V

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

DECLARATION OF BOYD NORTON

I, Boyd Norton, do declare as follows:

1. From 1960 to 1969 I was employed in reactor safety studies at the National Reactor Testing Station in Idaho. Five of those years were spent as Group Leader of the Nuclear Test Section of SPERT (Special Power Excursion Reactor Test). I was in charge of operation of both the SPERT I and SPERT III reactors. A more detailed description of my professional qualifications is attached.

2. The SPERT program centered on investigating the behavior of various reactors during power excursions in an attempt to better understand the potential vulnerability of certain induced by large reactivity insertions. A power excursion is an accident unique to nuclear reactors in which power can rise from zero to billions of watts in very much less than a second. This can result in melting of the fuel and explosive disassembly of the reactor core.

3. The SPERT I reactor's "D" core was eventually intentionally destroyed in such a power excursion. I was at the reactor controls

on November 5, 1962, the date of the final test, and was responsible for ejecting the neutron-absorbing rod from the reactor which made it supercritical and, milliseconds later, resulted in SPERT I-D's destruction, through extensive melting of the fuel accompied by a violent explosion.

5. Perhaps the most significant conclusion of the SPERT destructive tests was the unpredictability of destructive threshhold during power excursions initiated by large reactivity insertions, 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.

6. Thus, even after an extensive series of actual tests with the SPERT reactors, there is much about the behavior of <u>those</u> 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 the ones on which we performed our tests--for example, the UCLA Argonaut.

7. 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. I have reviewed these analyses and a number of other documents relative to the application of the University of California to obtain relicensing of the reactor situated on the

UCLA campus.[1/] 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, I 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.

8. Based on my review, it is my conclusion 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, given the population density nearby. The analyses done to date do not, in my opinion, 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. 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 at operating Argonaut-type reactors.

1/ Among the documents reviewed were those sections of the following which address excess reactivity issues: UCLA's 1960 Hazards Analysis and its 1980 Safety Analysis Report; the critique thereof contained in the August 25, 1980, Supplemental Contentions of the Committee to Bridge the Gap; the Contentions as admitted into the relicensing proceeding by the Atomic Safety and Licensing Board; "Analysis of Credible Accidents for Argonaut Reactors" by Hawley, Kathren, and Robkin; and a November 23, 1981, memorandum from P. Neogy of Brookhaven National Laboratory to J.F. Carew, subject: "Transient Analysis of the UCLA Argonaut." Certain other related documents were also reviewed and are identified in the text of this declaration.

9. For these and other reasons identified herein, I conclude that the original restrictions imposed on the UCLA reactor by the AEC in the initial license, particularly the condition limiting excess reactivity to less than that necessary for prompt criticality, were appropriate, given the operation of the device by students. The subsequent changes to the facility have, in my opinion, resulted in a gradual but significant erosion of important safety margins, increasing both the probability and consequences of a potentially serious accident at the facility. The bases for my conclusions are as follows.

10. UCLA stated in its original Hazards Analysis in 1960[2/]:

> 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.

Given the operation of the reactor by students, who can be expected to make mistakes, the limitation to 10 kw_{th} and approximately .6% $\Delta k/k$ excess reactivity was quite prudent.[3/]

11. The Intervenor points to a number of developments over the years at the UCLA reactor which have considerably altered the situation: a quadrupling of excess reactivity, a ten-fold increase in reactor power, discovery of smaller than expected negative

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2/ page 19

3/ 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.

reactivity coefficients and the unexpected discovery of several positive reactivity effects, plus the addition of a pneumatic tube "rabbit" system which makes possible new mechanisms for rapid insertion and removal of reactivity. The Intervenor also points to a series of violations of AEC/NRC regulations and license conditions, including several relating to rules and procedures designed to control excess reactivity, as well as calibration errors, bypassed interlocks, maintenance problems, and lapses in administrative controls--all of which it alleges increase the chances of accident at the UCLA facility. I concur that safety margins have been substantially reduced by these developments, which will be detailed below.

12. As I understand it, 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 there are no credible mechanisms for 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 rest most of its case in that regard on the assertion that the BORAX I[4/] and SPERT I tests conducted at the National Reactor Testing Station in Idaho in the nineteen fifties and sixties "proved" that the requested level of excess reactivity is safe in the UCLA Argonaut. As UCLA put it in its 1980 Application[5/]:

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

13. <u>That simply is not the case</u>. <u>In fact, the SPERT I reactor core was completely destroyed by a \$3.50 insertion</u>, which resulted in extensive melting and explosive disassembly of the core.

4/ The BORAX reactor was a reactor similar to SPERT used for a related series of excursion tests at NRTS in the early 1950s.

5/ p. V/3-6, cited in Contention I.3.f.

Non-explosive melting of fuel was observed with even smaller reactivity insertions.

14. 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% ± .3% Δ k/k).6/ 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. One of the workers was impaled on the ceiling by a control rod; because of the intense radiation field, it took six days to remove the body by use of a remotely operated crane and closed-circuit television. Staying in the building for mere seconds resulted in a year's allowable dose of radiation for the rescue workers.7/

15. 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 clearly demonstrated in the attached photos of melted plates from the $3.50 excursion--that would by itself say nothing about whether plate-type 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 12 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 another 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

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6/ $3.54 would be the equivalent of between approximately 2.3% and 2.7% Δ k/k, depending upon the value used for the delayed neutron fraction.

7/ 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.

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.

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

17. 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.

18. The differences are significant. Plate and meat thicknesses are different, as are coolant channel widths. We 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 in water, corroding the cladding, 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. 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. Furthermore, 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 is increased, as by oscillation, are other important differences. UCLA's fuel meat composition is in the eutectic range, thus melting at a significantly lower temperature than would the BORAX or SPERT fuel meat. There are a number of other differences as well.

19. 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 attached plot of energy versus reactor period, taken from Thompson and Beckerly's Technology of Nuclear Reactor Safety, inculded in the Intervenor's Supplemental Contentions at p. V-23. As is shown there, BORAX produced substantially more energy than SPERT, and SL-1 more than either, given the same initial reactor period. Seemingly minute differences in metal-water ratios, temperature and void coefficients, etc. had marked effects on total energy released.

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20. 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.

21. 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 National Reactor Testing Station in Idaho. UCLA relies largely on the BORAX tests in its orginal analysis; Hawley et al on the SPERT ID series of tests; and Neogy on the SPERT IA series. (Surprisingly, none even touch 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.

22. Based on extensive involvement with the SPERT tests, I take substantial issue with such an assumption. First of all, we never intended the SPERT tests to be used in such a fashion. We were attempting 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, we 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.

23. If the SPERT core was destroyed with a $3.50 insertion, we would have been quite concerned to learn of a reactor operator using 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. We never intended our SPERT tests to be so used--the uncertainties are just too large. To say, as the Hawley et al review essentially does, that the SPERT I-D 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 our data.

24. 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 expulsion, fuel plate expansion, Doppler effect, density changes, "warm neutron" effects, and the final shutdown mechanism, rapid disassembly of the reactor core.

25. With the above comments about the difficulties inherent in such extrapolations, I will now discuss the three attempts that have been made to extrapolate the BORAX and SPERT data to the UCLA case.

The 1960 Hazards Analysis (1980 SAR)

26. The UCLA Argonaut-type reactor was designed for a maximum power of 10 kw_{th} and maximum excess reactivity of about .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 .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 .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 to show the magnitude of the safety margin and to provide further support for the .6% limitation.

27. 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.

28. 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.)

29. There is some confusing language in the text of the Analysis which I trace to the fact that it was apparently copied, virtually verbatim, from the University of Florida Hazards Analysis of a few years earlier. (sections attached). 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 the analyst states that the reactor will tolerate a power excursion of at least that magnitude without melting occurring at the hottest part of the fuel. While this could simply be viewed as asserting that a certain point is the end of the safe zone instead of saying it is the beginning of the danger zone, another interpretation is made clear by reviewing the U of F Hazards Analysis from which UCLA's is copied, essentially word for word, including the language in question.

30. UCLA, in its Hazards Analysis, acknowledges the Hazards Report of the University of Florida, asserts that the reactors are similar, with the "only significant difference between the two reactors" being the fuel enrichment--20% for U of F, 90% for UCLA. A comparison of the analyses shows a related difference--the U of F reactor's fuel was 46 w/o U-Al, whereas UCLA's is right at the eutectic point, 13.4. (See page 1 of U of F and UCLA's "Estimation of Effects of Assumed Large Reactivity Additions.") U of F's fuel meat (where the hot spot would be

located) melts considerably <u>above</u> the melting point of aluminum, unlike UCLA's, which melts below the critical temperature for aluminum. Thus, the original Hazards Analysis for U of F was correct in asserting that its limiting period and excess reactivity would not cause melting (because it would be sufficient merely to raise the center of the plate to the melting point of aluminum, whereas the center melts at a considerably higher temperature.)[8/] UCLA, when it copied the U of F analysis, failed to correct for the different composition of the fuel, thus keeping in language contradicted by the calculations. At UCLA, the same peak temperature predicted by the U of F analysis as <u>not</u> causing melting <u>would</u> cause such melting.

31. As the original Hazards Analysis calculations make clear, 2.3%Δ k/k would be sufficient to cause fuel melting at UCLA, if the assumptions employed are correct. I have made clear above my objection to such 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 I 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 .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 .6% so there would be a margin of safety for errors in calculation or operational errors that might slightly exceed the license limits. As I read that analysis, it shows melting at 2.3%Δ k/k, and supports a .6% limitation. It cannot be used, in my view, to justify a limit at or close to 2.3%.

8/ Peak temperature will occur in the fuel meat, where the energy is generated. If the meat melts above the melting point of aluminum, the meat won't melt; and the aluminum, which is not where the peak temperature is generated, will not get hot enough (just barely) to melt. This would not be the situation at UCLA, where the meat melts at a lower temperature than the clad.

32. The Intervenor in its Supplemental Contentions points to a number of aspects of the original Hazards Analysis with which it takes issue. For example, the Hazards Analysis uses a void coefficient[9] for UCLA of -.18% k/% coolant void, whereas the current application cites a value of -.164% (p. III/6-5). 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% Δ k/k reactivity insertion. Uncertainties in the precise void coefficient (which can vary by region of the core and other variables) adds substantial reason for added margins of safety.

33. 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.

34. The Intervenor also points out that the Hazards Analysis used a non-conservative delayed neutron fraction (β) of .0074, whereas the Application now cites a figure of .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

9/ A void coefficient is a measure of the effectiveness of the primary inherent safety mechanism in light-water-moderated, highly enriched flat-plate reactors--production of voids in the moderator. In a power excursion, the power rise causes the temperature to rise, which eventually can cause the nearby water to boil. The density reduction reduces the amount of moderation and increases the neutron leakage, slowing the nuclear reaction and eventually stopping the excursion. The larger the negative coefficient, the more reactivity can be compensated by the voids.

results in a shorter exponential period for the same reactivity insertion, and thus more energy release and higher fuel temperature.[10/] 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.

35. 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.

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

37. 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

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10/ The version of the inhour equation cited by Hawley is

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

of the numerical values are changed to reflect more appropriate values (β, void coefficient, and eutectic melting point), substantially less than 2.3% Δ k/k would appear to be sufficient to induce melting--if the methodological assumptions employed are correct. If measured warm, even smaller levels are tolerable.

38. I have indicated above, however, that I have considerable reservations about methodology. To use such a method of extrapolating from one reactor to a different one--to three significant figures--without error bars, assumes that there exists a complete knowledge of all the differences between the reactors and how those differences 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. Substantial errors bars are required, or margins of safety, which is why the Hazards Analysis was correct in concluding that excess reactivity at this facility should be limited to about .6% k/k. The Analysis cannot properly be used to demonstrate that \$3.54, or \$3.00, or any similar value is "safe"; in fact, if the methodology of the Hazards Analysis were to be accepted, it must be taken to demonstrate that such levels of excess reactivity could lead to fuel melting in the UCLA reactor.

<u>The Hawley, Kathren, and Robkin Review</u>

39. 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.

40. 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 I-D, and then scale temperature linearly to the peak temperature estimate during the SPERT I-D destruct test. These approximations are very rough, as can be seen from the plot of energy release to time of peak power vs. reciprocal period, taken from p. 11 of the Miller, Sola, and McCardell report on the SPERT I Destruct Test, which Hawley _et al_ used to estimate energy release for a 7.2 millisecond period.

41. 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^{o} below the melting temperature. No error bars whatsoever are provided for the extrapolation steps nor the final conclusion. (I note that there appears to be a subtraction error in that Hawley _et al_ assert on page 19 of their report that a hot spot of $586^{o}C$ would be $74^{o}C$ below the melting point of the fuel meat, which they cite on the previous page as being $640^{o}C$.)

42. I do not consider 50^{o} to be 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. We noted, for example, at SPERT 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

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

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

45. 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 2½ times the energy assumed in the Hawley report is not explained.

46. Note also that a 7 msec period in the SPERT I-A core is reported to have released 23 Mw-sec of energy, nearly twice that assumed by Hawley based on SPERT I-D 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 indicate an unacceptable probability of a destructive power excursion, one that could release significant amounts of fission products.

47. I should add 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 to me 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 extrapolation from SPERT I-D 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.

The Neogy Memorandum

48. 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 <u>ramp</u> 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 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."

49. 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 I-A 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 I-A transients is reported, let alone against SPERT I-D, BORAX I, or SL-1 transients is indicated.

50. Certain non-conservative assumptions appear to have been used in addition. The values for UCLA's void coefficient, prompt neutron lifetime and delayed neutron fraction are all larger than values reported elsewhere. In addition, 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, and the energy release and temperature estimates that follow therefrom are thus substantially too low.

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

52. 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

53. There are numerous credible mechanisms for initiating a power excursion accidentally at the UCLA reactor. For example, it is my understanding that the facility has had repeated problems with control blades becoming stuck, and that 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, as I understand it, located outside the reactor shield. While normal insertion rate of reactivity with the control blades is limited by the motor, 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.)

54. Another such mechanism for rapid insertion of reactivity is for a large negative worth sample to be irradiated (either in an irradiation port or through the pneumatic tube "rabbit" system) and for the sample to be removed without the reactor operator remembering to first reinsert the control rods. Having to rely upon the reactor operator to follow correct procedure, particularly with student operators, is precisely the opposite of the basic premise of an educational reactor--inherent safety such that the worst mistake possible cannot cause injury.

55. 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 fissionable material directly into the core. I understand that UCLA at one point requested 250 grams of U-235 for irradiation in the reactor's thermal column. If the material were instead placed in an irradiation port, a very sizeable positive insertion would result. That such material could be inserted in an irradiation port instead of the thermal column--as a prank, by mistake, in an act of sabotage, or as a modification of an experiment--could certainly occur, particularly if there had been a history of weak administrative controls at the facility.

56. Furthermore, the fact that 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 Tech Specs. A history of measurement errors or Tech Spec violations at such a facility would substantially increase the probability of excess reactivity limitations being violated.

57. 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 into the core would 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 airpressure 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.

58. 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 control blades 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.

59. I understand also that as a result of vibration tests on the reactor, it was determined that reactivity oscillations were detected, traced to the fact that the reactor is substantially undermoderated with its present plate spacing. If true, this information would be significant because core distortions, for example, those created in an earthquake or an otherwise non-destructive power excursion involving rapid steam formation and water expulsion, could potentially have the equivalent of increasing plate spacing and thus amount to positive reactivity insertion. Furthermore, an undermoderated core presents the possibility of power excursion through increased moderation being introduced. The Hawley review indicated that up to 18.5% $\Delta k/k$ extra reactivity could result from catastrophic mechanical rearrangement or flooding of the core (p. 27), but concluded that such perfect rearrangement or complete flooding was not credible. However, with 18.5% available, far less than perfect rearrangement or complete flooding is necessary for a disastrous power excursion, which all the analyses would appear to accept as occurring at least with a 3% insertion, if not considerably less. Thus, flooding from broken pipes, the shield tank, or the failure of a nearby upstream reservoir could result in a substantial positive insertion, as could the use of water to fight a reactor fire.

60. 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. We 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. We called such behavior "chugging," and found on several occasions incidents 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 by us 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, if the UCLA Argonaut does not have the deflector plates described in the original Hazards Analysis as designed to prevent such repeated excursions by preventing expelled water from returning to the core, then an important safety feature is lacking.)

61. A fire in this reactor raises serious reactivity questions as well. If water, or some other moderating substance, were used to suppress the fire, a power excursion might result. If the control blades melted out of position, the equivalent of a large positive reactivity insertion might ensue. Furthermore, the positive temperature coefficient of the graphite means that as the temperature rose in the graphite, reactivity could increase as well. All of these factors would necessitate very careful plans for fire response, andcould make a fire at the reactor quite serious.

