivalent of a substantial positive reactivity insertion. This could occur during experiments which vary core water level, or through partial failure of the dump valve due to loss of full air pressure which holds it in place. The latter would cause some water level drop, which could rapidly be reversed were a surge of air pressure to fully close again the dump valve. Violations of excess reactivity restrictions during core water level experiments, or problems with air pressure to the dump valve, could thus have serious safety implications. (See 1978 Annual Report, p.3; and Radiation Use Committee Minutes, December 22, 1977, p.4, attached.)

133. Some event which induces some coolant boiling could also result in positive reactivity insertion. If coolant channels were partially restricted, or coolant flow or heat removal slowed, or power slightly overshot, localized boiling might occur, reducing moderator density and requiring further withdrawal of controlblades to keep the reactor critical. A sudden fluctuation altering the amount of boiling could result in an insertion of positive reactivity because of the negative void coefficient in the central region or the positive coefficient in the higher region.

Multiple Failure Modes

134. A number of unfortunate features of the UCLA Argonaut-type reactor create potentials for multiple and common mode failures. For example, the reactivity change occasioned by small shifting of reactor bundles within the fuel boxes (changing slightly the gap between the bundles)

could result in positive reactivity being added in the midst of an excursion which might not, of itself, be sufficient to cause melting. Similarly, expansion or bowing effects that increased the plate spacing could push an excursion "over the top," as could the initially positive effect noted upon dropping the water.

135. There are numerous other possibilities as well. One entails a power excursion not sufficient to cause melting by itself but which does involve expulsion of the water moderator. It was noted with the SPERT reactors that such expulsion would on occasion lead to repeated criticality as the expelled water condensed and dropped back into the core. An excursion limited by moderator expulsion, as at SPERT or BORAX, can send a plume of water and steam high in the air. When that water returns, it does so at a significant velocity, which amounts to a very rapid insertion of substantial excess reactivity. Such behavior is called "chugging", and on several occasions incidents occurred in which the initial reactivity insertion was not sufficient to cause damage, but the repeated excursions caused by repeated reintroduction of the moderator after expulsion caused increasingly larger excursions which, had the event not been terminated through scramming the reactor, might have essentially torn the reactor apart. (A history of sticking control blades which could make final termination of such a series of excursions impossible would thus have safety significance. Similarly, the lack of deflector plates descrived in the original Hazards Analysis as designed to prevent such repeated excursions by preventing expelled water from returning to the core, means that an important safety feature is missing.)

136. The positive temperature coefficient for the graphite is troubling as well. A research reactor used by students needs to be inherently safe. Inherent safety necessitates large negative temperature and void coefficients. Any positive coefficients (which are thereby autocatalytic) are to be strongly avoided. This is especially true when the value attributed to the positive temperature coefficient for the graphite ($+0.006\% \Delta k/k/^{o}F$) is larger than the negative temperature coefficient cited for the water ($-0.0048\% \Delta k/k/^{o}F$).

137. During a power excursion the positive temperature coefficient of the graphite could provide a feature which makes the excursion more destructive than would otherwise be the case. A portion of the energy liberated in a power excursion is given off as prompt neutron and gamma radiation, resulting in a prompt temperature rise in the graphite and other surrounding materials bombarded by that radiation. Even a few degree rise in the graphite temperature would mean the addition of positive reactivity at a time when negative reactivity is needed to limit the power excursion. The addition of even relatively small amounts of positive reactivity can produce a slight delay in the shutdown mechanism taking hold; because of the exponential nature of the excursion, even a milli-second additional delay can be significant. Given the extremely small margins of safety, e.g., Hawley's $40\text{-}50^{o}$, even assuming all the assumptions made are correct and the absence of other uncertainties, a slight addition of positive reactivity during the excursion can cause a small margin of safety to become far smaller.

138. Hawley (p. 15) has pointed out that excess reactivity in Argonaut-type reactors is usually measured under normal operating conditions and that the negative temperature coefficient of the water thus makes it possible that a reactor with a measured level of excess reactivity of, say,

$3.00, will at times of cold coolant have considerably more than $3.00 of excess reactivity available. The same is true in reverse for the positive coefficient for the graphite.

139. Graphite temperatures rise significantly after an extended run of several hours in the Argonaut. Plots of reactivity versus time and temperature [6] indicate a rise of approximately 100°F in 2 hours, to a temperature significantly above the temperature of the water coolant, apparently because the water's heat is continually extracted by the reactor's heat removal system for the coolant and because much of the graphite temperature rise is due to the cumulative effect of heating by radiation from the fuel. Coolant temperature levels off rapidly after start-up and then remains constant; graphite temperature is shown to steadily and continually rise.

140. Thus, if excess reactivity of, say, $3.00, was measured near the beginning of a run, or during a short run, when the water was warm but the graphite temperature rise not yet anywhere near its maximum level after a long run, that $3.00 could actually be the equivalent of substantially more at the end of such a long run, where the coolant temperature would be the same as at the time of the measurement but the graphite, with its positive coefficient for temperature, would be substantially warmer.

141. The positive temperature coefficient has been reported as approximately $+0.006\%\Delta k/k/^{\circ}F$ (AEC inspection report 50-142/68-1, p.6, attached). A temperature rise of 100°F in the graphite, as normally observed after a two hour run, could thus mean an <u>increase</u> in reactivity of $0.6\%\Delta k/k$, or nearly a dollar. A reactor, thus, that was thought to be limited to $3.00 could at times have available $4.00 because of the positive temperature coefficient. Conversely, to keep to a licensed limit of $3.00, it would be necessary to have a <u>measured</u> maximum of around $2.00, if these figures are correct.