62. The positive temperature coefficient for the graphite is troubling for other reasons 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 (+.006% $\Delta k/k/^{\circ}F$) is larger than the negative temperature coefficient cited for the water (-.0048% $\Delta k/k/^{\circ}F$).

63. 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 millisecond additional delay can be significant. Given the extremely small margins of safety, e.g. Hawley's 50°, 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.

64. 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 <u>measured</u> 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.

65. Graphite temperatures rise significantly after an extended run of several hours in the Argonaut. Plots of reactivity versus time and temperature, included as Exhibit A in UCLA's November 9, 1981, interrogatory answers to CBG, 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.

66. 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 as at the time of the measurement but the graphite, with its positive coefficient for temperature, substantially warmer.

67. The positive temperature coefficient has been reported as approximately + .006% $\Delta k/k/^{\circ}F$ (AEC inspection report 50-142/68-1, p. 6). 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 .6% $\Delta k/k$, or nearly a dollar. A reactor, thus, that was thought to be limited to \$3.00 could thus at times have available \$4.00 because of the positive temperature coefficient.

Conversely, a keep to a licensed limit of \$3.00, it would thus be necessary to have a measured maximum of around \$2.00, if these figures are correct.

68. 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 additional 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. I have indicated elsewhere that I think \$3.00 itself is dangerously excessive.

69. The UCLA Argonaut, with the levels of reactivity being requested, is not inherently safe. Because of the large amount of excess reactivity, 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,

strict 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 happening. 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--these would all have safety significance.

70. Permitting unlicensed operators like high school or junior high school visitors to operate the reactor controls would likewise increase the risks. SPERT III, for example, suffered a serious accidental power excursion when an unlicensed operator was permitted to bring the reactor to critical. A licensed operator was present in the room, but the unlicensed operator was at the controls and, in violation of our rules at the time, a senior operator was not also present.

71. Use of a low-enriched fuel would add some safety margin to the facility, because of the increased Doppler effect. At SPERT we found a low-enriched, uranium oxide core able to withstand larger reactivity insertions than the high enriched uranium-aluminum plates.

72. Samples of large reactivity worth, negative or positive, could be inserted in the reactor, either through the pneumatic tube system or into irradiation ports. There are a number of substances, such as cadmium, that are highly absorbing and would represent significant negative reactivity worth. 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. If large samples such as these are not anticipated to be used, then there is no reason not to bring the excess reactivity level back down to less than .6%$\Delta k/k$, where it used to be, because there would never be large samples to irradiate for which substantial excess reactivity was necessary to compensate for the neutron absorption.

73. Steam explosion and metal-water reaction are possible if a power excursion of the magnitude that appears possible at UCLA were to occur. SPERT, BORAX, and SL-1 all had such reactions explosively occur. As indicated above, the onset of such reactions and the initiating conditions are not fully understood. I am not certain, for example,that such reactions could not occur even if maximum temperatures attained in the fuel were slightly below the melting temperature. An incident that approached but did not quite reach the melting temperature could still release fission products through cladding failure, etc.

74. Furthermore, were substantial Wigner energy stored in the graphite, an excursion not producing enough energy to melt the fuel alone may still have enough to trigger the Wigner release, which could add enough energy to bring the fuel to melting or ignition of either the graphite or fuel. Furthermore, the means of shutdown, rapid expulsion of water, generates substantial pressure pulses capable of substantial alterations to the core configuration. This could increase pathways for airflow to feed the fire. If the excursion were very severe, shield blocks weighing several tons could be thrown in the air; as mentioned above, the SL-1 reactor vessel itself was lifted nine feet in the air during the excursion.

CONCLUSION

75. Based upon the reviews that have been done to date of the potential for power excursions capable of causing fission product in an Argonaut reactor with a $3.00 or $3.54 licensed limit on installed excess reactivity, such an accident is credible at the UCLA facility.

76. The fission product release from such an accident could be substantial. The consequences would be considerably greater than those arising from damaging one of the reactor's 24 fuel bundles during a fuel handling accident. A considerably larger fraction of the core inventory can be released in an accident at this facility than the .2% of the gaseous fission products assumed in the Hawley study (2.7% of the gaseous products in a single bundle containing 7% of the core inventory) at p. 48. An accident involving dropping a single fuel bundle is not the maximum credible accident at the UCLA reactor.

77. Poor managerial controls, violations of regulations and technical specifications, inadequate maintenance, aging equipment, operation by unlicensed operators, operational unreliability, and an attitude that no accident is possible could all substantially increase the probability and magnitude of a potentially destructive power excursion.

78. The UCLA reactor is not inherently safe, with the levels of excess reactivity being requested. And even were the licensed level substantially reduced, mechanisms for insertion of larger-than-licensed amounts would still remain (violation of the license, mistaken insertion of a large positive sample, core flooding, etc.). Its lack of inherent safety makes it essential that the strictest obedience to regulations and procedures be observed, adequate managerial controls enforced, and excellent maintenance and calibration be conducted. Because it is used for student instruction, these controls cannot be counted on, which is why inherent safety or added engineered safeguards are necessary in training reactors.

79. Copying of Hazards Analyses, which turned out to be copied from other analyses for other reactors, is a poor approach which can reproduce and magnify errors.

80. The lack of containment structure, emergency core cooling system, emergency radioactivity removal systems, shields for the control blade drive mechanisms located outside the reactor shield, HEPA filters, spare control blade motors, emergency radioactivity holding tanks, and most particularly, siting characteristics which provide substantial distance between the reactor and densely populated areas all can exacerbate the consequences of an accident at the facility and/or increase the likelihood thereof. Sticking control blades and the problems discussed above regarding graphite add to the potential for trouble.

81. A destructive power excursion, of the SPERT/BORAX/SL-1 type, cannot be ruled out at the UCLA Argonaut-type reactor. Such an accident can credibly occur there, and the consequences could be severe, given the high population density and lack of containment.

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

Boyd Norton

Boyd Norton

Dated this 6 day of Jan, 1983 at Evergreen, Colorado

Professional Qualifications

BOYD NORTON

My name is Boyd Norton. From 1960 to 1969 I was employed in reactor safety studies at the National Reactor Testing Station in Idaho.

During that period I was in charge of operation of both the SPERT I and SPERT III reactors. The SPERT program (Special Power Excursion Reactor Test) was designed to investigate the vulnerability of certain kinds of reactors to accidents induced by large reactivity insertions and to understand better the behavior of reactors during power excursions.

From 1960 to 1962 I was staff physicist at NRTS, assigned to SPERT I.

From 1963 to 1968 I was Group Leader of the Nuclear Test Section of SPERT. I was in charge of operation of both SPERT III and the rebuilt SPERT I.

In 1968 I became Section Chief, Experiment and Analysis Section, of the Power Burst Facility at NRTS. I was in charge of the Safety Analysis Report for the PBF.

Prior to my arrival at NRTS, I received a Bachelor of Science degree in physics from Michigan College of Mining and Technology. I worked summers during 1954-59 at M&C Nuclear, later a subsidiary of Texas Instruments. M&C Nuclear was a metallurgical laboratory where I did research on fuel elements for nuclear submarines.

In 1969 I was offered, as a result of extensive involvement in conservation work in Idaho, a job with the Wilderness Society, where I remained till 1971. I am now a freelance writer and photographer, mainly on conservation themes but also on nuclear matters.

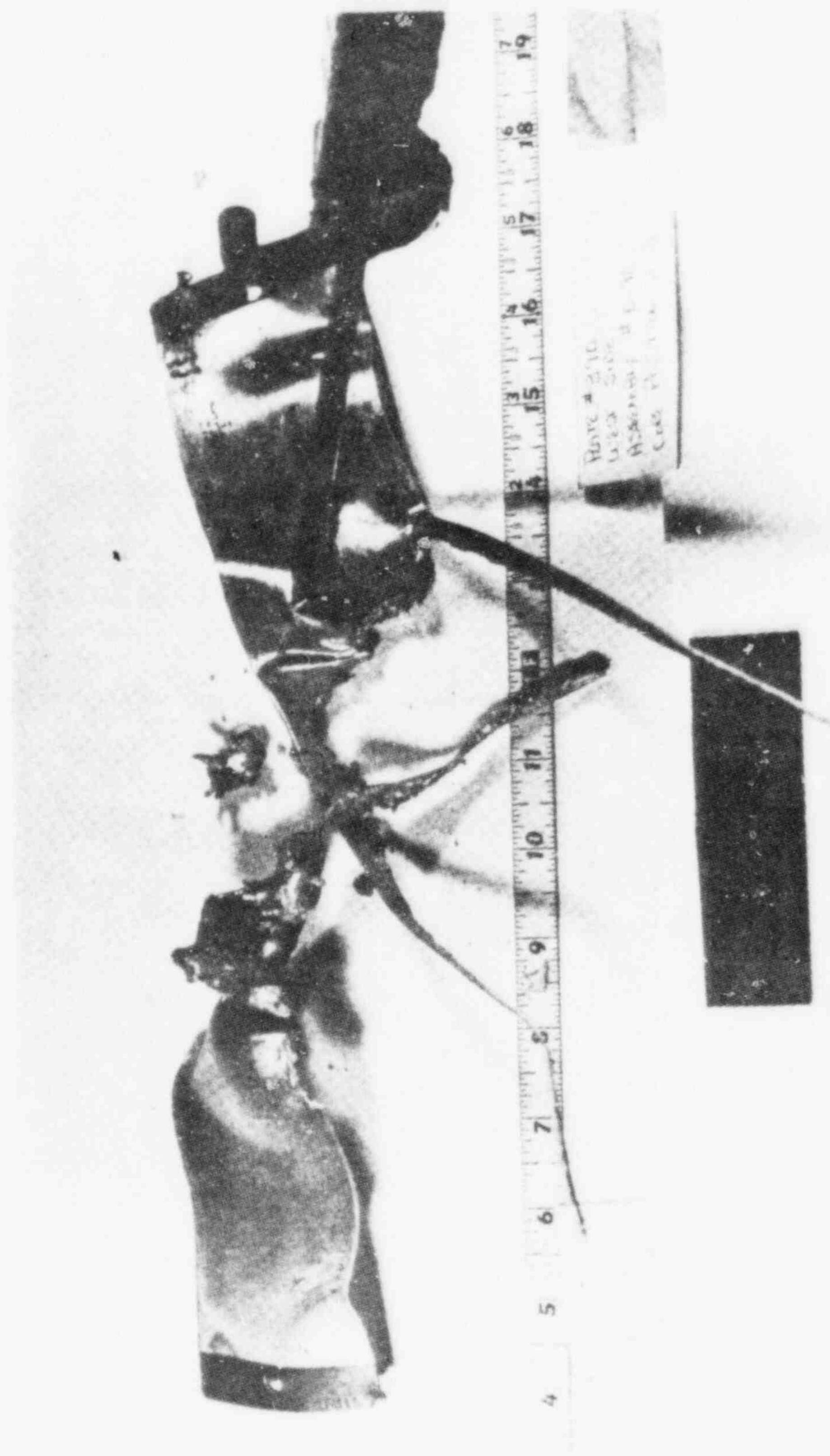

Fig. 39 Fuel plate end fragments from assembly D5.

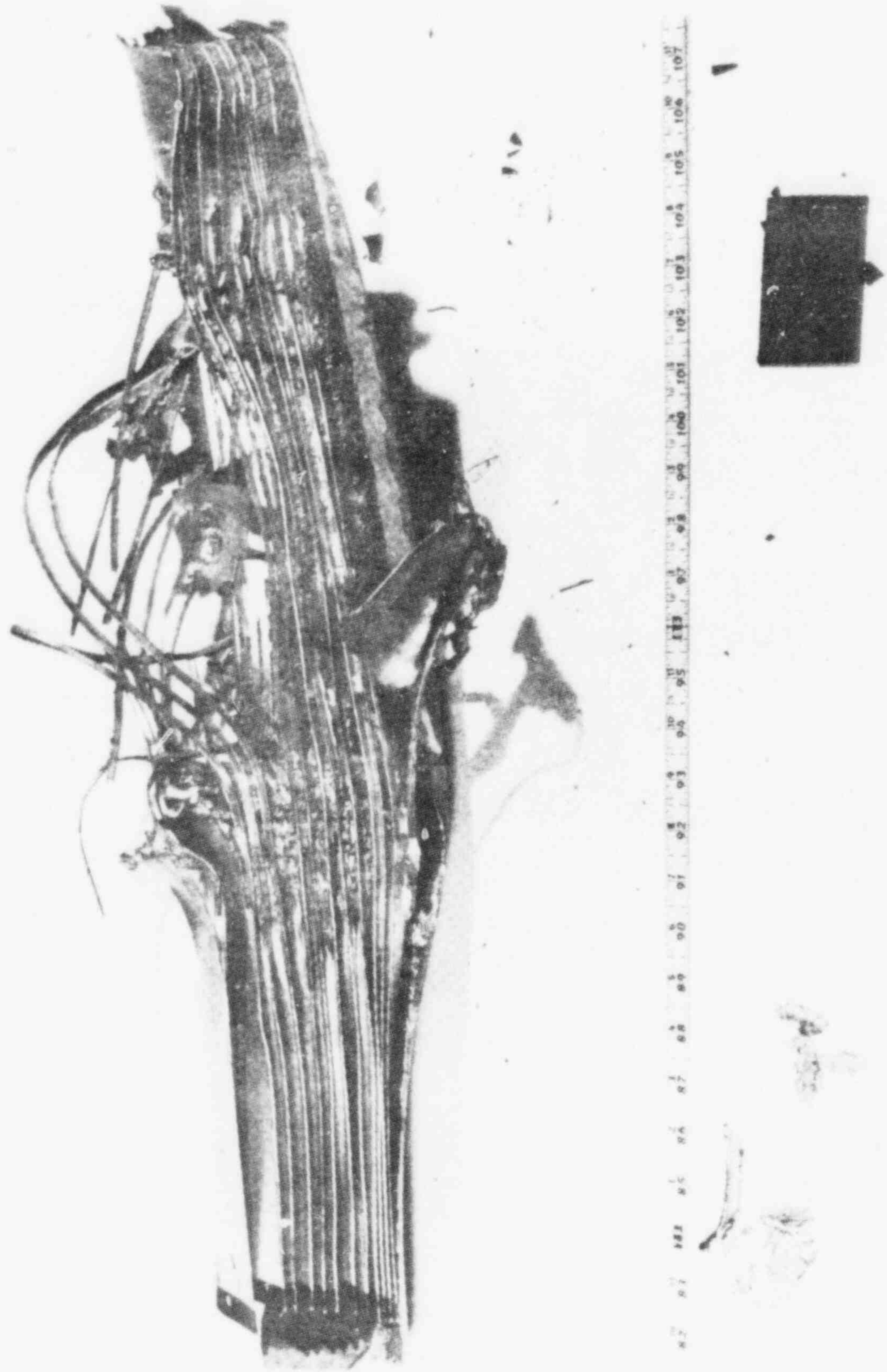

Fig. 38 Typical plates from peripheral assembly-edge view, position G6.

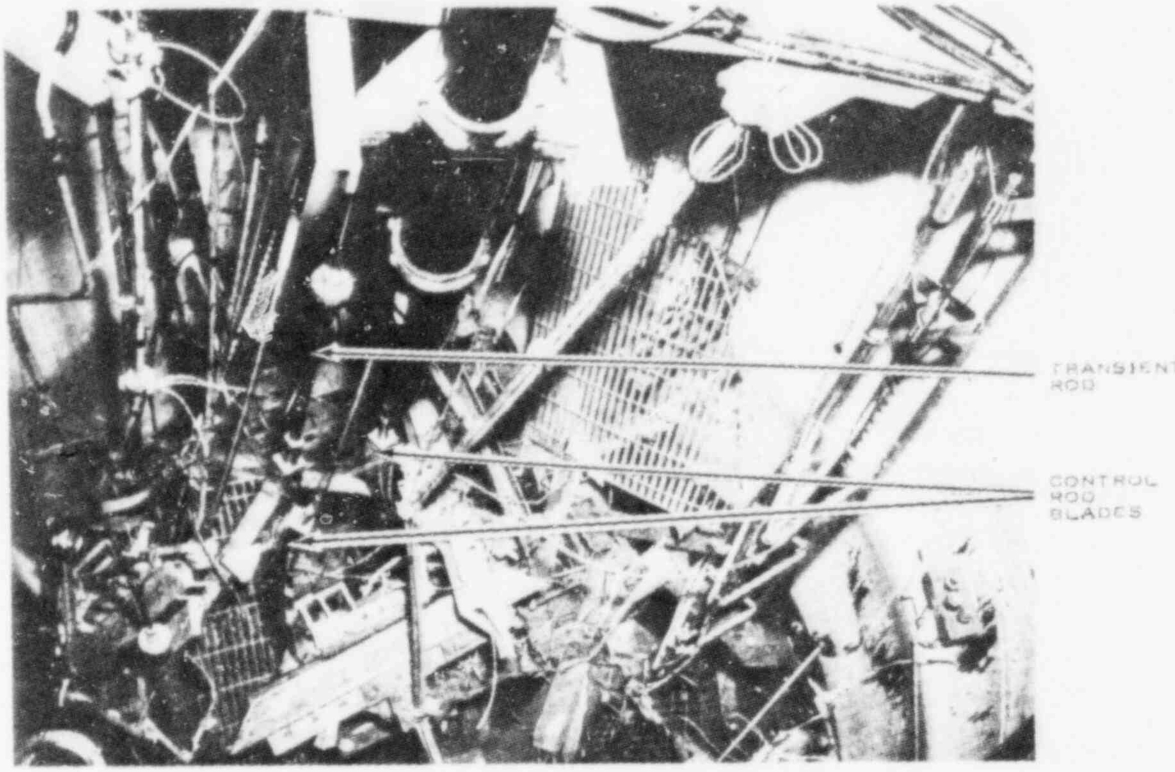

Fig. 36 Close view in vessel after destructive test.

Fig. 37 Close view in vessel after destructive test.

A+

E-1	E-2
2.4	2.4
S,V	S,V
2.3	1.4
175	125
1.2	2.7
1000	2000
4.8	1.0
7.6	2.9
64	50
59.1	59.4
130	245

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…imately 4 msec
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had just reached
…060°C (3767°F).
…in. or 0.889 mm)
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Fr… the start
he excursion, 5%

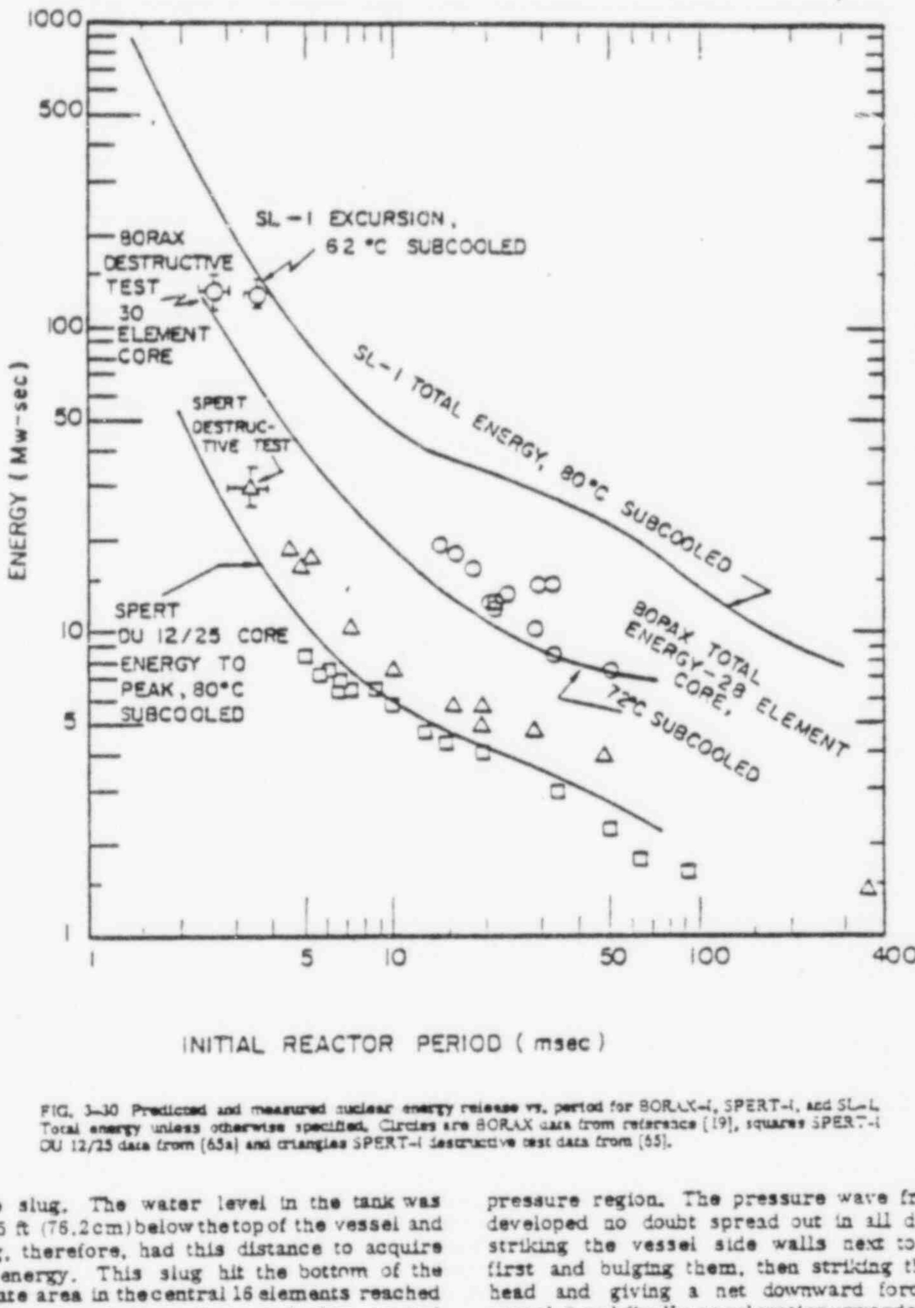

FIG. 3-30 Predicted and measured nuclear energy release vs. period for BORAX-I, SPERT-I, and SL-1. Total energy unless otherwise specified. Circles are BORAX data from reference [19], squares SPERT-I OU 12/25 data from [65a] and triangles SPERT-I destructive test data from [65].

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

Docketed
USNRC
JAN 17 1983
Office of the Sec.
Docketing & Ser.
Branch

APPLICATION FOR A CLASS 104 LICENSE

FOR A RESEARCH REACTOR FACILITY

Based on

Code of Federal Regulations, Title 10, Part 50

to

U.S. Nuclear Regulatory Commission

R. R. O'Neill, Dean
School of Engineering and Applied Science
University of California
Los Angeles

February 1980

TABLE III/6-1(a) COMPARISON TABLE - GENERAL

REACTOR LOCATION	UNIVERSITY OF CALIFORNIA, LOS ANGELES LOS ANGELES, CALIFORNIA 90024	UNIVERSITY OF FLORIDA GAINESVILLE, FLORIDA, 32611	UNIVERSITY OF WASHINGTON SEATTLE, WASHINGTON, 98195
LICENSE NO.	R-71	R-56	R-73
DOCKET NO.	50-142	50-83	50-139
OWNED BY	REGENTS OF THE UNIVERSITY OF CALIFORNIA	COLLEGE OF ENGINEERING	UNIVERSITY OF WASHINGTON
OPERATED BY	NUCLEAR ENERGY LABORATORY UCLA	DEPARTMENT OF NUCLEAR ENGR. SCIENCE, U. OF FLORIDA	NUCLEAR ENERGY DEPARTMENT COLLEGE OF ENGINEERING, U OF W
REACTOR TYPE	HETEROGENEOUS, THERMAL, LIGHT WATER COOLED & MODERATED, GRAPHITE REFLECTED, 93% ENRICHED URANIUM	*, EXCEPT ORIGINAL FUEL HAS 20% ENRICHED	*
DESIGNED BY	GENERAL NUCLEAR ENGR. (PRINCIPAL CONSULTANT TO REACTOR SYSTEM) STANTON & STOCKWELL, ARCHITECT AMF (REACTOR)	GENERAL NUCLEAR CORP., (REACTOR SYSTEM) G.C. FULTON, ARCHITECT TO THE STATE BOARD OF CONTROL (BUILDING) *	LOVETT, STREISSGRUTH & ZEMA ARCHITECT *
CONSTRUCTION BY	JONES BROTHERS CONST. CO. & LOUIS C. DUNN INC.-BUILDING A.I.-FUEL ELEMENTS (2ND TIME) HONEYWELL-CONTROLS, ELECTRONICS	COOPER CONSTRUCTION CO.-BUILDING AMF ATOMICS-REACTOR COMPONENTS SYLCOR NUCLEAR-FUEL ELEMENTS HONEYWELL-CONTROLS	JENTOFT & FORBES-CONTRACTOR THE MARTIN CO. (FUEL SET)
ORIGINAL DESIGNED POWER	100kw	*	*
NORMAL OPERATING POWER	100kw (1963)	* (1966)	* (1967)
NORMAL POWER (LINEAR, SPECIFIC POWER)	.19kw/FT, 23.12kw/kg U^{235}	*	*
OPERATING SCHEDULE	VARIABLE, 13-16 MEGAWATT HRS PER YEAR	VARIABLE, 15 TO 20 MW HRS PER YEAR	VARIABLE
PRINCIPAL USE OF REACTOR	ACTIVATION ANALYSIS, REACTOR OPERATOR TRAINING, AND EDUCATION OF NUCLEAR ENGR. & STUDENTS	TRAINING AND EDUCATION OF NUCLEAR ENGR. AND SCIENTISTS, REACTOR OPERATORS, ACTIVATION ANALYSIS & LASER EXP'S.	REACTOR OPERATOR TRAINING, CLASS DEMONSTRATIONS, ISOTOPE PRODUCTION, NAA RESEARCH
OPERATING STAFF	3 PER SHIFT, 1 SHIFT 4 PART TIME STUDENT OPERATORS	2 PER SHIFT, 1 SHIFT NORMALLY 2 PART TIME OPERATORS	2 PER SHIFT, 1 SHIFT
STATUS: DATE FIRST CRITICAL	10/60	1/59	4/61
DATE FULL POWER 10kw, 100kw	2/61 & 10/63	4/59, 3/62	4/61, 5/67
SOLID FUEL: FUEL ELEMENT A) SHAPE	FLAT PLATE (MTR TYPE)	*	*
B) FUEL COMPOSITION	13.4 WT.% U-AL ALLOY	*	*
C) FUEL DIMENSIONS (MEAT)	24" x 2" x .040"	*	23.5" x 2.3" x .040"
D) CLADDING MATERIAL	1100 ALUMINUM	*	*
E) CLADDING THICKNESS	.015 IN.	*	*
F) TYPE OF SUBASSEMBLY	PARALLEL PLATES BOLTED TOGETHER	*	*
G) # OF ELEMENTS PER ASSEMBLY	11	*	*
H) PLATE DIMENSIONS	25-5/8 x 2-7/8 x .070 IN.	*	*
I) SUB-ASSEMBLY (FHU) DIMENSIONS	27" (INCLUDES HANDLING RING) x 2-7/8" x 2-5/8"	*	*
J) # OF FHU'S	24	*	*
K) ARRANGEMENT OF SUB-ASSEMBLY	4 ELEMENTS IN EACH OF 6 BOXES ARRANGED IN 2 PARALLEL ROWS	*	*
L) LIFETIME	INDEFINITE	*	*
METHOD OF REFUELING	MANUAL, USING HAND TOOLS, TRANSFER CASK, REMOTE MIRROR	*	*

TABLE III/6-1(b) PRIMARY COOLANT

FLUID	H_2O	*	*
CIRCULATION A) DIRECTION OF FLOW	UPWARDS	*	*
B) FLOW INDUCED BY	FORCED FEED, GRAVITY RETURN	*	*
C) NORMAL FLOW RATE	16 GPM	40 GPM	23 GPM
D) MEAN VEL. THRU CORE	1.3 CM/SEC	4 CM/SEC	2.1 CM/SEC
E) AVERAGE INLET TEMP.	100°C	95°C	99°C
F) CORE ΔT	42°F	16.6°F	22°F
HEAT DISSIPATION METHOD	WATER-TO-WATER TUBE-SHELL S.S. HEAT EXCHANGER DOUBLE-PASS SECONDARY (TUBE)	*-U TUBE PRIMARY, SINGLE PASS SECONDARY-(1 MW RATED)	*
MEANS OF PURIFICATION	PRIMARY SYSTEM: CONTINUOUS ½ GPM BYPASS THRU CARTRIDGE FILTER THEN THRU DEMINERALIZERS AND RETURN TO DUMP TANK SHIELD TANK: INDEPENDENT, CONTINUOUS, 30 GPM, FILTER AND 2 PARALLEL DEMINERALIZER CARTRIDGES	*	*(900 ML/MIN)

NUCLEAR DATA

FUEL LOADING A) COLD CLEAN CRITICAL MASS	3.2 KG U^{235}	*	3.29 KG U^{235}
B) NORMAL FRESH FUEL LOADING	3.6 KG U^{235}	*	3.439 KG U^{235}
C) EXCESS K, FRESH LOADING	2.3% ΔK/K	*	*
FLUX @ 100KW A) PEAK THERMAL FLUX	1.5×10^{12} N/CM^2 SEC		1.4×10^{12} N/CM^2 SEC
B) PEAK FAST FLUX	1.5×10^{10} N/CM^2 SEC	*	5.0×10^{11} N/CM^2 SEC
C) PEAK THERMAL COLUMN THERMAL FLUX	1.0×10^{10} N/CM^2 SEC		3.0×10^{10} N/CM^2 SEC
REACTIVITY COEFFICIENTS (WATER) A) TEMP.	-0.48×10^{-2}% ΔK/K (−0.74¢/°F)	-1.0×10^{-4} ΔK/K/°C	−0.05% ΔK/K/°C (H_2O), +0.00014% ΔK/K/°F (GRAPHITE)
B) VOID	-0.164×10^{-2}% ΔK/K/%VOID (−25¢/% VOID)	−0.21 ΔK/K/% H_2O VOID	.0002% ΔK/K/CM^3 VOID
C) MASS COEFFICIENTS U^{235}	0.39% ΔK/K (16 GRAM)	*	0.01% K/K/GM U^{235}
D) CORE EXCESS	2.3% ΔK/K (3.5$)	*	
BURNABLE POISONS	NONE	*	*
NEUTRON SOURCE	6.6 MCI RABE	UP TO 5 CI Sb-Be AND 1 CI PuBe	Sb-Be (2 CI PuBe IF NEEDED)