142. There can, furthermore, be occasions when the positive graphite coefficient and negative water coefficient interact in such a fashion as to produce a greater reactivity addition than can either coefficient acting alone. Because heat is extracted so much faster from the water coolant than from the graphite moderator/reflector, temperature can drop more slowly in the graphite than the water after shutdown, particularly if the reactor coolant system remains functioning after the control blades are reinserted. Thus, after an hour shutdown or so, the reactor might have water substantially cooler and graphite still substantially hotter than the conditions at which the $3.00 limiting value of excess reactivity was measured. One could then have far more than $3.00 available because of the hotter-than-normal graphite, and <u>additional</u> reactivity on top of that because of the cooler-than-normal water. This, in fact, may be the normal reactivity situation of the reactor a few hours after shutdown from a few-hour run. These reactivity coefficients then would necessitate limiting the measured value of reactivity to less than $2.00 in order to ensure that no more than $3.00 is ever available. We have indicated elsewhere that $3.00 itself is dangerously excessive.

143. Thus, factors such as those discussed above could mean that the reactor, at the time when reactivity was measured, was below the licensed limit, but at other times, due to positive reactivity effects, was above. Furthermore, the existence of an excess reactivity limit in the license does not mean that that limit will not otherwise be exceeded, unless the reactor's inherent design does not permit any more reactivity than that level, which is not the case with the UCLA Argonaut. Reactivity is controlled by the amount of the fuel and the effectiveness of the moderator, both of which are easily modified in the Argonaut. (Note the large quantity of heavy water stored next to the UCLA reactor, for example.) UCLA has a substantial quantity of spare fuel on site; reactivity is readily added by removing dummy fuel plates from the core and replacing with actual ones. This is how refueling is done to compensate for burnup and other factors.

144. Therefore, the fact that the Technical Specifications may contain a limitation of $3.54, or $3.00, on excess reactivity does not mean that that limit will not be overshot from time to time, given errors in measurement or violations of Technical Specifications. A history of measurement errors or Tech Spec violations at such a facility would substantially increase the probability of excess reactivity limitations.

145. Interactions may potentially occur between power excursion accidents and accidents of other types. For example, various core disruption events could cause or contribute to positive reactivity insertions. Flooding, be it by pipe break or other event, could add moderation (because of the dangerously undermoderated nature of the reactor) and thus cause a positive excursion. Core-crushing could move the core to more of an optimal arrangement for moderation. Seismic jolt could cause a negative sample, or a control blade, to move out of the core region. An event which caused the fuel bundle spacing or fuel plate spacing to alter could likewise contribute positive reactivity. A small seismically-induced excursion, not sufficient in itself to cause melting, could increase the maximum fuel temperature reached in a crushed core. (These will be discussed in more detail in the panel on core disruption.)

146. Fire could likewise cause some positive reactivity effects. Were the low-melting cadmium control blades to melt out of the core region, a positive reactivity effect could be observed. Were the graphite to heat up substantially, the positive coefficient could add reactivity to the core. Were firefighters to use water (or perhaps other moderating substances) to fight the fire, again a positive insertion might result. A power excursion could provide the initial heat necessary to start such a fire. These matters will be discussed in more detail in the chemical reaction panel, as will be the steam explosions and explosive metal-water reactions which have accompanied several destructive power excursions such as SL-1, BORAX and SPERT. And a power excursion of insufficient magnitude to melt the fuel by itself may be sufficient to trigger Wigner energy release, which could add sufficient energy to either melt the fuel or ignite parts of the core.

CONCLUSION

147. The UCLA Argonaut, in its current configuration, is not inherently safe. Because of the large amount of excess reactivity, and features by

which reactivity can be added, its safety is dependent upon the proper functioning of engineered safety features, strict adherence to proper procedures, absence of operator errors, thorough and careful calibration and maintenance of the equipment, adequate funds and attention devoted to keeping the facility in good condition, strong managerial and administrative controls, adherence to regulations and technical specifications, and perhaps most importantly, a healthy respect for the danger to the public that could result from an accident. A belief that no operator error, equipment failure, or other event could possibly cause an accident such as a destructive power excursion would greatly increase the probability of such an accident occuring. So would failures of the Radiation Use Committee to adequately review proposed experiments or new procedures. So would a pattern of violation of regulations and technical specifications, as would a pattern of operational unreliability evidenced by repeated unintended scrams caused by equipment malfunctions or operational errors, or, more worrisome, causes that could not be determined. Failure to calibrate adequately devices which activate scram systems, malfunction of such devices, stuck recording pens that lead to reactivity increases, permitting unlicensed operators like high school visitors to operate the reactor controls--these would all have safety significance.

148. The UCLA Argonaut-type reactor is not inherently protected against destructive power excursions of the type which destroyed the SPERT, BORAX, and SL-1 cores. The primary inherent self-limiting feature of the UCLA reactor (a relatively slow and small voiding effect in one of the reactor's two moderators) is far less effective than that of other university reactors and insufficient to prevent a serious power excursion from damaging fuel.

149. Safety margins, which have eroded over the years, are unacceptably small. The current and proposed licensed limits on excess reactivity appear capable of causing core melting in a power excursion. Even were the licensed limit of core reactivity substantially reduced, such as back to the design level, mechanisms for insertion of larger-than-licensed amounts would still remain (insertion of large positive worth samples, core flooding or crushing, positive moderator effects, violation of the Technical Specifications, etc.).

150. The effect of a serious power excursion on fuel elements would obviously be quite severe. One might expect substantial fission product release, perhaps 25% of the radioiodines and a significant fraction of other isotopes. The consequences of damage to the fuel from a power excursion would be considerably greater than those arising from damaging one of the reactor's 24 fuel bundles during a fuel handling accident, the Hawley maximum credible accident, which assumes a 2.7% release of the gaseous fission products from the dropped bundle. An accident involving dropping a single fuel bundle is not the maximum credible accident at the UCLA reactor.

151. The type of accident which destroyed the SL-1 reactor could happen on the UCLA campus. The UCLA Argonaut is not only not inherently safe, but it is not so by a wide margin.