TABLE III/6-1(c) REACTOR CHARACTERISTICS

ℓ_P	1.4×10^{-4} SEC		2.55×10^{-4} SEC (MEASURED)
ASSUMED ℓ_P	2×10^{-4} SEC		
τ	.035 SEC		
ASSUMED τ	.085 SEC		
β	~.0070		.0074
ASSUMED β_{EFF}	.0065		
BIOLOGICAL SHIELD	CONCRETE	CONCRETE	CONCRETE
FULL SCRAM (DROP ROD & WATER)	A) POWER FAILURE B) MANUAL (FULL SCRAM BUTTON) C) SHORT PERIOD (LESS THAN 5 SEC) D) HIGH FLUX (POWER > 125 KW) E) CLOSURES OPEN ABOVE 1 WATT F) DUMP VALVE OPEN	A) MANUAL SCRAM BAR B) 3 SECOND OR LONGER C) 1 OF 2 125% (125 KW) (2 SAFETY CHANNEL) D) NONE E) RESULTS IN LOW PC LEVEL, AND NO PC FLOW SCRAM. INTERLOCKS WITH PC PUMP. KEY TURNED OFF WITH 2 OR MORE BLADES UP = FULL SCRAM WITH WATER DUMP	A) POWER FAILURE B) MANUAL SCRAM C) SHORT PERIOD < 5 SEC D) HIGH FLUX = 125 KW E) OPEN DUMP VALVE
DROP ROD SCRAM	A) KEY TURNED OFF B) LOSS OF HIGH BAY VENTILATION C) LOSS OF PRIMARY PUMP POWER D) LOSS OF PRIMARY COOLANT FLOW E) LOW CORE WATER LEVEL F) LOW SHIELD TANK WATER LEVEL G) HIGH SECONDARY EFFLUENT RADIATION MONITOR (> 8 x 10^{-6} µCI/CL I-132 IN H_2O) H) DROP ROD BUTTON I) LOGIC CONDITION	A) IF ONLY 1 BLADE UP = MAG CLUTCHES OFF B) LOSS OF DILUTING OR VENT FAN POWER C) LOSS OF PC PUMP POWER D) PC FLOW DROPS TO 30 GPM FROM 40 GPM E) PC WATER LEVEL DROPS ABOUT 2 INCHES (STILL ABOVE FUEL PLATES) F) SMALL DROP - MOVABLE LEVEL SWITCH G) NO SCRAM. WEEKLY SAMPLE ANALYZED	A) TURN KEY OFF B) LOW FLOW PRIMARY C) LOW CORE LEVEL PRIMARY D) LOSS PRIMARY FLOW E) LOW SHIELD TANK WATER LEVEL F) DROP ROD BUTTON G) DILUTION FAN (AIR EXHAUST)
INHIBITS	A) NEUTRON START-UP SOURCE COUNT LESS THAN 2 CPS B) PERIOD LESS THAN 6 SEC C) CLOSURE OPEN (BELOW 1 W) D) LOG-N AMPLIFIER NOT IN OPERATE MODE	A) SAME AS SCRAM BAR B) LOSS OF 20 WATER ABOVE 1 KW. PC TEMP 150°F OR HIGHER C) 10% DROP IN NEUTRON CHAMBER HIGH VOLTAGE, EITHER OF 2 HV SUPPLIES D) 2 CPS INHIBIT OF UP DRIVE ONLY E) LOGIC - CANNOT WITHDRAW 2 OR MORE BLADES SIMULTANEOUSLY F) 10 SECOND PERIOD UP DRIVE G) NONE H) WIDE RANGE (LOG N + STARTUP) CHANNEL ALWAYS IN OPERATION. INHIBIT IF CALIBRATE OR TRIP TEST SWITCHES NOT IN OPERATE MODE (FOR UP DRIVE)	A) PERIOD < 10 SEC B) CORE INLET < 60°F C) BF_3 < 10 CTS/SEC
ALARMS (LIGHT & HORN)	A) HIGH PRIMARY COOLANT EXIT TEMPERATURE - 120°F B) HIGH AREA RADIATION > 5 MR/HR NORTH & SOUTH HIGH BAY > 10 MR/HR RADIOACTIVE STORAGE > 100 MR/HR RABBIT ROOM C) AIR ARGON 41 IN STACK > 1.3 x 10^{-5} µCI/ML IN AIR	A) 150°F = AUDIBLE ALARM B) ANY OF 5 AREA MONITORS = ALARM LIGHT @ 2.5 MR/HR = AUDIBLE ALARM & LIGHT @ 10 MR/HR. TWO CHANNELS @ 10 MR/HR ALSO = EVACUATION ALARM SIRENS. LOSS OF GREEN "NO FAIL" LIGHT IF CHANNEL MODULE NOT RECEIVING SIGNAL FROM AREA MONITOR CHAMBER C) 2 CUPS WATER IN PC PIT SUMP = AUDIBLE ALARM IN CONTROL ROOM, LIGHT D) ALARM LEVEL, STACK EFFLUENT ADJUSTABLE ACCORDING TO POWER OF OPERATION	A) HIGH PRIMARY EXIT SET 140 F CTS 160 F B) HIGH RADIATION AREAS REACTOR FLOOR > 1.5 MR/HR RABBIT EXIT > 10 MR/HR REACTOR TOP > 5 MR/HR HIGH ARGON 41 C) ANY SCRAM OR ROD DROP
ALARM LIGHT ONLY	A) LOW PRIMARY COOLANT RESISTIVITY < 1 x 10^6 OHMS B) BF3 DETECTOR IN HIGH FLUX > 0.022 WATTS C) ANY INHIBIT	A) LOSS OF POWER OR OPEN CIRCUIT IN EVACUATION SIREN = LOSS OF GREEN LIGHT B) PC SOLU BRIDGE MONITORS O/M INLET AND OUTLET RESISTIVITY ALARM LIGHT ADJUSTABLE ON INSTRUMENT C) AIR CONDITIONER TRIP BY SIREN = RED LIGHT D) B PROPORTIONAL COUNTER WHEN ENERGIZED HAS RED "EXTENDED RANGE" LIGHT. AUTOMATIC HV CUT OFF @ 400 CPS E) FAST PERIOD INHIBIT LIGHT & SWITCHES F) FAST POWER INHIBIT LIGHT & SWITCHES NOT IN OPERATE INHIBIT MODE	A) LOW PRIMARY CONDUCTIVITY < 500,000 OHM B) BF3 HIGH FLUX > 2 WATTS C) INHIBITS
		TWO ALARM LIGHTS FOR 20 WATER FLOW (140 GPM & 60 GPM) 60 GPM ALARM = SCRAM AT OR ABOVE 1% POWER AFTER 10 SECOND WARNING LIGHT B) LOSS OF LIGHTS = FAILURES IN RAD MON STANDBY BATTERY POWER PACK C) AUDIBLE ALARM & RED LIGHT IF BACK DOOR IS OPENED. RED AND GREEN LIGHTS MONITOR OUTER ACCESS DOORS TO FACILITY	

TABLE III/6-2 TRAINING REACTOR CHARACTERISTICS

DATE	1960	1980
TYPE	HETEROGENEOUS, THERMAL	•
POWER	10 KW	100 KW
FLUX LEVEL (AT 10 KW)	1×10^{11} n/cm^{2}sec	1.5×10^{12} n/cm^{2}sec
EXCESS REACTIVITY (TECH SPEC LIMIT)	0.58% ρ AT 50°F	2.3% ρ AT ROOM TEMP
EXCESS REACTIVITY INSTALLED	1.5% ρ AT ROOM TEMP	1.8% ρ AT ROOM TEMP
CLEAN COLD CRITICAL MASS	3200 GM U-235	•
EFFECTIVE PROMPT NEUTRON LIFETIME	1.4×10^{-4} SEC	2×10^{-4} SEC
UNIFORM WATER VOID COEFFICIENT	-0.19% ρ/% VOID	-0.164% ρ/% VOID
TEMPERATURE COEFFICIENT	-0.48×10^{-2}% ρ/°F	$-.401 \times 10^{-2}$% ρ
U-235 MASS COEFFICIENT	+0.31% ρ/% U-235 MASS	+.3% ρ/% U-235
START-UP SOURCE	2 CURIE Pu Be	6.6 M CI PA Be
REFLECTORS	GRAPHITE (1.67 GM/CC)	•
MODERATOR	H_2O AND GRAPHITE	•
DELAYED NEUTRON FRACTION	0.0068	.0075
FUEL PLATES		
FUEL	93% ENRICHED, U-AL ALLOY	•
FUEL LOADING	3,446.2 GM U-235	3,556 GM U-235
PLATE THICKNESS	0.070 IN.	•
WATER CHANNEL	0.137 IN.	•
ALUMINUM TO WATER RATIO (VOL)	0.51	•
MEAT COMPOSITION	13.4 WT% U-AL	•
COOLANT	H_2O	•
FLOW	10 GPM	16 GPM
TEMPERATURE, IN	103°F	100°F
TEMPERATURE, OUT	110°F	120°F
CONTROL BLADES	CD, SWINGING VANE, GRAVITY FALL	•
NUMBER	3 SAFETY: 1 REGULATION	•
INSERTION TIME	0.324 SEC (CALCULATED)	0.5 SEC (MEASURED)
REMOVAL TIME	90 SEC (MINIMUM)	100 SEC
BLADE WORTH, SAFETY	3 RODS 1.5% ρ = 4.5% ρ	3 RODS ~ 1.6% ρ = 4.8%
BLADE WORTH, REGULATING	1 ROD 0.6% ρ = 0.6% ρ	1 ROD ~ 1% ρ
	TOTAL = 5.1% ρ	TOTAL ~ 5.8% ρ
REACTIVITY ADDITION RATE, MAX.	0.025 ρ/SEC	.05% ρ/SEC
	PARRAFIN	
SHIELD (CONCRETE)		
SIDES, CENTER	6 FT. 0 IN. CAST, MAGNETITE	•
SIDES, SHIELD TANK END	6 FT. 8 IN. CAST, ORDINARY	•
SIDES, THERMAL COLUMN END	6 FT. 8 IN. CAST, MAGNETITE	•
MIDDLE	CAST CONCRETE BLOCKS	•
ABOVE CORE	5 FT. 10 IN. MAGNETITE BLOCKS	• PLUS 30" OF BORATED PARRAFIN
ENDS	3 FT. 4 IN. MAGNETITE BLOCKS	•
EXPERIMENTAL FACILITIES		
THERMAL COLUMN, HORIZONTAL	5 FT. x 5 FT. x 4 FT. 11 IN. LONG	60 IN. x 52 IN. x 43 IN. LONG REMOVABLE
THERMAL COLUMN, VERTICAL	PROVISION FOR INSTALLATION	•
SHIELD TEST TANK	5 FT. x 5 FT. x 14 FT. 6 IN. HIGH	•
EXPERIMENTAL HOLES	2 - HORIZONTAL, 6 IN. DIAMETER	•
	5 - HORIZONTAL, 4 IN. DIAMETER	4 - HORIZONTAL, 4 IN. DIAMETER
	3 VERTICAL, 1½ IN. DIAMETER	3 - VERTICAL, 1-7/8 IN. DIAMETER
EXPERIMENTAL HOLES, THERM. COL.	15 REMOVABLE GRAPHITE STRINGERS	•
FOIL SLOTS	11 - HORIZONTAL, 1/8 IN. x 1/2 IN.	•
	16 - VERTICAL, 3/8 IN. x 1 IN.	•

8.0 DESIGN BASIS, ACCIDENTS, AND CONSEQUENCES

Calculations pertinent to DBA's and Consequences were presented in the 1960 "Hazards Analysis" (Reference 1) that attended the original license application. These are reproduced in the original form and attached herewith as Attachment A and B.

Attachment A (titled Appendix B in "Hazards Analysis") treats with step changes of reactivity ($\Delta k/k \leq 0.006$) and limits the maximum licensed excess reactivity to 0.006 to avoid prompt criticality. Amendment 7, approved in 1966, increased the maximum allowable excess reactivity to 0.023 with the restriction that no single exposure cavity would contain an excess reactivity of 0.006. The excess reactivity limit of 0.023 was justified on the basis of SPERT and BORAX experimental results (see Attachment A). The provisions of Amendment 7 remain in effect today.

Attachment B (titled Appendix C in "Hazards Analysis") treats with Radiation Doses Resulting from Release of Fission Products into (the) Atmosphere. The release is not causally related to a specific accident, and from the SPERT and BORAX experiments, one can only state that the calculations attempt to suggest a scale of events that might follow a catastrophe of unknown cause. The calculation of fission product inventory is based upon a steady state equilibrium inventory at 10 kwt, and certain assumptions concerning leak rate from the building.

The consequential dose calculations were apparently unreviewed in the approval of Amendment 3 (1963) that increased the maximum licensed power level to 100 kwt. They were reviewed by the Division of Licensing and Regulation in processing the application for Amendment 7 (referred to above). In view of the current restriction of the UCLA Reactor operating hours to 5% of the year, the maximum average power is now 5 kwt, a factor of two less than the 10 kwt used in the original calculations.

Because the basis of the earlier calculations are not exceeded in the present application, those representations and actions are herewith incorporated in the Technical Specifications.

APPENDIX III

ARGONAUT SAFETY ANALYSIS REPORT (ASAR)

Attachment A

Estimation of Effects of

Assumed Large Reactivity Additions

from

UCLA TRAINING REACTOR HAZARDS ANALYSIS,
Final Report, R. D. MacLain
UCLA Department of Engineering
Report #60-18, UCLA-NEL 2
March, 1960 (there titled Appendix B)

ESTIMATION OF EFFECTS OF ASSUMED LARGE REACTIVITY ADDITIONS

It has been demonstrated repeatedly in the Borax and SPERT reactors that water-cooled, water-moderated reactors of suitable design may have a very substantial self-protection against the effects of reactivity accidents, even in the absence of corrective action by the reactor control system. This self-protection is provided by the negative steam-void coefficient of reactivity and the negative temperature coefficient of reactivity, both of which can result in important reactivity reductions as the reactor power rises. The UCTR has been designed with a high degree of self-protection of this type. In this appendix estimates are made of the behavior of the reactor under various hypothetical conditions of excess reactivity addition with no corrective action by the control system.

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. Its behavior can be predicted most reliably by utilizing the Borax I data with simple correction factors to convert them to the UCTR conditions.

The significant quantitative characteristics of the UCTR and the Borax I reactor are compared in Table B-1.

TABLE B-1

COMPARISON OF UCTR AND BORAX I CHARACTERISTICS

CHARACTERISTIC	UCTR	BORAX I
Fuel plate "meat"	13.4 w/o U-Al alloy 90% enriched	18 w/o U-Al alloy fully-enriched
Fuel plate cladding	1100 aluminum	1100 aluminum
"Meat" thickness	0.040 inch	0.020 inch
Cladding thickness	0.015 inch	0.020 inch
Coolant-channel thickness	0.137 inch	0.117 inch
Core volume (approx.)	71 liters	106 liters
Void coefficient of reactivity (calculated)	-0.18% k/% coolant void	-0.24% k/% coolant void
Temperature coefficient of reactivity (room temperature)	-0.009% k/°C (estimated)	-0.01% k/°C
Effective prompt-neutron lifetime	1.4×10^{-4} sec (calculated)	0.65×10^{-4} sec
Power ratio in core, maximum average	1.53	1.82

In addition to the quantitative differences, the UCTR differs from Borax I in that the maximum coolant water level is only a few inches above the upper ends of the fuel plates (instead of about 4 ft) and the coolant water, once it has been ejected forcibly from the core by a power excursion, cannot fall or flow back into the core.

Effect of 0.6% Excess Reactivity

An excess reactivity of 0.6% k_{eff} will be available in the reactor if its temperature is abnormally low (nearly freezing).

The addition of all this excess reactivity will cause the reactor to operate at a power such that the reactivity losses associated with the temperature increase and the voids formed will equal the initial excess reactivity.

If the reactivity is added slowly, after the reactor is critical, the power will approach such an equilibrium level slowly as the reactivity is added. If the reactivity is added suddenly, when the reactor is initially subcritical or at very low power, the power will at first rise exponentially with a period not shorter than 0.8 sec which is the asymptotic period corresponding to the full excess reactivity of 0.6% k_{eff}. Many experiments with the Borax reactors have demonstrated that for periods of this order of magnitude, the transition from the exponential power rise to the equilibrium power level (in which excess reactivity is balanced by temperature and steam void coefficients) is a smooth one involving little or no power overshoot. On the basis of this experience, it can be said that the magnitude of the power excursion which results from the 0.6% reactivity addition will not depend greatly on whether the reactivity is added suddenly or relatively slowly and in neither case will it approach a level which would cause a fuel plate to burn out.

In order to compute the power level at which the reactor will operate after the addition of the 0.6% excess reactivity discussed in the foregoing, it is necessary to know the water-temperature coefficient of reactivity. The relative importance of the two moderators, graphite and water, in determining the effective neutron temperature introduce uncertainties in the theoretical computation of this coefficient. The coefficient cannot, however, have an absolute magnitude less than that of the water-*density* coefficient of reactivity referred to a temperature scale, i.e., the coefficient computed on the assumption that:

$$\frac{d\,k_{eff}}{dT} = \frac{\delta\,k_{eff}}{\delta\rho}\cdot\frac{\delta\rho}{\delta T}$$

where ρ refers to the water density and T to the temperature. On the assumption that this minimum value is the true value, a rise of water temperature from near 0°C to 80°C would reduce reactivity by 0.6% k_{eff}.

The capacity of the reactor-coolant system is such that if the outside air temperature were 0°C and the average water temperature in the reactor were 80°C, energy would be removed at the rate of 365,000 BTU/hr or 107 kw. Under these conditions the reactor water-inlet temperature would be 60°C and the exit temperature, coincidentally, would be 100°C. It is, therefore, concluded that if the full available excess reactivity of 0.6% k_{eff} were added to the reactor on a cold day with the coolant system operating, the reactor would operate at an equilibrium power level about ten times higher (100 kw) than its normal maximum with little or no net steam production. Before reaching the equilibrium power, when the water in the coolant system would be heated to the equilibrium value, the reactor would operate at a somewhat higher power level and some net steam production might occur. If the coolant were not flowing during the time of excess reactivity addition, the equilibrium power level would be quite low and equal to the heat losses. In no case would the power level approach a high enough value to justify any fear of fuel-plate burnout.

Maximum Tolerable Sudden Reactivity Addition

In order to assess the safety factor which exists between the normal excess reactivity available in the reactor and the excess reactivity necessary for a serious power excursion, 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. Such an excursion would damage the reactor core but would not result in any substantial release of fission products.

The first step in the procedure is the estimation of the exponential period corresponding to the excess reactivity which would have characterized a power excursion of similar effect in Borax I. The estimate requires that (1) a relationship be established between the maximum temperature of the fuel plate and the energy release of the excursion and (2) the energy release be related to the period of the excursion.

For the case of power excursions of short period, with reactor water at saturation temperature, it is shown in Reference 1 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.* Measurements of the same type with cold reactor water (the case directly applicable to the UCTR) showed a similar relationship but with a proportionality constant of only about 10°F per MW-sec (Reference 3). The difference is not an unreasonable one since the subcooled water represents a more effective heat sink than the saturated water. However, the experiments with the saturated water were carried to short periods in the range of interest whereas the subcooled experiments were limited to longer periods. Therefore, more conservative saturated water data will be used.** To raise the maximum temperature of the fuel plate from the temperature of boiling water to the melting point of aluminum, a temperature change of approximately 1000°F, would require a power excursion with a total energy release of $\frac{1000°F}{24.4°F/MW\text{-sec}}$ or 41 MW-sec.

According to the data of Reference 3, replotted in Figure D-1a "subcooled" power excursion of reciprocal period 150 sec^{-1} would give an energy release of 41 MW-sec in addition to the energy necessary to raise the fuel plate temperature to the saturation temperature of water. It is, therefore, concluded that a power excursion of period at least as short as 1/150 sec (6.7 millisec) could have been tolerated by Borax I with subcooled water without melting at the hottest point in the fuel plates.

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 of the three related variables--neutron lifetime, excess reactivity, and exponential period--which characterize the neutron physics of a power excursion, it is the exponential period which determines the total energy release and the temperatures attained. The excess reactivity and neutron lifetime have large effects only as they jointly determine the period. This supposition is consistent, for example, with

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*Actually, the energy data of Reference 1 were revised in Reference 2 because of later and better calibrations of the instrumentation. The numbers above are taken from the later (more pessimistic) data.

**If subcooled data were used, the case directly applicable to UCTR, this analysis would indicate that step reactivity additions 2.4 times as large as those discussed here would not damage the reactor.

that the total energy transferred to the coolant water during a power excursion is many times the amount which would vaporize enough water to compensate for the excess reactivity, and that the actual reactivity reduction which occurs during the excursion is much larger than the initial excess reactivity. The extension of the Borax results to the UCTR is made on the basis of this evidence.

It is convenient, first, to treat only the effects of the slightly greater fuel-plate spacing and the slightly lower void coefficient of reactivity of the UCTR relative to the Borax I. Information will also be drawn from the Borax II experiments. The Borax II reactor differed from Borax I in that the coolant-channel thickness was greater in the ratio $\frac{0.264 \text{ in.}}{0.117 \text{ in.}} = 2.26$ and that the calculated void coefficient of reactivity was lower in the ratio.

$$\frac{0.10\% \ k_{eff}/\% \text{ void}}{0.24\% \ k_{eff}/\% \text{ void}} = \frac{1}{2.4} = 0.416$$

Both of these differences would be expected to cause a higher energy release per fuel plate in Borax II than in Borax I for a power excursion of given period. The measurements made with subcooled water at periods down to 23 millisec showed that the energy release per fuel plate in Borax II was between 1.7 and 2.0 times that of Borax I, with the smaller ratio applying to the shorter periods (Reference 2). Therefore, it seems quite conservative to assume, in the case of any two reactors, (1) and (2), of the Borax type having a ratio of fuel plate spacings, S_1/S_2, and a ratio of void coefficients of reactivity, C_1/C_2, that the ratio of energy release per fuel plate for a subcooled power excursion of given period will be no greater than $E_2/E_1 = S_2/S_1$ or $E_2/E_1 = C_1/C_2$ whichever is the larger. For the UCTR and Borax I the ratios are

$$\frac{S_{UF}}{S_{Bo}} = \frac{0.137}{0.117} = 1.17 \quad \frac{C_{Bo}}{C_{UF}} = \frac{0.24}{0.18} = 1.33$$

It is concluded, therefore, that a Borax reactor having a coolant-channel thickness and a void coefficient of reactivity equal to those of the UCTR would release not more than 1.33 times as much energy per fuel plate as Borax I. The limiting nonmelting period for such a reactor would be that which in Borax I gave an energy release of 41/1.33 = 31MW-sec. The period obtained from Figure B-1, corresponding to a total energy release of 31MW-sec, is 8.3 millisec.

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 ecause a large fraction of the total energy released is stored in the fuel plate during the important stage of the reactor shut-down. For example, a reactor composed of fuel plates of high heat capacity undoubtedly will experience a larger total energy release, but not necessarily a higher maximum temperature, during a power excursion of given period, than a reactor having plates of low heat capacity.

From examination of the Borax results, it seems clear that two distinct phases of the reactor shut-down process occur consecutively and that both may be important in determining the maximum center temperature of a fuel plate. The first phase covers the interval before an important amount of boiling occurs at the fuel-plate surface. During this interval, the heat loss to the water is small and the important consideration is evidently the ratio of fuel-plate surface temperature (which determines the start of boiling) to center temperature. For periods in the range under consideration, this temperature ratio is theoretically not far from unity (0.76 minimum for a 10-millisec period in Borax I). Experimentally, the temperature ratio was unity for periods down to 5 millisec in the Borax I measurements. Since the total effect is small and since the temperature ratio for Borax and UCTR fuel plates should not be much different, the thinner cladding will tend to balance the effect of the poorer "meat" conductivity. It is concluded, therefore, that there will be no important difference in fuel plate performance during this initial phase of the excursion.

The second phase of the power excursion begins when a significant rate of boiling is established at the plate surface. Reactivity and consequently generation are reduced at a rate which must be a function of the rate at which heat can be transferred into the boiling water. At the same time, the transfer of heat into the water removes heat from the fuel plate and limits its temperature rise. The important characteristic of the plate during this phase of the excursion is the heat flux which it can supply to the water for a given temperature difference between the plate center and surface. A figure assumed to be roughly indicative of the relative performance or merit of fuel plates during this phase is the ratio of heat flux to temperature difference under steady-state conditions. This ratio (figure of merit) will overemphasize the difference between fuel plates since the temperature distribution in the plate will be more peaked during a steady-state conduction than during conduction when the general temperature level is rising. The ratio of these figures of merit for Borax I and for the UCTR is

$$\frac{\left[\dfrac{\text{Heat Flux}}{\Delta T_{c-s}}\right]_{\text{UCTR}}}{\left[\dfrac{\text{Heat Flux}}{\Delta T_{c-s}}\right]_{\text{Borax}}} = 0.82$$

A conservative procedure would be to apply the above factor to the permissible total energy of excursion on the Borax I curve. At the same time, however, the difference in gross maximum to average power ratio for the two reactors should be taken into account since it is the temperature of the hottest point in the hottest fuel plate which is being considered. The power ratio for the two reactors is

$$\frac{\dfrac{\text{Max}}{\text{Ave}}_{\text{Borax}}}{\dfrac{\text{Max}}{\text{Ave}}_{\text{UCTR}}} = \frac{1.82}{1.63} = 1.12$$

The combination of these two factors reduces the permissible equivalent energy of the Borax-type excursion to

$$31 \times 0.82 \times 1.12 = 28.4 \text{ MW-sec}$$

The corresponding exponential period from Figure B-1 is 9.1 millisec. It is, therefore, concluded that the UCTR will tolerate a power excursion of period at least as short as 9.1 millisec without the melting of any part of any fuel plate. The excess reactivity corresponding to this period is 2.3% k_{eff}.

Successive Power Excursions

It is typical of the Borax and SPERT reactors, unless the excess reactivity is removed by external means, that an initial power excursion which terminates itself by expelling water from the reactor core will be followed by subsequent excursions as the water falls and flows back into the core. An exception to this behavior occurs when the initial excursion is violent enough to cause a permanent loss of reactivity by throwing a large amount of water completely out of the reactor tank. In the UCTR the total quantity of water in the core is small, the submergence of the core is small, and baffles above the core are so arranged that any water splash is directed to the outside so that it cannot return to the core. Consequently, even a relatively mild power excursion (e.g., one having an exponential period of from 20 to 30 millisec) in the UCTR should result in permanent self-induced shutdown of the reactor. By these same design features, the possibility of large successive power excursions, such as those studied in the SPERT project, resulting from the ramp addition of excess reactivity is eliminated. It can be anticipated that the UCTR will be safe against quite large ramp additions (larger than 2.3% k_{eff}) provided only that the ramp rate is not so rapid as to add an excess reactivity of more than 2.3% k_{eff} before the reactor power reaches a high level. To exceed this limit the ramp rate would need to be of the order of 1.0% k_{eff} per second or larger.

Beam Tube Reactivity Effects

The UCTR has two 6-inch diameter beam tubes which extend to within 11 inches of the fuel-graphite interfaces. The maximum change on the core reactivity which can be effected by these two beam-tube facilities was calculated to be 0.18% $\Delta K/K$ or 0.09% $\Delta K/K$ per beam tube. The calculation is based upon the effect of a black absorber six inches in diameter placed in the same position as the beam tubes. The reduction in the reflector savings due to the black absorber was calculated using the following equation.

$$\delta \text{ (reflector savings)} = \frac{D(\text{core})}{D(\text{reflector})} \cdot L\text{ (reflector)} \cdot \tanh \frac{T(\text{reflector thickness})}{L(\text{reflector})}$$

Reference: *Elements of Nuclear Reactor Theory*, Glasstone and Edlund.

The reflector savings for the 49.5 cm and 28.0 cm of graphite were calculated to be 7.83 cm and 5.26 cm respectively. The area of the black absorber is 13% of the adjacent core face area. Using the reflector savings given above and the area weighting factors, the reflector savings with and without the six-inch diameter black absorber were calculated to be 7.33 cm and 7.83 cm respectively.

The reactivity effect of the single six-inch diameter black absorber was then determined calculating the critical buckling with and without the black absorber.

Using the value of -0.09% $\Delta K/K$, for a single beam tube the shortest period which the reactor could go on, due to the sudden withdrawal of a black absorber from the six-inch beam tube, would be approximately 80 seconds. Therefore, the reactivity change which can be effected by the beam tubes does not represent a hazard to reactor operation.

In addition to the two 6-inch beam tubes which penetrate the outer reflector, there are four 4-inch beam tubes which terminate outside of the reflector. No calculations were made for the 4-inch tubes since their effect on reactivity will be much smaller than that of the 6-inch tubes.

REFERENCES

1. Dietrich, J. R. and D. C. Layman, *Transient and Steady State Characteristics of a Boiling Reactor. The Borax Experiments, 1953*, AECD-3840, Argonne National Laboratory, February, 1954.

2. Dietrich, J. R., *Experimental Determinations of the Self-Regulation and Safety of Operating Water-Moderated Reactors*, A/Conf. 8/P/481, International Conference on the Peaceful Uses of Atomic Energy, June 30, 1955.

3. Dietrich, J. R., *Experimental Investigation of the Self-Limitation of Power During Reactivity Transients in a Subcooled, Water-Moderated Reactor. Borax-I Experiments, 1954*, AECD-3668, Argonne National Laboratory, August 17, 1955.

4. Lennox, D. H. and C. N. Kelber, *Summary Report on the Hazards of the Argonaut Reactor*, ANL-5647, December 1956.

APPENDIX III

ARGONAUT SAFETY ANALYSIS REPORT (ASAR)

Attachment B

Radiation Doses Resulting From

Release of Fission Products into Atmosphere

from

UCLA TRAINING REACTOR
HAZARDS ANALYSIS, Final Report.
R.D. MacLain, UCLA Department of Engineering
Report #60-18, UCLA-NEL 2
March, 1960 (there titled Appendix C)

RADIATION DOSES RESULTING FROM RELEASE OF FISSION PRODUCTS INTO ATMOSPHERE

Estimates have been made of the radiation dosages which would be received by persons outside the reactor building should there be a release of reactor fission products into the reactor building and leakage of the building air to the outside. The radiation exposures considered here are those which would result from the passage of the air-borne cloud of radioactive contaminants over the ground. These include the external beta and gamma radiation exposures and the internal exposure of critical body organs resulting from inhalation of the air-borne contaminants. The most important of the internal exposures are the iodine dose to the thyroid and the strontium dose to the bones.

The radiation exposure received by a person standing at a given distance from the reactor building obviously depends on such factors as (a) curies of fission products stored within the core at the time of release, (b) fraction of the core fission products escaping into the building air, (c) building out-leakage rate, and (d) atmospheric dispersive properties. Hence, in the analysis, certain basic assumptions are required as to the circumstances surrounding the release of the fission products, as to atmospheric conditions and, as to the tightness of the building at the time of release. The results obtained here are based on assumptions which, except for the arbitrary one that a release has occurred, are considered reasonable for the reactor and building design. The calculation method is described and illustrated in sufficient detail that additional calculations based on other assumptions can be made if desired.

The material presented here is divided into three sections. The first section describes the model assumed for the release and spread of radioactivity and gives the necessary references and formulae used in calculating the radiation doses. The second section illustrates the calculation procecure. The third section presents the results obtained for the radiation exposure hazards with the assumed model.

Method and Assumptions used in Dose Calculations

Although such an event is not considered even plausible because of the limitations on available excess reactivity and because of the inherent self-limiting characteristics of the reactor, it is postulated that an accident has occurred in which the reactor power level has risen to the extent that local melting of the fuel plates has occurred. The reactor is assumed to have been operated continuously at the 10 kw power level long enough to have attained equilibrium concentrations of the relatively short-lived fission products, i.e., the iodine, bromine, and krypton isotopes. The incident is assumed to result in the transfer of 10% of the volatile fission products from the reactor fuel plates to the building air. It is assumed further that none of the nonvolatile fission products are transferred to the building air although they may be released to the reactor coolant water and retained within the reactor building.

The foregoing set of circumstances is consistent with the reasonable assumption made here that the incident is not violent enough to blow off the top and side biological shields so as to cause an intense spray of water-steam-radioactivity mixture into the building air. The release of 10% of the volatile fission products is probably too high for the assumed incident but is used to give an upper limit to the radiation exposure involved. The volatile fission products are bromine, krypton, iodine, and xenon. Hence, the fission product chains which must be considered are of atomic masses 82 to 90 and 131 to 135. Reference 1 presents tables for each chain giving the equilibrium activity of each of the fission product isotopes in the chain and the decay (or buildup) following reactor shut-down. The information presented in Reference 1 is used here for the fission product activity release into the building air following the assumed incident.

The most likely point at which radioactive contamination of the room air would be detected is in the reactor room exhaust duct, since the air is pulled from the reactor room, and exhausted through the fan room atop the building. The air would not be considered contaminated until the activity exceeds that associated with the A^{41} normally being discharged. Upon detection of radioactivity the air conditioning unit will be shut off, and the dampers in the inlet and outlet ventilation will be closed. (The major avenues of leakage of the volatile fission products and daughter nongaseous products) from the reactor room are the two high bay room entrance doors from the control room and the three emergency exits in the reactor room. Access to the control room is by way of an electrically controlled door from the reception area in Engineering Unit III. All access doors will be weather-stripped and emergency doors leading directly to the outside, caulked and sealed for minimum leakage. These doors will be closed at all times during reactor operation and any breeching will be indicated in the control room on an audiovisual alarm system.

To obtain an upper bound for the radiation doses, the outleakage, L, in curies per hour of fission product activity, is obtained from

$$L_i = \frac{B}{V_g} C_{i,max} \tag{1}$$

and is assumed constant during the exposure time. In Equation (1), $\frac{B}{V_g}$, the building leakage rate, has been taken as 20% of the reactor room volume, B, per hour for a 30 MPH wind. This leak-rate value is assumed to be directly proportional to wind velocity. The quantity $C_{i,max}$ is the maximum activity in curies of isotope ,i, present outside the fuel plates following the assumed release of 10% of the volatile fission products. For most of the isotopes in the volatile fission product chains, $C_{i,max}$ is the activity of the isotope at the time of release from the fuel plates. The important exceptions are Sr-89 and Sr-90 and are formed outside the reactor and reach a maximum activity outside the reactor at some time after the fission product release. The tables in Reference 1 permit easy calculation of the activities of Sr-89 and Sr-90 attributable only to the decay of the isolated parent products.

The concentration of fission product activity in the atmosphere outside the reactor building and the resultant radiation exposure will depend on the wind direction and velocity and the degree of atmospheric turbulence. The highest dose rate is obtained when the person exposed is directly downwind from the leak. The computation method is based on O. G. Sutton's formula and utilizes equations and curves given in Reference 3.