Footnotes in the text

1. from Proceedings of the International Conference on the Peaceful Uses of Atomic Energy, United Nations, New York, 1956. Vol. 13, p. 79-87

2. Source: J.R. Dietrich, "Experimental Determinations of the Self-Regulation and Safety of Operating Water-Moderated Reactors", in the same volume as preceeding source, p. 89, 99

3. Source: Miller, Sola, McCardell, "Report of the SPERT I Destructive Test Program on an Aluminum, Plate-type, Water-Moderated Reactor", IDO-16883, June 1964

4. See inspection report 68-01, attached as Exhibit C-I-6, page 6.

5. See AEC Inspection Report dated March 1, 1962, p. 4, attached as Exhibit C-I-1

   See also Radiation Use Committee Minutes, 9/2/81, p. 6-7, attached as Exhibit C-I-2

6. see Exhibit A in UCLA's November 9, 1981, interrogatory answers to CBG

REACTOR RUNAWAY

Exhibit List

Exhibit Number	Description
C-I-1	AEC Inspection Report, 3/1/62
C-I-2	Minutes of Radiation Use Committee, 9/2/81
C-I-3	Notice of Violation, UCLA to NRC, 11/16/81
C-I-4	1978 UCLA Annual Report
C-I-5	Minutes of Radiation Use Committee, 12/22/77
C-I-6	AEC Inspection Report No. 68-01
C-I-7	Excerpts from 1980 UCLA SAR
C-I-8	Excerpts from 1960 UCLA Hazards Analysis
C-I-9	Excerpts from 1959 AMF Hazards Summary Report
C-I-10	Excerpts from 1958 University of Florida Hazards Summary Report
C-I-11	Full text, 1960 UCLA Hazards Analysis
C-I-12	film/videotape, BORAX I destruct test
C-I-13	film/videotape, SPERT I destruct test
C-I-14	photos taken within the Nuclear Energy Lab

P. A. Morris, Assistant Director for
Reactors, Division of Compliance, Headquarters
THRU : R. B. Engelken, Inspection Specialist (Reactors)
R. T. Dodds, Inspection Specialist (Reactors)
Region V, Division of Compliance

MAR 1 1962

ORIGINAL SIGNED B
R. T. DODDS

UNIVERSITY OF CALIFORNIA AT LOS ANGELES (UCLA)
TRAINING REACTOR

DOCKET NO. 50-142

SYMBOL: CO:V:RTD

Attached is our report describing a recent visit to the subject facility. The principal results of the visit were discussed with Mr. N. Klug of DL&R on January 30, 1961.

The report describes two items that appear to constitute violations of the reactor operating license. Specifically, we refer to the following:

1. During the Spring of 1961, the licensee conducted experiments in the reactor facility that at that time were not specifically authorized by the license. Until the issuance by DL&R of Amendment No. 2 on June 28, 1961, thelicensee was only authorized to perform experiments outlined in Section III of the Hazards Analysis - Final Report. These experiments are very restrictive in that they consist of: Preoperational Testing; Initial Loading Procedure; Calibration and Shakedown; Routine Start-up and Operation and Fuel Handling. The licensee is now authorized by the broad scope of Amendment No. 2 to conduct additional experiments including those that were performed prior to the issuance of the amendment.

2. During a reactor shielding survey in April 1961, the reactor was operated between 20 and 25 Kw. Licensed power is 10 Kw. Apparently, this was done without the knowledge or consent of the Reactor Supervising Engineer. The overpower operation appeared to be an error in judgement on the part of the reactor operator. It is our opinion that this could have been prevented had the scram settings for the safety amplifier circuits been set to prevent operator errors in judgement of this magnitude. In the UCLA Hazards Analysis, Section III-C, Calibration and Shakedown, it states that, "The safety circuits will be recalibrated and their trip points adjusted to 150% of normal power." As an additional requirement, we

two
pages

feel that the licensee should set these trips so the reactor will be scrammed at a maximum of 150% of licensed power (i.e. 15 Kw) and not just 150% (full scale) of the amplifier instrument reading. The overpower operation occurred one month after the reactor reached maximum licensed power of 10 Kw. We would like to emphasize the fact that the only high flux trips on the reactor are those in the two safety amplifiers.

We were favorably impressed with the experimental programs that were being conducted at the reactor facility. It was gratifying to see a truly active reactor program at a university. The present reactor staff appears to be very competent.

Enclosure:
Inspection Report
Original and 1 copy

D R A F T

U. S. ATOMIC ENERGY COMMISSION
REGION V
DIVISION OF COMPLIANCE

TO : P. A. Morris, Assistant Director for Reactors, Division of Compliance, Headquarters DATE: MAR 1 1962

FROM : R. T. Dodds, Inspection Specialist (Reactors) Region V, Division of Compliance

SUBJECT: UNIVERSITY OF CALIFORNIA AT LOS ANGELES (UCLA)TTRAINING REACTOR DOCKET NO. 50-142

SUMMARY

The University of California at Los Angeles, California, was visited on January 25 and 26, 1962, for the purpose of inspecting the University Argonaut-type 10 Kw training reactor.

1. A recirculating system has been installed for the shield tank water.

2. Reactor room structure and usage was observed to be different than that described in the original application.

3. Start-up and operation of the reactor at 10 Kw was observed.

4. Water level variation experiments indicate that when the reactor water is dumped, the reactor power initially increases before descreasing.

5. Reactor experiments were conducted in the Spring of 1961 prior to receipt of authorization for these experiments. Amendment No. 2, which authorizes these and other experiments, was issued by DL&R on June 28, 1961.

6. During a reactor shielding survey in April 1961, the reactor was operated at apparent powers of up to 25 Kw. Licensed power levels are limited to 10 Kw.

7. The Reactor Hazards Committee appears to be active.

8. Results of a shielding survey during 10 Kw operation indicate the existance of gamma plus neutron radiation levels of up to approximately 275 mrem/hr on the reactor top.

9. The reactor shield tank walls were cleaned by lowering a "frogman" into the tank. Personnel radiation exposure was below the sensitivity of the film badges worn by the individual who performed the operation.

Reviewed by: ______________________
R. H. Engelken, Region V,
Division of Compliance

II. Results of Visit (Continued)

temperature, i.e.: approximately 70°F.

MacLain said that on January 15, 1962, aluminum rod spacers were installed between the fuel elements to keep them from shifting. He said that the spacers were placed so the fuel elements would be in their most reactive positions.

3. At the request of the inspector, the reactor power was raised to 10 Kw. After the system had stabilized, the reactor water in and out temperature differential was noted to be 7°F at a flow rate of 10 gpm. MacLain said that under equilibrium conditions heat balance studies and demonstrated that 7 F was equivalent to a thermal power of 10 Kw. It was observed that the Log-N recorder read 100, the #1 level recorder 97%, and the #2 level recorder 88%. The power was held at this level until completion of a radiation survey of the facility. The results of this survey are reported under report section F, Radiation Monitoring.

4. MacLain said that experiments conducted under the license amendment for Water Level Variations had demonstrated that, if the reactor water is dumped and the reactor not scrammed, initially the reactor power will rise approximately 15% and then decrease. He said this was due to the over-moderation of reflected neutrons above the core. The normal water height above the core is approximately 12 inches. A four inch lead shield is located above the water and on top of the lead is a graphite reflector. The neutrons, traveling through 12 inches of water, become over moderated. As the water level is lowered, the over moderation decreases until ideal moderation is reached when the water height is between 2 and 3 inches above the core. MacLain said that the net effect is worth approximately 0.05% delta K/K.

MacLain demonstrated this effect to the inspector. The reactor power was lowered from 10 Kw to 8 watts. As permitted by the amendment authorizing the Water Level Variations experiment, the core water level, primary coolant pump, and water flow safety interlocks were bypassed. The dump valve was opened and the water level rapidly dropped. The Log-N recorder initially rose from 0.08 to 0.1 and the linear level recorder rose from 8% to 12% before power rapidly decreased.

The water level variation experiment was discussed during a recent phone conversation with Dr. Babb of the University of Washington. He said that they had conducted a similar experiment on their Argonaut-type reactor and had determined, by holding the reactor power constant with the regulating blade as the water level was slowly lowered, that the loss of water level from 8 inches above the core to the top of the fuel was worth an estimated +0.12% delta K/K. He stated that dropping the water level the next 4 inches below the top of the fuel decreased reactivity by 0.53% delta K/K.

5. MacLain siad that the minumum magnet holding current was 90 - 110 milliamps. The normal magnet current during operation was set at 150 milliamps. He said that the blade drop time measurements have demonstrated a total drop time for each blade of approximately 300 milliseconds. The time was measured from the time of actuation of a scram signal until the blade seat light was activated. He said that the blade drive system has had trouble-free operation.

6. Reactor Instrumentation. Horner said the instrumentation had been relatively trouble-free. Poor power regulation for the compensated ion chambers had resulted in some spurious scrams. To correct this situation, a John Fluke regulated power supply was installed in April, 1961. Occasionally the period meter had indicated spurious periods. This condition has been corrected by cleaning the commutator of the generator for the instrument AC supply.

Horner pointed out to the inspector that the Log-N amplifier is kept in the "Hi Calibrate" position when the reactor is not in use. Trouble had been experienced with the instruments drifting and getting out of calibration during periods of low power operation. He said that

MINUTES OF THE R U C MEETING
OF 2 SEPTEMBER 1981

TO: Members of the Radiation Use Committee & Guests

FROM: Anthony Zane, Secretary
Radiation Use Committee
2567 Boelter Hall, Nuclear Energy Lab

MEMBERS PRESENT:	GUESTS:
I. CATTON	J. ALBERT/INTERNAL AUDIT/UCLA
R. CONN	C. ASHBAUGH/NEL
J. GARRICK	N. OSTRANDER/NEL
J. KAUFMANN	R. REYES/HP/ROS
W. WEGST	K. SIME/NEL
A. ZANE	G. SMITH/INTERNAL AUDIT/UCLA

Mr. Zane noted that a quorum was present and called the meeting to order.

No old business was discussed as the general status of unfinished business had not changed appreciably.

The annual report was the first item of business. Mr. Zane mentioned that one scram was not recorded in the annual report. The auditors found the recording error during the audit after the report was written. That scram will be added to the report.

Dr. Wegst commented that the statement regarding prudent scheduling for an increase in port operating hours with a concurrent decrease in actual operating time provided a good example of ALARA if anyone wanted to know what kind of an ALARA program we practice.

Dr. Kaufmann noted that Table 3 on page 16, film badges 203 and 265, should have the beta symbol appended to their values under the 4th quarter and total columns to be consistent with the rest of their readings as it was beta that was read. Mr. Ostrander commented that as far as we know, gamma exposure of those two badges has never been definitively observed.

Dr. Garrick inquired about unscheduled shutdowns (scrams) for which the cause is unknown, and asked what we do about those. In particular, he inquired about the scram where rods number 1 and number 4 dropped without apparent reason.

Mr. Zane stated that we looked into the causes but couldn't find any cause. Later, during an N.R.C. inspection, one of the inspectors suggested we put a megger on the magnets to test for short circuits. This was done and all magnets were clear of shorts up to 600 volts. The cause remains unknown.

Dr. Garrick noted that this was a 1980 report, and wondered if any new developments had shed additional light on the cause. Mr. Zane replied that nothing new had evolved.

Analyses. Mr. Zane replied that the Technical Specification applied to all new experiments.

The first experiment was that submitted by Malcom W. Ewell of the California Institute of Technology, ESA number 81-17. Mr. Zane said that the novelty of this experiment was the use of an electrical heater in the thermal column which entails a shielding re-arrangement to bring the heater wires from the thermal column cavity. The main apparent question is the slight streaming of the radiation through the crack left in the thermal column entrance as the inner block will have to go where the outer block fits and the outer block will have to back the other on the outside of the shield. He said that the power had been changed from the indicated 10 kw to 500 watts to produce the desired fluence of ten to the eleventh. Professor Conn remarked that the results of this kind of experiment are quite insensitive to the rate at which the fluence is accumulated. Mr. Ostrander asked if the chamber was pressurized. "Under vacuum," said Mr. Zane. Mr. Zane described the chamber as aluminum with a top cap of aluminum and a bottom cap of lucite. He stated that the chamber contained a minute quantity of U-235 in the form of fission foils and that the amount of fission heat given off would be in the order of 10 microwatts and that the heater power was 3 amps at some low voltage so that the electric power would probably be in the order of 15 watts. He suggested that consideration be given to the partially open thermal column. Professor Catton wanted to know in which direction the streaming would be. "Toward the east and Tokomak facility," responded Mr. Zane. Professor Catton remarked that this was toward the thick concrete wall.