For calculation of the external beta dose and inhalation doses from the radioactive iodines and strontiums, the concentration of activity in the atmospheric air was calculated by the formula in Reference 3, Page 153. For ground-level continuous emission of radioactivity, this formula reduces to

$$X = \frac{2L}{3600\,\pi C^2\, u\, x^{2-n}} \tag{2}$$

where,

X = concentration of activity, curies per cubic meter of air

L = continuous source strength, i.e., building out-leakage rate in curies per hr

x = distance downwind from source, meters
u = mean wind speed, meters per second
C = generalized diffusion coefficient, meters n/2
n = dimensionless parameter associated with atmospheric stability

The following representative values of the diffusion parameters for two different atmospheric conditions are used to calculate the concentration of activity, X, for a specified leak rate and atmospheric condition in the outside air for various distances, x, from the leakage source.

Atmospheric Condition	n	C^2	u
severe inversion	0.5	.008	1
mild lapse	0.25	.024	3

The external beta dose rate during passage of the cloud of radioactive fission products is obtained from the following equation given in Reference 3, page 100.

$$D_\beta = (0.5)\ (0.64)\ \frac{1r}{6.8 \times 10^{10}\ \frac{Mev}{m^3}}\ X_\beta \qquad (3)$$

where X_β is the concentration of β-energy in Mev per sec per cubic meter of air and D_β is the external beta dose rate in roentgens per sec. The relation between X_β and X of Equation (2) is

$$X_\beta = 3.7 \times 10^{10}\ X\ E \qquad (4)$$

where E is the effective beta energy in Mev per disintegration.

The activity A deposited per second in the critical organs is given by

$$A = J\ F_o\ X \qquad (5)$$

where

A = activity deposited in organ, millicure/sec
J = inhalation rate, 17/60 liters per sec
F_o = inhaled fraction of activity retained in critical organ

The corresponding initial internal dose rate for a person standing in the fission product stream is given by the expression

$$D = A\ 3600\ t\ \frac{62\ E}{W} \qquad (6)$$

where,

D = initial internal dose rate rep/day
t = time of exposure, hr
W = weight of critical organ, kg

The total integrated dose to the critical organ is related to the initial internal dose rate by the equation

$$TID = D\,1.44\,T \tag{7}$$

where,

TID = total integrated dose, reps
T = effective half-life of the radioisotope, days.

The values of F_o, E, W, and T appearing in Equations (5), (6), and (7) may be obtained from Reference 4 for the various radioisotopes and critical organs involved. Reference 5 gives additional information on the various iodine isotopes.

For calculation of the external gamma dose rate, the J. Z. Holland nomogram given as Figure 8.3, Reference 3 is used. The nomogram gives the gamma dosage resulting from sudden discharge into the atmosphere of the contents of a nuclear reactor which has been operating at a steady power level. The dosage read from the nomogram first must be corrected to account for the fact that none of the nonvolatile and only a fraction of the volatile fission products are assumed to escape from the reactor for the case being considered. Also, since the activity is not immediately released into the atmosphere, but leaks out of the building at a finite rate, the dosage obtained by use of the nomogram must be converted to dose rate. The corrections (actually scaling) applied to the values obtained from the nomogram were calculated as follows:

The gamma activity of the volatile fission products assumed to be escaping from the reactor was determined by use of the curves given in Reference 6 for 20 min time after shut-down. Equation (8.5) of Reference 3 is used to calculate the gamma activity of all the fission products for the same time after shut-down and the same reactor power level (10 kw). From this the fraction of the total fission product gamma activity attributed to the assumed 10% escape of the volatile fission products is determined and the gamma dose read from the nomogram is scaled down by this fraction. To obtain the dose rate resulting from the finite rate of radioactivity leak into the atmosphere, the gamma dose scaled from the nomogram multiplied by the quantity B/T_g.

Illustrative Calculations

Problem 1: Calculate the I-131 dose to the thyroid of a person standing at a distance of 61 meters downwind of the leak for eight hours. During the exposure a severe inversion condition exists in the atmosphere.

Solution: From Reference 1, the equilibrium curies of I-131 in the reactor fuel plates following 10 kw operation is

$$\frac{9.307 \times 10^{18} \times .003610}{1.36 \times 10^{14}} = 247 \text{ curies}$$

For a 30 MPH wind, $B/V_g = 0.20$. For a severe inversion u = 1 meter/sec (2.25 MPH) so that $B/V_g = 0.20 \times \frac{2.25}{30} = .015$. For 10% release of I-131 from the reactor,

$$L = .015 \times (247 \times .10) = 0.37 \text{ curies/hr}$$

From Equation (2)

$$\chi = \frac{2 \times 0.37}{3600\pi \times .008 \times 1 \times 61^{1.5}} = 1.7 \times 10^{-5}\ \frac{\text{curies}}{\text{meter}^3}$$

The thyroid dose from I-131 is calculated from Equations (5), (6), and (7)

$$A = 17/60 \times .15 \times 1.7 \times 10^{-5} = 7.2 \times 10^{-7}\ \text{millicurie/sec}$$

$$D = 7.2 \times 10^{-7} \times 3600 \times 8 \times \frac{62 \times .22}{.020} = 14\ \text{rep/day}$$

$$TID = 14 \times 1.44 \times 7.7 = 155\ \text{rep}$$

The values of F_3, E, W, and T used in the calculations were obtained from Reference 4.

__Problem 2:__ Calculate the external β-dose from the I-131 isotope for a person standing at a distance of 61 meters downwind of the leak for eight hours.

__Solution:__ For $\chi = 1.7 \times 10^{-5}$ curies/m^3 obtained in Problem 1,

$$\chi_\beta = 3.7 \times 10^{10} \times 1.7 \times 10^{-5} \times 0.22 = 1.4 \times 10^5\ \text{Mev/sec m}^3.$$

From Equation (3)

$$D_\beta = 0.5 \times 0.64 \times \frac{1.4 \times 10^5}{6.8 \times 10^{10}} = 6.6 \times 10^{-7}\ \text{r/sec}$$

For eight-hour exposure,

$TID = 6.6 \times 10^{-7} \times 3600 \times 8 = .019\ r = 19$ mr attributed to only the I-131 isotope.

To obtain the total external β-dose, the same procedure must be followed for all of the fission products assumed to be escaping from the reactor. For the conditions of this problem, the air concentration of all of the fission products assumed to be escaping from the reactor is 1.2×10^7 Mev/sec m^3, in which case the total external beta dose for eight-hour exposure is 1.6. Because the decay of the fission products in the building and en-route to the person outside the building was neglected, the dose value calculated is higher than the actual value which would be obtained for the assumed conditions.

__Problem 3:__ Calculate the total gamma dose to a person standing 61 meters downwind of the leak for eight hours.

__Solution:__ Direct use of the nomogram (Reference 3) gives 12r for the total gamma dose caused by sudden release of the total contents of the core into the atmosphere. From Reference 6, the gamma activity of 10% of the volatile fission products is 6.9×10^{12} Mev/sec at 20 minutes after shut-down. For all of the fission products at 20 minutes after shut-down, Equation (8.5) of Reference 3 gives a total gamma activity of 5.2×10^{14} Mev/sec. Hence, on the average, 10% of the volatile products gives a gamma activity equal to .01324 of the activity of all the fission products. Thus, the dose due to sudden release of 10% of the volatile products into the atmosphere is

$$12 \times .01325 = .16\ r = 160\ mr$$

Inasmuch as the contaminants leak out from the building at a finite rate ($B/V_g = .015\ hr^{-1}$), the gamma dose rate is simply

$$160 \times .015 = 2.4\ mr/hr$$

and for an eight-hour exposure, the accumulated gamma dose is

$$2.4 \times 8 = 19.2\ mr$$

Results of Radiation Exposure Calculations

The results for eight-hour exposure at four different distances downwind of the point of release under two different atmospheric conditions, calculated as illustrated above, are tabulated below. In all cases, 10% of only the volatile fission products are assumed to be released from the reactor fuel plates. The building leakage rates are $B/V_g = .015\ hr^{-1}$ for the severe-inversion condition and $B/V_g = .045\ hr^{-1}$ for the mild-lapse condition (caused by difference in wind speeds assumed to be prevailing for the different atmospheric conditions).

TOTAL INTEGRATED DOSE (rep) FROM AN EIGHT-HOUR EXPOSURE AT VARIOUS DISTANCES DOWNWIND FROM REACTOR BUILDING LEAK

x, meters	External Beta Dose	Gamma Dose	Thyroid Dose	Bone Dose
Severe Inversion				
15	14.0	.080	1800	.006
61	1.6	.019	220	.0007
152	0.4	.010	59	.0002
305	0.15	.005	20	---
Mild Lapse				
15	2.2	.040	290	.001
61	0.19	.007	26	.0001
152	0.04	.004	6	---
305	0.012	.002	2	---

REFERENCES

1. Faller, I. L., T. S. Chapman and J. M. West, *Calculations on U-235 Fission Product Decay Chains*, ANL-4807, Argonne National Laboratory.

2. *Heating Ventilating Air Conditioning Guide*, American Society of Heating and Ventilating Engineers, 1958.

3. *Meteorology and Atomic Energy*, U. S. Department of Commerce, Weather Bureau, July, 1955

4. *Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water*, National Bureau of Standards, Handbook 52.

5. Dunning, G. M., *Thyroid Dose from Radioiodine in Fallout, Nucleonics*, Vol. 14, p. 40, February, 1956.

6. Clark, F. H., *Decay of Fission Product Gammas*, NDA-27-39, Nuclear Development Associates, Inc., December, 1954.

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MARCH 1960

DEPARTMENT OF ENGINEERING

ucla training reactor hazards analysis

FINAL REPORT

R.D. MAC LAIN

REPORT NO. 60-18

UNIVERSITY OF CALIFORNIA, LOS ANGELES

Report No. 60-18
March 1, 1960

UCLA TRAINING REACTOR
HAZARDS ANALYSIS

FINAL REPORT

80-213-478-9711
x 7109

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INTRODUCTION

This report has been prepared for submission to the U. S. Atomic Energy Commission as part of the facility license application for a nuclear training reactor which is being constructed by the University of California at Los Angeles.

The UCLA reactor is similar to the training reactor at the University of Florida at Gainesville, Florida, and reported in *University of Florida Training Reactor Hazards Summary Report*, J. M. Duncan, Bulletin Series, No. 99, Vol. XII, No. 10, Florida Engineering and Industrial Experiment Station. The only significant difference between the two reactors is the fuel enrichment. The UCLA reactor uses 90% enriched fuel as opposed to the 20% fuel used by the UFTR.

SECTION I

A. REACTOR SITE

The reactor is located in a newly-constructed, permanent, reinforced concrete building on the campus of the University of California at Los Angeles. The location of the building and its relationship to its surroundings is indicated in Figure I-1.

The normal building populations during a school day are given in Figure I-2. At times other than school days, campus building populations are reduced to small fractions of the figures shown.

The 400-acre campus is located on coastal plain approximately five miles east of the Pacific Ocean and 15 miles west of the Los Angeles civic center. To the south of the campus is a business and shopping district, and to the north, west and east are residential areas. A map of this general area is given in Figure I-3.

Geology

The UCLA campus is situated on a coastal plain, and is approximately 400 feet above sea level. The coastal plain consists of a terraced alluvial fill, 200 feet deep at the reactor site, overlying sedimentary rock of rather recent origin. The coastal plain lies at the base of the Santa Monica Mountains which are 2000 feet high. The most important formation in these mountains is Santa Monica slate, an old sedimentary layer 2000 feet thick. Overlying this slate stratum are several more recent sedimentary layers. A cross section of the coastal plain near the campus is given in Figure I-4. This section is at right angles to the anticlinal folding of the Santa Monica Mountains.

Hydrology

No deep wells have been drilled on the campus of UCLA or in the vicinity of the campus. The water table is estimated to lie 200 feet below the surface in this area. A log of a typical test well made by a foundation engineer near the site of the reactor building is shown in Figure I-5.

Surface runoff water is collected in concrete-lined storm drains which empty into the ocean. This drainage system has been adequate to prevent any flooding of the campus by heavy winter rains. The maximum rainfall in any 24-hour period during the last 75 years was ten inches, as indicated in Figure I-6. It is barely conceivable that runoff from the watershed area north of the campus could flood Westwood Boulevard and the area to the west of the reactor site. However, the reactor core lies about ten feet above this level, and a rainfall equal to the largest ever recorded would not flood the reactor. In the unlikely event that such flooding should occur, it would pose an extreme operational inconvenience, but would not create any radiation hazard.

Seismology

Southern California is seismically active. The locations of known active faults are indicated in Figure I-7. The nearest of these to the reactor site is the Inglewood fault running in a north-westerly direction about two miles east of the campus. In Southern California, the region from the Mojave Desert to beyond the off-shore islands is traversed by a series of active faults. These faults extend from 20 to 50 to many hundreds of miles in length, and the trend is generally between north and west. However, they are only roughly parallel, and in certain instances a major fault zone divided into two or more well defined faults. In general, these faults are from five to twenty miles apart and apparently extend to depths of 15 or more miles below the surface.

SECTION II

TRAINING REACTOR DESCRIPTION

Introduction

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. These requirements are met in this reactor by combining a water-moderated, plate-type fuel section with a graphite system for maintaining a fixed geometrical arrangement.

There is no credible way in which the fission products of this reactor can be made to escape, and the amount of contained fission products will be relatively small since it is limited to a maximum steady state power of ten kilowatts. Nevertheless, because of the reactor location on the campus environs, it is housed in a structure with a minimum number of penetrations sealed against gas leakage.

A. BASIC DESIGN FEATURES

This reactor is of the same general type as the Argonaut Reactor and similar to the University of Florida Training Reactor.* The basic element of the reactor is a rectangular prism (5 x 5 x 9 feet) constructed of graphite bars. The fissionable material is introduced into the graphite prism in the form of aluminum-uranium alloy plates in six aluminum boxes, each of which contains a small amount of water. The object of this construction is to have the convenience of a solid moderator-reflector and the safety of the water plate arrangement. The convenience of the structure is illustrated by the fact that three large, plane neutron sources are obtained. These plane neutron sources can be converted to thermal columns merely by adding sufficient graphite, or one of them can be used in shielding and other experiments merely by placing the structure to be tested against the plane source. The upward direction can be used conveniently as a neutron source for exponential experiments. The distribution of weight on the reactor is a minor problem because the reflector structure is solid graphite.

The 90% enriched uranium-bearing plates are immersed in sufficient water so that a power excursion which would eject the water from the aluminum boxes certainly would reduce the reactor to below criticality. Since the reactor is to be operated at a maximum of 10 kw, only a small amount of reactivity is required for the temperature coefficient and xenon poisoning. Thus, it is possible to operate the reactor with an amount of excess reactivity which is well below that required for prompt criticality. Under these conditions, the reactor meets the safety requirements of a training reactor and can tolerate considerable operational error without damage.

The fuel is contained in MTR type plates assembled in bundles. These fuel bundles are contained in six watertight aluminum boxes set in a two-slab array in a 5-ft prism of graphite bars. The control rods are the swinging-arm type similar to those used on CP-3 and CP-5, and University of Florida reactors. Four cadmium vanes protected by magnesium shrouds operate within the spaces between the fuel boxes. These are moved in

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*_University of Florida Training Reactor Hazards Summary Report_, J. M. Duncan, Bulletin Series No. 99, Vol. XII, No. 10, Florida Engineering and Industrial Experiment Station.

APPENDIX B

ESTIMATION OF EFFECTS OF ASSUMED LARGE REACTIVITY ADDITIONS

It has been demonstrated repeatedly in the Borax and SPERT reactors that water-cooled, water-moderated reactors of suitable design may have a very substantial self-protection against the effects of reactivity accidents, even in the absence of corrective action by the reactor control system. This self-protection is provided by the negative steam-void coefficient of reactivity and the negative temperature coefficient of reactivity, both of which can result in important reactivity reductions as the reactor power rises. The UCTR has been designed with a high degree of self-protection of this type. In this appendix estimates are made of the behavior of the reactor under various hypothetical conditions of excess reactivity addition with no corrective action by the control system.

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. Its behavior can be predicted most reliably by utilizing the Borax I data with simple correction factors to convert them to the UCTR conditions.

The significant quantitative characteristics of the UCTR and the Borax I reactor are compared in Table B-1.