Mr. Zane then described a 1980 event when some film badges were irradiated in the open thermal column at various reactor power levels. At 700 watts, the reactor scrammed on a secondary radiation alarm, possibly by a neutron interaction with the scintillator. The area monitors were reading approximately 0.5 mR/hour at the time. He said that with the partial shielding there really should be no similar problem, but the health physicist should monitor the operation, attend the student in the reactor room, and that all personnel in the high bay will be badged. The committee felt that this experiment represented no hazard to the reactor, personnel, or the general public, and signed the ESA.

The next experiment was 81-18. Zane explained that there really was no evident hazard associated with this experiment, but it was new, and committee review was required. Mr. Ostrander described the experiment as trying to create a positive reactivity sample by inserting a good moderator into a void space and therefore chose to insert polyethylene into the center vertical hole. He stated that since no one can remember ever seeing a sample exhibiting positive reactivity, this is an item of curiosity. Ms. Sime actually drew up the ESA and the reactivity of the polyehtylene is to be measured at 1 watt using 2-inch increments of polyethylene.

The committee approved the experiment [But see the amended version suggested below by Mr. Ashbaugh and approved by the committee. AZ].

Dr. Garrick asked if there was any way that one could get a positive void coefficient, as some of the old systems using aluminum plate fuel with wider channel spacing had demonstrated that effect. Dr. Garrick was curious because of Mr. Ostrander's mention that we had never seen a positive reactivity experiment. Mr. Ostrander said that he had been referring to conventional samples rather than to experiments. Mr. Zane stated that we can see that kind of effect when we dump the water as we have more than optimum water above the fuel. Dr. Garrick

responded that he was hoping that we were both talking about the same reactor [Dr. Garrick is quite familiar with the UCLA reactor].

Mr. Ashbaugh, with an after-thought, asked if the ESA was already signed. Mr. Zane said yes, but if there were any suggestions to state them now. Mr. Ashbaugh suggested that we insert the whole slug of polyethylene at once, do an approach to critical and in that way, only handle the sample once. His suggestion was added to the ESA and initialed by four members of the committee.

The last experiment 18-19 was described by Mr. Ostrander. He noted that the interveners had brought up the positive graphite temperature coefficient, and that the University of Washington had actually measured a value for this coefficient. Our reactor does exhibit a positive coefficient of some sort after an hour or so of operation. Tests show a linear relationship composed of a negative coefficient associated with the water temperature and a positive coefficient associated with the graphite temperature. If the reactor is run long enough so that equilibrium is achieved, we get the kind of coefficient that people normally report for Argonaut reactors. The positive value from our transient tests is approximately the same value as reported by the U. of W. and might be equivalent to the offset of the negative coefficient in the graphite of the Brookhaven reactor as reported to the Geneva convention in 1955, attributed to the expulsion of nitrogen in the graphite or the decrease in the air density. This experiment will follow a different kind of a transient, plus it would allow us to deviate from our normal procedures and also it would allow the primary to get hotter than our normal operating range. Mr. Zane put a limit on the primary flow at 40 gallons per minute. Dr. Wegst suggested a limit on the primary outlet temperature of 180 degrees F. After some discussion, Dr. Wegst pointed out that running with the secondary cooling off constituted the unique part of the experiment. The committee approved the ESA.

The meeting was adjourned at 11:40 A.M.

A. Zane, Secretary
Radiation Use Committee

UNIVERSITY OF CALIFORNIA, LOS ANGELES UCL

BERKELEY · DAVIS · IRVINE · LOS ANGELES · RIVERSIDE · SAN DIEGO · SAN FRANCISCO SANTA BARBARA · SANTA C

COMMUNITY SAFETY DEPARTMEN
OFFICE OF RESEARCH & OCCUPATIONAL SAFET
LOS ANGELES, CALIFORNIA

1981 NOV 18 AM 9:21

November 16, 1981

Director
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Docket 50-142
License R-71

Dear Sir:

Two possible violations of UCLA's Technical Specifications were reported by telephone to the USNRC Region V on October 29, 1981. The action cons[illegible]ting the first possible violation was committed on October 23. On Octobe[illegible], it occurred to us that we might have violated the Technical Specifications, and we subsequently reported the action on October 29. We believe the reporting delay may also be classified as a violation.

The first apparent violation was a failure to insert all control blades prior to removal of a sample of large negative reactivity (UCLA Technical Specification VII.B.2) No physical consequences ensued as a negative shut down margin of 70 to 80 cents remained after removal of the sample.

If Technical Specification VII.B.2 was indeed violated then the reporting delay is also a violation under Technical Specification VIII.M.1A.

The precise nature of the violation is uncertain and depends upon the interpretation of Technical Specification VII.B.2. That specification clearly applies to a critical reactor; but does it also apply to a sub-critical reactor containing a sample of know negative reactivity and known shutdown margin? UCLA requests NRC clarification of this question.

UCLA's Radiation Use Committee was convened on November 2, 1981 to review the circumstances of the apparent violations. (A copy of the meeting minutes is available if desired). The following is a management summary of those minutes:

First Violation - Findings

1. A new and novel experiment was run on October 23, 1981. The experiment was intended to identify a possible sample of positive reactivity.

2. The written procedure for the conduct of the experiment assumed that the sample would display a positive reactivity.

3. The Reactor Supervisor reviewed the written procedure and assumed that standard procedures would govern in the case of unforseen developments.

4. The written procedure was not submitted to the Radiation Use Committee.

5. The Senior Reactor Operator running the reactor followed the procedure as written, but failed to implement the standard procedure when it was found that the sample reactivity was in fact negative.

6. The Senior Reactor Operator acted with unnecessary haste, in prosecuting what may have been a deficient procedure. Although he knew his procedure was safe, he did not consider the possibility of a technical violation.

7. The principal cause of this apparent violation was a failure to anticipate, and correctly respond to an unexpected development.

First Violation - Recommendation/Approvals

1. The Committee approved a recommendation that the procedural aspects of new and novel ESA's must be reviewed by the Radiation Use Committee prior to implementatio

Second Violation - Findings and Recommendations.

2. All personnel normally expected to notify the NRC were reminded of their responsibilities in that regard.

Very truly yours,

Walter F. Wegst

Walter F. Wegst
Director
Research & Occupational Safety

WW/jr
cc: Walnut Creek USNRC Reg. V

ANNUAL REPORT

UCLA NUCLEAR REACTOR

1 January 1978 through 31 December 1978

A. Reactor Operating Experience

The operations of the UCLA R-1 Reactor totaled 340 port operating hours (a port operating hour is the number of irradation ports used times the irradiation time) for the year 1978, and expended 20.3 megawatt hours of thermal energy. Of the 340 port hours of operation, 71.8% were devoted to research, 15.8% to class instruction and demonstrations (includes the training of new reactor operators), 10.5% to reactor maintenance which includes calibrations and test runs, and 1.9% to demonstrations for high school groups and other miscellaneous tours.

The total operating time was up 17% over that of 1977; an increase attributable to a combination of increased demand and reduced down time. Table 1 shows the overall comparison with the four previous years.

TABLE I

	1974	1975	1976	1977	1978
Research Hours	177	146	158	188	244
Class Instruction	28	39	27	88	60
Maintenance	52	31	23	14	36
TOTAL	257	216	208	290	340
Mwh(th)	14.8	11.9	13.1	15.9	20.3

B. Unscheduled Shutdowns and Abnormal Occurrences

1. Unscheduled Shutdowns

a. 30 January 1978. A drop rods scram was initiated by the loss of air pressure which caused the dump valve to partially open. This was reported as an abnormal occurrence and will be described in that section.

b. 10 February 1978. A full scram occurred approximately one and a half hours into the run when the UCLA campus suffered a mometary power failure. Since the reactor instrumentation is designed to shut the reactor down under these conditions, the reactor operator continued the run upon restoration of power.

c. 17 March 1978. A High Flux scram occurred approximately 5 seconds after a rabbit sample worth $.10 was fired out of the reactor ($.10 worth of reactivity will cause the reactor to go on a stable period of 99 seconds). The scram was initiated by channel 1 of the safety amplifier which under test showed a trip level of 118% rather than 125%. Since the Ar 41 recorder which has a slow response time also showed a transient spike at the time of scram, it was concluded that a line transient when the reactor was on a positive transient period above 100 KW could have caused the safety amplifier to trip.

d. 24 March 1978. A Period Scram occurred as the operator was making his approach to critical. Since the Log N and Period Amplifier was recently replaced and not as yet optimized to the reactor operating system, the operator, being unfamiliar with the new unit, made a normal approach to critical as he had done in the past. When the Log N and Period Amplifier came on scale, the transient jump resulted in a period scram. The operator was cautioned to proceed more slowly until the instrument came on scale then continue in a normal manner. All other operators were again verbally forewarned about the problem.

e. 31 March 1978. A Period Scram similar to that of 24 March 1978 occurred. All operators were verbally cautioned to hold power at .02W until the Log N and Period Amplifier came on scale and then proceed to the designated power in a normal manner. In the mean time, continued communication with the manufacturer finally produced an explanation of the behavior. Since the step junction through 0 is normal according to the manufacturer, a possible solution would be to decrease the high impedance damping while increasing the low impedance damping. The suggestion was adopted and the action corrected the problem.

f. 18 May 1978. A High Flux scram was initiated by the safety amplifier when the student trainee, unfamiliar with the controls attempted to level off at 100KW and over shot the mark and caused the safety amplifier to initiate shut down when the power reached 125KW. It was recommended that the trainees be more closely supervised while approaching full power until they develope the proper feel and technique for leveling off at an assigned power.

2. Abnormal Occurrances

a. During a run on 30 January 1978, the reactor experienced a drop rods scram. The scram was caused by the drop in pressure of the Engineering air supply. This allowed the dump value to open enough to activate the "dump value closed" sensing micro switch. The loss of air pressure was due to the operation of an air turbine in another building of the Engineering complex.

The unanticipated reactivity change came about when the dump value, being partially opened, bypassed part of the primary flow through the core, thereby causing an increase in moderator temperature. Since the reactor has a negative temperature coeffieient, the rod under automatic control began driving out to compensate for this loss in reactivity. The reg rod reached 80% before it was noticed by the operator and the reactor shutdown before the operator could take further action. A check of the hourly readings showed the reg rod at the 50% position before the drop in air pressure occured.

The problem of the loss of air pressure had been noted previously and had been brought before the Radiation Use Committee which recommended the purchase and installation of a back-up air compressor system to the reactor air supply. The purchase order for that compressor was initiated on 1 January 1978.

In view of the scram of 30 January 1978, the Radiation Use Committee met on 31 January 1978 and recommended that until the air compressor back-up system was operational, the reactor could continue to run if the dump value was powered by a high pressure nitrogen cylinder and an observer stationed at the turbine laboratory. This was immediately implemented by an engineering change order.

The back-up compressor system was installed and became operational on 8 March 1978; at which time the operating procedures were returned to normal.

b. There were no other abnormal occurrances during 1978.

C. Preventive and Corrective Maintenance

The required annual tests and calibrations were completed and recorded in early 1978. Advantage was taken of a scheduled shut-down in December, 1978, to perform the tests and calibrations for 1979. The various radiation measuring systems were also calibrated on a semi-annual basis as required by the Technical Specifications.

The corrective maintenance having any safety significance is divided into two catagories, electrical and mechanical. A brief summary of the maintenance follows:

Electrical:

5 January 1978. The "Dump Value Closed" sensing switch checked and readjusted for proper operation.

17 March 1978. Safety amplifier checked for proper operation after a high flux transient scram. Channel 1 tripped at 118% and was re-adjusted to 125%. Amplifier checked OK and all circuits functioned properly.