TABLE B-1

COMPARISON OF UCTR AND BORAX I CHARACTERISTICS

CHARACTERISTIC	UCTR	BORAX I
Fuel plate "meat"	13.4 w/o U-Al alloy 90% enriched	18 w/o U-Al alloy fully-enriched
Fuel plate cladding	1100 aluminum	1100 aluminum
"Meat" thickness	0.040 inch	0.020 inch
Cladding thickness	0.015 inch	0.020 inch
Coolant-channel thickness	0.137 inch	0.117 inch
Core volume (approx.)	71 liters	106 liters
Void coefficient of reactivity (calculated)	-0.18% k/% coolant void	-0.24% k/% coolant void
Temperature coefficient of reactivity (room temperature)	-0.009% k/°C (estimated)	-0.01% k/°C
Effective prompt-neutron lifetime	1.4×10^{-4} sec (calculated)	0.65×10^{-4} sec
Power ratio in core, maximum average	1.63	1.82

In addition to the quantitative differences, the UCTR differs from Borax I in that the maximum coolant water level is only a few inches above the upper ends of the fuel plates (instead of about 4 ft) and the coolant water, once it has been ejected forcibly from the core by a power excursion, cannot fall or flow back into the core.

Effect of 0.6% Excess Reactivity

An excess reactivity of 0.6% k_{eff} will be available in the reactor if its temperature is abnormally low (nearly freezing).

The addition of all this excess reactivity will cause the reactor to operate at a power such that the reactivity losses associated with the temperature increase and the voids formed will equal the initial excess reactivity.

If the reactivity is added slowly, after the reactor is critical, the power will approach such an equilibrium level slowly as the reactivity is added. If the reactivity is added suddenly, when the reactor is initially subcritical or at very low power, the power will at first rise exponentially with a period not shorter than 0.8 sec which is the asymptotic period corresponding to the full excess reactivity of 0.6% k_{eff}. Many experiments with the Borax reactors have demonstrated that for periods of this order of magnitude, the transition from the exponential power rise to the equilibrium power level (in which excess reactivity is balanced by temperature and steam void coefficients) is a smooth one involving little or no power overshoot. On the basis of this experience, it can be said that the magnitude of the power excursion which results from the 0.6% reactivity addition will not depend greatly on whether the reactivity is added suddenly or relatively slowly and in neither case will it approach a level which would cause a fuel plate to burn out.

In order to compute the power level at which the reactor will operate after the addition of the 0.6% excess reactivity discussed in the foregoing, it is necessary to know the water-temperature coefficient of reactivity. The relative importance of the two moderators, graphite and water, in determining the effective neutron temperature introduce uncertainties in the theoretical computation of this coefficient. The coefficient cannot, however, have an absolute magnitude less than that of the water-*density* coefficient of reactivity referred to a temperature scale, i.e., the coefficient computed on the assumption that:

$$\frac{d\,k_{eff}}{dT} = \frac{\delta\,k_{eff}}{\delta\rho} \cdot \frac{\delta\rho}{\delta T}$$

where ρ refers to the water density and T to the temperature. On the assumption that this minimum value is the true value, a rise of water temperature from near 0°C to 80°C would reduce reactivity by 0.6% k_{eff}.

The capacity of the reactor-coolant system is such that if the outside air temperature were 0°C and the average water temperature in the reactor were 80°C, energy would be removed at the rate of 365,000 BTU/hr or 107 kw. Under these conditions the reactor water-inlet temperature would be 60°C and the exit temperature, coincidentally, would be 100°C. It is, therefore, concluded that if the full available excess reactivity of 0.6% k_{eff} were added to the reactor on a cold day with the coolant system operating, the reactor would operate at an equilibrium power level about ten times higher (100 kw) than its normal maximum with little or no net steam production. Before reaching the equilibrium power, when the water in the coolant system would be heated to the equilibrium value, the reactor would operate at a somewhat higher power level and some net steam production might occur. If the coolant were not flowing during the time of excess reactivity addition, the equilibrium power level would be quite low and equal to the heat losses. In no case would the power level approach a high enough value to justify any fear of fuel-plate burnout.

Maximum Tolerable Sudden Reactivity Addition

In order to assess the safety factor which exists between the normal excess reactivity available in the reactor and the excess reactivity necessary for a serious power excursion, 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. Such an excursion would damage the reactor core but would not result in any substantial release of fission products.

The first step in the procedure is the estimation of the exponential period corresponding to the excess reactivity which would have characterized a power excursion of similar effect in Borax I. The estimate requires that (1) a relationship be established between the maximum temperature of the fuel plate and the energy release of the excursion and (2) the energy release be related to the period of the excursion.

For the case of power excursions of short period, with reactor water at saturation temperature, it is shown in Reference 1 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.* Measurements of the same type with cold reactor water (the case directly applicable to the UCTR) showed a similar relationship but with a proportionality constant of only about 10°F per MW-sec (Reference 3). The difference is not an unreasonable one since the subcooled water represents a more effective heat sink than the saturated water. However, the experiments with the saturated water were carried to short periods in the range of interest whereas the subcooled experiments were limited to longer periods. Therefore, more conservative saturated water data will be used.** To raise the maximum temperature of the fuel plate from the temperature of boiling water to the melting point of aluminum, a temperature change of approximately 1000°F, would require a power excursion with a total energy release of $\frac{1000°F}{24.4°F/MW\text{-sec}}$ or 41 MW-sec.

According to the data of Reference 3, replotted in Figure D-1a "subcooled" power excursion of reciprocal period 150 sec^{-1} would give an energy release of 41 MW-sec in addition to the energy necessary to raise the fuel plate temperature to the saturation temperature of water. It is, therefore, concluded that a power excursion of period at least as short as 1/150 sec (6.7 millisec) could have been tolerated by Borax I with subcooled water without melting at the hottest point in the fuel plates.

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 of the three related variables--neutron lifetime, excess reactivity, and exponential period--which characterize the neutron physics of a power excursion, it is the exponential period which determines the total energy release and the temperatures attained. The excess reactivity and neutron lifetime have large effects only as they jointly determine the period. This supposition is consistent, for example, with

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*Actually, the energy data of Reference 1 were revised in Reference 2 because of later and better calibrations of the instrumentation. The numbers above are taken from the later (more pessimistic) data.

**If subcooled data were used, the case directly applicable to UCTR, this analysis would indicate that step reactivity additions 2.4 times as large as those discussed here would not damage the reactor.

that the total energy transferred to the coolant water during a power excursion is many times the amount which would vaporize enough water to compensate for the excess reactivity, and that the actual reactivity reduction which occurs during the excursion is much larger than the initial excess reactivity. The extension of the Borax results to the UCTR is made on the basis of this evidence.

It is convenient, first, to treat only the effects of the slightly greater fuel-plate spacing and the slightly lower void coefficient of reactivity of the UCTR relative to the Borax I. Information will also be drawn from the Borax II experiments. The Borax II reactor differed from Borax I in that the coolant-channel thickness was greater in the ratio $\frac{0.264 \text{ in.}}{0.117 \text{ in.}} = \boxed{2.26}$ and that the calculated void coefficient of reactivity was lower in the ratio.

$$\frac{0.10\%\ k_{eff}/\%\ \text{void}}{0.24\%\ k_{eff}/\%\ \text{void}} = \frac{1}{\boxed{2.4}} = 0.416$$

Both of these differences would be expected to cause a higher energy release per fuel plate in Borax II than in Borax I for a power excursion of given period. The measurements made with subcooled water at periods down to 23 millisec showed that the energy release <u>per fuel plate</u> in Borax II was between $\boxed{1.7 \text{ and } 2.0}$ times that of Borax I, with the smaller ratio applying to the shorter periods (Reference 2). Therefore, it seems quite conservative to assume, in the case of any two reactors, (1) and (2), of the Borax type having a ratio of fuel plate spacings, S_1/S_2, and a ratio of void coefficients of reactivity, C_1/C_2, that the ratio of energy release per fuel plate for a subcooled power excursion of given period will be no greater than $E_2/E_1 = S_2/S_1$ or $E_2/E_1 = C_1/C_2$ <u>whichever is the larger</u>. For the UCTR and Borax I the ratios are

$$\frac{S_{UF}}{S_{Bo}} = \frac{0.137}{0.117} = 1.17 \quad \frac{C_{Bo}}{C_{UF}} = \frac{0.24}{0.18} = 1.33$$

It is concluded, therefore, that a Borax reactor having a coolant-channel thickness and a void coefficient of reactivity equal to those of the UCTR would release not more than 1.33 times as much energy per fuel plate as Borax I. The limiting nonmelting period for such a reactor would be that which in Borax I gave an energy release of 41/1.33 = 31MW-sec. The period obtained from Figure B-1, corresponding to a total energy release of 31MW-sec, is 8.3 millisec.

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 because a large fraction of the total energy released is stored in the fuel plate during the important stage of the reactor shut-down. For example, a reactor composed of fuel plates of high heat capacity undoubtedly will experience a larger total energy release, but not necessarily a higher maximum temperature, during a power excursion of given period, than a reactor having plates of low heat capacity.

From examination of the Borax results, it seems clear that two distinct phases of the reactor shut-down process occur consecutively and that both may be important in determining the maximum center temperature of a fuel plate. The first phase covers the interval before an important amount of boiling occurs at the fuel-plate surface. During this interval, the heat loss to the water is small and the important consideration is evidently the ratio of fuel-plate surface temperature (which determines the start of boiling) to center temperature. For periods in the range under consideration, this temperature ratio is theoretically not far from unity (0.76 minimum for a 10-millisec period in Borax I). Experimentally, the temperature ratio was unity for periods down to 5 millisec in the Borax I measurements. Since the total effect is small and since the temperature ratio for Borax and UCTR fuel plates should not be much different, the thinner cladding will tend to balance the effect of the poorer "meat" conductivity. It is concluded, therefore, that there will be no important difference in fuel plate performance during this initial phase of the excursion.

The second phase of the power excursion begins when a significant rate of boiling is established at the plate surface. Reactivity and consequently generation are reduced at a rate which must be a function of the rate at which heat can be transferred into the boiling water. At the same time, the transfer of heat into the water removes heat from the fuel plate and limits its temperature rise. The important characteristic of the plate during this phase of the excursion is the heat flux which it can supply to the water for a given temperature difference between the plate center and surface. A figure assumed to be roughly indicative of the relative performance or merit of fuel plates during this phase is the ratio of heat flux to temperature difference under steady-state conditions. This ratio (figure of merit) will overemphasize the difference between fuel plates since the temperature distribution in the plate will be more peaked during a steady-state conduction than during conduction when the general temperature level is rising. The ratio of these figures of merit for Borax I and for the UCTR is

$$\frac{\left[\dfrac{\text{Heat Flux}}{\Delta T_{c-s}}\right]_{\text{UCTR}}}{\left[\dfrac{\text{Heat Flux}}{\Delta T_{c-s}}\right]_{\text{Borax}}} = 0.82$$

A conservative procedure would be to apply the above factor to the permissible total energy of excursion on the Borax I curve. At the same time, however, the difference in gross maximum to average power ratio for the two reactors should be taken into account since it is the temperature of the hottest point in the hottest fuel plate which is being considered. The power ratio for the two reactors is

$$\frac{\left.\dfrac{\text{Max}}{\text{Ave}}\right|_{\text{Borax}}}{\left.\dfrac{\text{Max}}{\text{Ave}}\right|_{\text{UCTR}}} = \frac{1.82}{1.63} = 1.12$$

The combination of these two factors reduces the permissible equivalent energy of the Borax-type excursion to

$$31 \times 0.82 \times 1.12 = 28.4 \text{ MW-sec}$$

The corresponding exponential period from Figure B-1 is 9.1 millisec. It is, therefore, concluded that the UCTR will tolerate a power excursion of period at least as short as 9.1 millisec without the melting of any part of any fuel plate. The excess reactivity corresponding to this period is 2.3% k_{eff}.

Successive Power Excursions

It is typical of the Borax and SPERT reactors, unless the excess reactivity is removed by external means, that an initial power excursion which terminates itself by expelling water from the reactor core will be followed by subsequent excursions as the water falls and flows back into the core. An exception to this behavior occurs when the initial excursion is violent enough to cause a permanent loss of reactivity by throwing a large amount of water completely out of the reactor tank. In the UCTR the total quantity of water in the core is small, the submergence of the core is small, and baffles above the core are so arranged that any water splash is directed to the outside so that it cannot return to the core. Consequently, even a relatively mild power excursion (e.g., one having an exponential period of from 20 to 30 millisec) in the UCTR should result in permanent self-induced shutdown of the reactor. By these same design features, the possibility of large successive power excursions, such as those studied in the SPERT project, resulting from the ramp addition of excess reactivity is eliminated. It can be anticipated that the UCTR will be safe against quite large ramp additions (larger than 2.3% k_{eff}) provided only that the ramp rate is not so rapid as to add an excess reactivity of more than 2.3% k_{eff} before the reactor power reaches a high level. To exceed this limit the ramp rate would need to be of the order of 1.0% k_{eff} per second or larger.

Beam Tube Reactivity Effects

The UCTR has two 6-inch diameter beam tubes which extend to within 11 inches of the fuel-graphite interfaces. The maximum change on the core reactivity which can be effected by these two beam-tube facilities was calculated to be 0.18% $\Delta K/K$ or 0.09% $\Delta K/K$ per beam tube. The calculation is based upon the effect of a black absorber six inches in diameter placed in the same position as the beam tubes. The reduction in the reflector savings due to the black absorber was calculated using the following equation.

$$\delta \text{ (reflector savings)} = \frac{D(\text{core})}{D(\text{reflector})} \cdot L \text{ (reflector)}. \tanh \frac{T(\text{reflector thickness})}{L(\text{reflector})}$$

Reference: *Elements of Nuclear Reactor Theory*, Glasstone and Edlund.

The reflector savings for the 49.5 cm and 28.0 cm of graphite were calculated to be 7.83 cm and 5.26 cm respectively. The area of the black absorber is 13% of the adjacent core face area. Using the reflector savings given above and the area weighting factors, the reflector savings with and without the six-inch diameter black absorber were calculated to be 7.33 cm and 7.83 cm respectively.

The reactivity effect of the single six-inch diameter black absorber was then determined calculating the critical buckling with and without the black absorber.

Using the value of -0.09% $\Delta K/K$, for a single beam tube the shortest period which the reactor could go on, due to the sudden withdrawal of a black absorber from the six-inch beam tube, would be approximately 80 seconds. Therefore, the reactivity change which can be effected by the beam tubes does not represent a hazard to reactor operation.

In addition to the two 6-inch beam tubes which penetrate the outer reflector, there are four 4-inch beam tubes which terminate outside of the reflector. No calculations were made for the 4-inch tubes since their effect on reactivity will be much smaller than that of the 6-inch tubes.

REFERENCES

1. Dietrich, J. R. and D. C. Layman, *Transient and Steady State Characteristics of a Boiling Reactor. The Borax Experiments, 1953*, AECD-3840, Argonne National Laboratory, February, 1954.

2. Dietrich, J. R., *Experimental Determinations of the Self-Regulation and Safety of Operating Water-Moderated Reactors*, A/Conf. 8/P/481, International Conference on the Peaceful Uses of Atomic Energy, June 30, 1955.

3. Dietrich, J. R., *Experimental Investigation of the Self-Limitation of Power During Reactivity Transients in a Subcooled, Water-Moderated Reactor. Borax-I Experiments, 1954*, AECD-3668, Argonne National Laboratory, August 17, 1955.

4. Lennox, D. H. and C. N. Kelber, *Summary Report on the Hazards of the Argonaut Reactor*, ANL-5647, December 1956.

UCLA

TRAINING REACTOR

HAZARDS SUMMARY REPORT

Prepared by

AMF ATOMICS
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Greenwich, Connecticut

October, 1959

SECTION III

REACTOR HAZARDS EVALUATION

A. GENERAL SAFETY CONSIDERATIONS

The inherent safety of this reactor is based on four points. First, the amount of excess reactivity in the reactor is limited to about 0.6 per cent. 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.

Although the reactor is designed to operate at a maximum steady power of 10 kw, it is not planned to operate it at this power level continuously. Much of the operation for the training program will be at considerably lower power levels and will be intermittent. It is estimated that the reactor will be in operation (about 1800 hours each year or about 20 per cent of the total time).* With this type of operational program, no very large amount of new fission products will ever exist in the core.

The excess reactivity will be limited to 0.6 per cent k by adjustment of the original core loading. (Any additional fuel plates will be kept in a locked storage cabinet or fuel storage holes as described in Section II-E).* Loading or rearrangement of fuel in the reactor will be

* To be verified by University.