21 March 1978. The log CIC was removed and checked for proper electrical connections when it appeared to be driving negative even with the chamber gamma compensation voltage removed. All connections checked out OK although there may have been moisture in the connectors. They were dried with a hot air gun before reinstallation.

4 May 1978. The new log N and period amplifier function switch was installed and the unit calibration checked.

10 May 1978. The linear recorder was cleaned and serviced due to irratic operation.

11 May 1978. The Solu Bridge was serviced and checked because of dimenished brightness of the eye tube.

18 May 1978. When the integrity of the auto-controller was questioned, a systems check was performed on the unit. The system was found to be well within specifications.

23 May 1978. The inhibit bypass switch on the Log N recorder required readjustment as it appears the trainees were using this disc to turn the recorder during the check out procedure.

24 May 1978. The start-up channel was responding to switching transients. While the problem was probably due to a faulty ground connection in the preamp, all power connections were bypassed, tube shields left off during the previous maintenance were replaced, and the system cleaned and checked. The unit now functions properly.

19 June 1978. Fuse replaced in area radiation monitor power supply.

26 June 1978. The voltage reference source in the linear recorder failed. The unit was rer[illegible] and the system checked using the Keithley calibration current so[illegible]

Mechanical:

11 May 1978. The filter in the shield tank demineralizer system was replaced.

2 June 1978. A pint of 30% H_2O_2 was added to shield tank to remove the algae growth.

Minutes of the Radiation Use Committee

December 22, 1977

Members present: I. Catton, C. K. Chan, J. Hornor, G. Pomraning, A. Zane

Members absent: V. J. Dhir

Guests: C. Ashbaugh, N. Ostrander

Dr. Catton called the meeting to order. The first order of business was Mr. Hornor's report on his attempts to effect an exchange of in-depth reviews of the reactor operations and procedures with other similar facilities.

Since cost was an important factor, Mr. Hornor limited the area of interest to a 200 mile radius which encompassed the following reactors: Cal Poly, UCSB, Northrup, General Atomic, and UCI. The commercial reactors, Northrup and G.A. were immediately eliminated since much of their work was proprietary. The Cal Poly and UCSB reactors are of low power and are in no way similar to ours, which then narrows the field down to UCI. The reactor HP at UCI informed Mr. Hornor that the in-depth review at UCI is done quarterly by a different faculty member and that while this does take one man day, they like the program as it allows each faculty member to become familiar with the reactor program. It was suggested that our director contact UCI's director if this approach is to be pursued further.

Dr. Pomraning suggested that perhaps the nuclear faculty members should take on the task on a rotating basis, to which Dr. Catton named the available members as being Drs. Pomraning, Dhir, Chan, and Kastenberg. Dr. Catton being the director would have a vested interest in the operation and therefore would abstain from serving.

Dr. Pomraning then inquired as to what was involved in this review. Mr. Hornor replied that the review mainly consisted of reading the operations log, the reactor supervisors' logs and the procedures for any inconsistencies and possible violations of the technical specifications.

Dr. Pomraning then suggested that Mr. Hornor do the review for 1977 and that the four aforementioned faculty members conduct a quarterly review for 1978 on a rotating basis under the guidance of Mr. Hornor and possibly Mr. Zane provided there is no conflict of interest.

The second order of business introduced by Mr. Zane was Mr. Ashbaugh's and Mr. Ostrander's plan to train personnel from nuclear oriented electrical utilities in hot fuel handling. Dr. Catton stated that the reason to consider this type of operation was strictly economic and that monies earned could go the the support of graduate student research. One of the ways to earn this money was through the use of the reactor as a training tool. In conversations with the utility companies in the area, one of

their desires is to give their own people hands-on experience in handling hot fuel. Furthermore, the characteristics of our reactor are quite similar to combustion Engineering's, PWR as well as GE's PWR.

Dr. Pomraning inquired as to what would be required in order to satisfy the NRC. Dr. Catton replied that the NRC would like to send an observer the next time we did any fuel handling and that they would be invited.

Mr. Ostrander then stated that there may be some complications as it is not quite clear as to what extent the hazards analysis has been superceded by the technical specifications and that the hazard analysis specifics that fuel handling will be done by the reactor operating staff. This may be questioned by the NRC.

Mr. Hornor then stated that in checking with EH&S they see no reason why this fuel handling operation can't be handled. However they want a written plan that is approved by this committee since this is the single most hazardous operation that is conducted in this laboratory. A person handling the fuel cask would require just 4 seconds of exposure for a reportable incident should the shielding of the cask suddenly be removed and a few minutes to a lethal dose.

The approved written plan along with the invitation to the NRC should be submitted to EH&S at least 90 days prior to the actual operation to allow the NRC time to review the operation for compliance with ALARA.

Another point brought out by EH&S as related by Mr. Hornor was the fact that their trainees who will be handling the fuel are considered students and consequently since the campus community now becomes involved, we must also obtain approval by the Campus Radiation Safety Committee because the liability for student injury falls back on the University.

Mr. Zane then brought up the alternative of having the reactor staff remove and store the hot fuel then allow the trainees to load and unload the reactor using the new fuel which for all practical purposes can be handled by hand.

Dr. Catton agreed that this alternative is a good one but we should let them believe that they are handling the hot stuff as part of this training is performance under stress.

Dr. Pomraning inquired as to what the problems are in unloading and inspecting the fuel and about how long would it take? Mr. Ashbaugh replied that in 1974 when he conducted the operation, 16 fuel bundles were removed and inspected in one day and the rest was completed the following day. To this, Mr. Hornor remarked that the reactor should be shutdown for at least three weeks as the fission decay curve doesn't start to linearize until after the third week. This would then entail a realistic down time of at least six weeks which includes the training and the reloading of the original fuel.

Dr. Catton and Mr. Zane agreed in response to Dr. Pomraning's question that the six week down time would not adversely affect the normal reactor operation as our average use factor is approximately 6 hr/week and by advanced notification to our users, they could adjust their schedules accordingly.

Mr. Ashbaugh stated that this was really only a feeler to see is SCE would actually go for a proposal such as this whereby Dr. Catton emphasized that while we only expected somewhere in the order of $3000 for this initial operation, it was the visibility that was important as we would like to develop an extensive training program for the nuclear oriented utilities. Once they were aware of our ability to perform there services, we would expand the training program to satisfy these needs.

Dr. Pomraning suggested that a detailed plan be implemented, Dr. Catton commissioned Mr. Ashbaugh to prepare the detailed plan which would be brought before the committee for discussion and approval, after which Dr. Catton and Mr. Ashbaugh would present the plan to the Campus Radiation Safety Committee, and report back to this committee.

The next order of business concerned an experiment proposed by Applied Nucleonics Corporation as explained by Dr. Catton. A shield wall cable penetration is to be tested both for neutron and gamma leakage. Special cast concrete blocks will be fabricated to accommodate the cable penetration. These blocks with a combined thickness of 6 feet will replace the two original two blocks above the reactor core.

Mr. Hornor stated that EH&S wished to know how we intended to dispose of 20 tons of garbage and would we require an amendment to do the experiment? Mr. Hornor futher stated that he would try to get a reading from the state as to the level of activity an item could have and still be considered non-radioactive.

Since there may be follow-on experiments and other multi-uses for these blocks as well as disposal problems it was decided by the committee to give considerable throught to their design such as casting in layers to minimize the disposal problem and to accommodate the original 5 foot thickness should the need arise.

The committee agreed that the project can be executed with no increased probability or consequences of any previously analyzed accident nor did it create the possibility of a new type accident provided the radiation levels above the shield were tightly monitored so as not to exceed the current levels above the reactor at full power. The NRC would be advised of the actual equipment once it becomes firmed up so that they may review it before hand.

Other non-agenda business included Mr. Ashbaugh's report on the reactor air supply. The reactor air is supplied from the main supply of Engineering Unit I. At times there are experiments conducted in other engineering labs that utilize this supply and can drag the supply pressure down to a point where it could cause the dump value to release which would scram the reactor.

Some serious implications of this problem are that the dump valve closed sensing switch could be misadjusted which would allow the dump valve to partially open before initiating a scram. This would cause reduced flow through the core which would cause a greater temperature rise and consequently a reduction in activity which would be made up by the automatic withdrawal of the reg rod. Should the air pressure suddenly revert to normal, it could cause the dump value to fully close thus restoring the normal flow through the core which would result in a positive reactivity excursion.

The shops and maintenance personnel suggested as a solution to the problem that we install a backup air supply tank and compressor with the appropriate check valves that would isolate our supply from the main supply should that supply fail to maintain pressure.

Upon discussion of the problem, the committee recommended that the laboratory purchase an air compressor system as a back up and install it in the reactor control air supply.

The final order of business brought forth by Mr. Ashbaugh was a request for a power increase to the reactor. The advantages would be that we could increase the sensitivity of our activation analysis as well as possibly enter the isotope production field.

Dr. Catton suggested that we do this in stages. The first stage being to install a proper heat exchanger and clean up the process pit plumbing. The heat exchanger should be at least 1 megawatt and preferably 2 megawatts. A cooling tower should also be investigated.

The meeting was adjourned at this point.

A. Zane, Secretary
Radiation Use Committee

AZ/li

U. S. ATOMIC ENERGY COMMISSION
DIVISION OF COMPLIANCE
REGION V

Report of Inspection

CO Report No. 50-142/68-1

Licensee:	University of California at Los Angeles (UCLA) License No. R-71 Category E
Date of Inspection:	May 22-23, 1968
Date of Previous Inspection:	October 11-12, 1967
Inspected by:	W E Vetter 6-7-68 W. E. Vetter Reactor Inspector
Reviewed by:	G. S. Spencer 6/7/68 G. S. Spencer, Senior Reactor Inspector
Proprietary Information:	None

SCOPE

Type of Facility:	Argonaut
Power Level:	100 kwt (500 kwt for short periods of time)
Location:	Los Angeles, California
Type of Inspection:	Routine, announced
Accompanying Personnel:	None

SUMMARY

Safety Items - No significant safety items were noted.

Noncompliance Items - No items of noncompliance were noted during the visit.

stabilize nuclear channel long-term operation so that the need for detector relocation can be kept to a minimum.

According to the facility records, during conduct of the 500 kw operation experiment (see Section S. of this report) the nuclear power level channels were set to trip at 115% of 500 kw. In addition, a special linear power level channel was provided and set to initiate a reactor scram at 115% of each test power level, i.e., 115, 230, 345, 460 and 575 kw.

4. Graphite Temperature Coefficient

A report to the Commission by the University of Washington (letter to D. J. Skovholt from A. L. Babb, dated January 4, 1968) deals with a positive graphite temperature coefficient which had been noted during operation of the University of Washington Argonaut reactor. As a result of the subject report, an effort was made during the current visit to identify possible similar effects relative to operation of the UCLA Argonaut reactor.

Dr. Smith informed the inspector that he had received a copy of A. L. Babb's letter and that he had attempted, unsuccessfully, to measure the affect of graphite heating in the UCLA reactor. He said that preparations for the test had involved the fabrication of a graphite log, which was to be inserted adjacent to a fuel can and heated, incrementally, to determine possible reactivity effects. Smith said the experiment had never been performed because the heater wires around the graphite log persistently "burned out" during out-of-core tests. He said the problem was one of inadequate heater wire insulation.

However, during the review of the console logbook, the inspector noted that several, three to four hour, reactor operating periods at 100 kw had been performed. By reference to the console logbook data concerned with core reactivity changes as a function of time and the temperature of the water moderator, it appears that a positive graphite temperature of 0.006% $\Delta k/k/^{o}F$ exists. This is about one-half of the coefficient measured during the University of Washington experiment. Dr. Smith said that in spite of the foregoing, he intended to experimentally determine the graphite temperature coefficient as soon as promising test equipment could be developed.

G. Core and Internals

1. Control Rod Drop Times

Although control rod drop times are not required by the facility license, Mr. Horner agreed during the previous inspection visit that drop times should be measured on a routine basis (see CO