E. INITIATION OF REACTOR HAZARD BY EXTERNAL MEANS

1. Fire

Since none of the materials of construction of the reactor are inflammable, and since the reactor core is very well protected from any external fire by the concrete shield, damage from fire resulting in the release of fission products is extremely remote. (The reactor building is fireproof construction and will not be used for storage of quantities of inflammable materials.) *

2. Phenomena of Nature

None of the phenomena of nature such as hurricanes, tornadoes, lightning or floods offers a credible means of initiating a hazard related to the presence of the reactor. Adequate production has been incorporated into the reactor design to prevent lateral movement of the shielding blocks due to earthquake forces, and to SCRAM the reactor in the event of any displacement of the shield blocks due to these forces. (Hurricanes and tornadoes have been rare and not severe over Los Angeles. Lightning is frequent but, with proper construction and protective devices, should offer no hazard to the reactor. The heaviest rainfall for a 24-hr period recorded during the last 20 years occurred in ( ) ** with no evidence of flooding in the area of the reactor site.) *

* To be verified by UCLA.

** To be specified by UCLA

APPENDIX A

REACTOR CALCULATIONS

Two-group diffusion-theory calculations have been made to estimate the reactor-physics characteristics. For the calculations, the actual configuration of the training-reactor core and reflector assemblies was transformed into an equivalent one-dimensional model using the concept of reflector savings to account approximately for the neutron leakage in the other two directions. Figure 15 is a sketch of the one-dimensional model showing the three separate regions considered in the calculations. Region I is the central graphite zone coupling the two core slabs, Region II are the core slabs, and Region III is the external graphite reflector.

For the training-reactor configuration, a reasonably accurate value of over-all reflector savings is required to account properly for the neutron leakage, principally from the graphite regions, in the two directions transverse to that considered in the one-dimensional model (two directions perpendicular to the X-direction shown in Fig. 15). In addition to the diffusion leakage of neutrons (accounted for by use of the reflector savings), streaming of neutrons occurs through the four narrow air slots (3/4-in. air gap) provided for the control blades. The leakage current of streaming neutrons cannot be determined accurately by any practical calculational technique and so can be estimated only roughly. For the initial calculation, the leakage of streaming neutrons through the air slots is neglected. Its effect on the critical-mass requirement is roughly estimated later.

To obtain a reliable value of over-all reflector savings, advantage was taken of the reactivity measurements made at the Argonne National Laboratory and reported in ANL-5647 for the Argonaut reactor. There are no air slots in the Argonaut design. One of the Argonaut configurations for which reactivity values were experimentally determined consisted of two groups of fuel-box clusters arranged so as to approximate closely two core slabs facing each other and separated by two ft. of graphite. The reflector-savings value used is based on obtaining agreement between calculated and measured critical-mass requirements of the Argonaut two-slab configuration. In the calculations, the two-group nuclear constants used for the core regions were determined by standard methods generally found satisfactory for water-moderated cores. The resonance absorption of the U-238 in the 90-per-cent-enriched fuel used was included in the calculations. The nuclear constants for graphite of average density equal to 1.6 gm/cc were obtained from Appendix B of ANL-5647.

The critical-mass requirement for the training reactor, neglecting the effect of neutron streaming through the control-blade air slots, was determined in the same manner as that used to check the Argonaut critical-mass value. In addition, calculations were made of (1) the uniform water-void coefficient; (2) the temperature coefficient; (3) the effective prompt-neutron lifetime; (4) the reactivity change attributed to loss of water from the core slabs; (5) the reactivity change resulting from consumption of the U-235 content in the core slabs; (6) the effect on the critical-mass requirement of bringing the two core slabs together.

The core aluminum-to-water volume ratio is 0.51 and the separation distance between core slabs is 1.0 ft. for the UCTR design. The fuel is 90-per-cent-enriched U-235. The critical mass required for the training reactor in the cold-clean condition, neglecting the effect of neutron streaming through the air slots, was calculated as 2.6 kg of U-235. Additional calculated results obtained are tabulated as follows:

Uniform water-void coefficient -------------	-0.18% k/% void
Temperature coefficient ------------------	-0.9×10^{-4} k/°C
Prompt-neutron lifetime ------------------	1.4×10^{-4} sec
U-235 mass coefficient -------------------	0.31% k/% U-235 mass

complete removal of water from the core slabs gives a k_{eff} of 0.56 which represents a large reactivity loss from the system.

Calculations made for the two core slabs placed side by side against each other with graphite completely surrounding them give a critical-mass requirement for the cold-clean condition of 1.9 kg.

In Figure 16, the critical masses calculated for the two-slab separations (zero and 1.0-ft separation) are plotted versus separation distance. In addition, the critical-mass data obtained for several Argonaut configurations are shown in the plot. The points plotted are (1) 2.6 kg. calculated for 1.0-ft separation used in the UCTR design; (2) 1.9 kg. calculated for zero separation between the two core slabs of the UCTR (3) 3.748 kg. measured for 2-ft separation in the Argonaut experiments; and (4) 2.2 kg. measured for a slab loading on one side only in the Argonaut experiments. The slab dimensions are indicated on the plate. The core compositions for the fuel boxes used in the UCTR design and in the Argonaut experiments are closely comparable in neutron slowing-down and thermal-neutron absorption properties. It is evident from Figure 16 that as the core slabs are brought together, the critical mass requirement decreases. This is undoubtedly due to increased coupling between the slabs (i.e. a greater fraction of the neutrons born in and leaking from one core slab causes fissions in the other slab). The points in Figure 16 show generally good agreement between calculated and experimentally measured critical-mass requirements.

The calculated-void coefficient, temperature coefficient, neutron lifetime, and U-235 mass coefficient for the UCTR are of the same order of magnitude as measured for the Argonaut two-slab configuration. The small differences are generally in the direction to be expected from the better coupling between core slabs in the UCTR design.

Data reported in ANL-5647 for the two-slab configuration show that a 4 x 6-in. void, 36 in. high, in the outer graphite reflector zone, next to a fuel slab, gives a reactivity decrease of -2.2 per cent k. This reactivity decrease corresponds to about an 8-per-cent increase in required critical mass. On this basis, it is estimated that the increase in critical mass for the UCTR, caused by the air slots could be as high as 20 per cent of that calculated for a structure with no air slots. This gives a cold-clean critical-mass requirement of about 3.2 kg.

The core average thermal and fast fluxes are calculated as 8.2×10^{10} and 1.9×10^{11} n/cm^2-sec, respectively, for 10-kw operation. Figure 17 is a plot of the thermal and fast flux distributions along the X-axis as defined in Figure 16. The abscissa in Figure 17 is measured from the line of symmetry of the reactor configuration (center line) of internal graphite zone. Also plotted in Figure 17 are the fast and thermal adjoint fluxes useful for determining the important functions for materials placed or changes made, at various positions in the reactor system.

Calculations have been made to establish a lower limit to the control effectiveness of the control blades used in the design. The calculations indicate that a minimum of 1.5 per cent k per blade (three blades) can be expected. The remaining blade used for regulation purposes will be adjusted to give 0.6 per cent k.

APPENDIX B

ESTIMATION OF EFFECTS OF ASSUMED LARGE REACTIVITY ADDITIONS

It has been demonstrated repeatedly in the Borax and SPERT reactors that water-cooled, water-moderated reactors of suitable design may have a very substantial self-protection against the effects of reactivity accidents, even in the absence of corrective action by the reactor control system. This self-protection is provided by the negative steam-void coefficient of reactivity and the negative temperature coefficient of reactivity, both of which can result in important reactivity reductions as the reactor power rises. The UCTR has been designed with a high degree of self-protection of this type. In this appendix estimates are made of the behavior of the reactor under various hypothetical conditions of excess reactivity addition with no corrective action by the control system.

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. Its behavior can be predicted most reliably by utilizing the Borax I data with simple correction factors to convert them to the UCTR conditions.

The significant quantitative characteristics of the UCTR and the Borax I reactor are compared in Table B-1.

TABLE B-1

COMPARISON OF UCTR AND BORAX I CHARACTERISTICS

Characteristic	UCTR	Borax I
Fuel plate "meat"	13.4 w/o U-Al alloy (20 per cent enriched)	18 w/o U-Al alloy (fully-enriched)
Fuel plate cladding	1100 aluminum	1100 aluminum
"Meat" thickness	0.040 in.	0.020 in.
Cladding thickness	0.015 in	0.020 in.
Coolant-channel thickness	0.137 in.	0.117 in.
Core volume (approx.)	71 liters	106 liters
Void coefficient of reactivity (calculated)	-0.18 per cent k/ per cent coolant void	-0.24 per cent k/ per cent coolant void
Temperature coefficient of reactivity (room temperature)	-0.009 per cent k/°C (estimated)	-0.01 per cent k/°C
Effective prompt-neutron lifetime	1.4×10^{-4} sec (calculated)	0.65×10^{-4} sec
Power ratio in core, maximum average	1.63	1.82

In addition to the quantitative differences, the UCTR differs from Borax I in that the maximum coolant water level is only a few inches above the upper ends of the fuel plates (instead of about 4 ft) and the coolant water, once it has been ejected forcibly from the core by a power excursion, cannot fall or flow back into the core.

Effect of 0.6 Per Cent Excess Reactivity

An excess reactivity of 0.6 per cent k_{eff} will be available in the reactor if its temperature is abnormally low (nearly freezing).

The addition of all this excess reactivity will cause the reactor to operate at a power such that the reactivity losses associated with the temperature increase and the voids formed will equal the excess reactivity.

If the reactivity is added slowly, after the reactor is critical, the power will approach such an equilibrium level slowly as the reactivity is added. If the reactivity is added suddenly when the reactor is initially subcritical or at very low power, the power will at first rise exponentially with a period not shorter than 0.8 sec which is the asymptotic period corresponding to the full excess reactivity of 0.6 per cent k_{eff}. Many experiments with the Borax reactors have demonstrated that for periods of this order of magnitude the transition from the exponential power rise to the equilibrium power level (in which excess reactivity is balanced by temperature and steam void coefficients) is a smooth one involving little or no power overshoot. On the basis of this experience, it can be said that the magnitude of the power excursion which results from the 0.6 per cent reactivity addition will not depend greatly on whether the reactivity is added suddenly or relatively slowly and in neither case will it approach a level which would cause a fuel plate to burn out.

In order to compute the power level at which the reactor will operate after the addition of the 0.6 per cent excess reactivity discussed in the foregoing it is necessary to know the water-temperature coefficient of reactivity. The relative importance of the two moderators, graphite and water, in determining the effective neutron temperature introduce uncertainties in the theoretical computation of this coefficient. The coefficient cannot, however, have an absolute magnitude less than that of the water-density coefficient of reactivity referred to a temperature scale, i.e., the coefficient computed on the assumption that:

$$\frac{d\ k_{eff}}{dT} = \frac{\delta\ k_{eff}}{\delta\ \rho} \cdot \frac{\delta\ \rho}{\delta\ T}$$

where ρ refers to the water density and T to the temperature. Note that $\frac{\delta \rho}{\delta T}$ is just the negative of the void coefficient of reactivity. On the assumption that this minimum value is the true value, a rise of water temperature from near 0°C to 80°C would reduce reactivity by 0.6 per cent k_{eff}.

The capacity of the reactor-coolant system is such that if the outside air temperature were 0°C and the average water temperature in the reactor were 80°C, energy would be removed at the rate of 365,000 BTU/hr or 107 kw. Under these conditions the reactor water-inlet temperature would be 60°C and the exit temperature, coincidentally, would be 100°C. It is, therefore, concluded that if the full available excess reactivity of 0.6 per cent k_{eff} were added to the reactor on a cold day with the coolant system operating, the reactor would operate at an equilibrium power level about 10 times higher (100 kw) than its normal maximum with little or no net steam

...on. Before reaching the equilibrium power, when the water in the coolant system would ...ed to the equilibrium value, the reactor would operate at a somewhat higher power level ...ome net steam production might occur. If the coolant were not flowing during the time of ...ess reactivity addition, the equilibrium power level would be quite low and equal to the heat ...sses. In no case would the power level approach a value high enough to justify any fear of ...uel-plate burnout.

Maximum Tolerable Sudden Reactivity Addition

In order to assess the safety factor which exists between the normal excess reactivity available in the reactor and the excess reactivity necessary for a serious power excursion, 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. Such an excursion would damage the reactor core but would not result in any substantial release of fission products.

The first step in the procedure is the estimation of the exponential period corresponding to the excess reactivity which would have characterized a power excursion of similar effect in Borax I. The estimate requires that (1) a relationship be established between the maximum temperature of the fuel plate and the energy release of the excursion and (2) the energy release be related to the period of the excursion.

For the case of power excursions of short period, with reactor water at saturation temperature, it is shown in Reference 1 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.* Measurements of the same type with cold reactor water (the case directly applicable to the UCTR) showed a similar relationship but with a proportionality constant of only about 10°F per MW-sec (Reference 3). The difference is not an unreasonable one since the subcooled water represents a more effective heat sink than the saturated water. However, the experiments with the saturated water were carried to short periods in the range of interest whereas the subcooled experiments were limited to longer periods. Therefore, more conservative saturated water data will be used. To raise the maximum temperature of the fuel plate from the temperature of boiling water to the melting point of aluminum, a temperature change of approximately 1000°F, would require a power excursion with a total energy release of $\frac{1000°F}{24.4°F/MW\text{-sec}}$ or 41 MW-sec.

According to the data of Reference 3, replotted in Figure 6.1A, a "subcooled" power excursion of reciprocal period 150 sec^{-1} would give an energy release of 41 MW-sec in addition to the energy necessary to raise the fuel plate temperature to the saturation temperature of water. It is therefore concluded that a power excursion of period at least as short as 1/150 sec (6.7 millisec) could have been tolerated by Borax I with subcooled water without melting at the hottest point in the fuel plates.

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 of the three related variables--neutron lifetime, excess reactivity, and exponential period--which characterize the neutron physics of a power excursion, it is the exponential period which determines the total energy release and the temperatures attained. The excess reactivity and neutron lifetime have large effects only as they jointly determine the period. This supposi-

*Actually, the energy data of Reference 1 were revised in Reference 2 because of later and better calibrations of the instrumentation. The numbers above are taken from the later (more pessimistic) data.

consistent, for example, with the observations that the total energy transferred to the water during a power excursion is many times the amount which would vaporize enough water to compensate for the excess reactivity, and that the actual reactivity reduction which occurs during the excursion is much larger than the initial excess reactivity. The extension of the Borax results to the UCTR is made on the basis of this evidence.

It is convenient, first, to treat only the effects of the slightly greater fuel-plate spacing and the slightly lower void coefficient of reactivity of the UCTR. relative to the Borax I. Information will also be drawn from the Borax II experiments. The Borax II reactor differed from Borax I in that the coolant-channel thickness was greater in the ratio $\frac{0.264 \text{ in.}}{0.117 \text{ in.}} = 2.26$ and that the calculated void coefficient of reactivity was lower in the ratio

$$\frac{0.10\%\ k_{eff}/\%\ \text{void}}{0.24\%\ k_{eff}/\%\ \text{void}} = \frac{1}{2.4} = 0.416$$

Both of these differences would be expected to cause a higher energy release per fuel plate in Borax II than in Borax I for a power excursion of given period. The measurements made with subcooled water at periods down to 23 millisec showed that the energy release per fuel plate in Borax II was between 1.7 and 2.0 times that of Borax I, with the smaller ratio applying to the shorter periods (Reference 2). Therefore, it seems quite conservative to assume, in the case of any two reactors, (1) and (2), of the Borax type having a ratio of fuel plate spacings, S_1/S_2, and a ratio of void coefficients of reactivity, C_1/C_2, that the ratio of energy release per fuel plate for a subcooled power excursion of given period will be no greater than $E_2/E_1 = S_2/S_1$ or $E_2/E_1 = C_1/C_2$ whichever is the larger. For the UCTR and Borax I the ratios are:

$$\frac{S_{UF}}{S_{Bo}} = \frac{0.137}{0.117} = 1.17 \qquad \frac{C_{Bo}}{C_{UF}} = \frac{0.24}{0.18} = 1.33$$

It is concluded, therefore, that a Borax reactor having a coolant-channel thickness and a void coefficient of reactivity equal to those of the UCTR would release not more than 1.33 times as much energy per fuel plate as Borax I. The limiting non-melting period for such a reactor would be that which in Borax I gave an energy release of 41/1.33 = 31 MW-sec. The period obtained from Figure 18, corresponding to a total energy release of 31 MW-sec, is 8.3 millisec.

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 because a large fraction of the total energy released is stored in the fuel plate during the important stage of the reactor shut-down. For example, a reactor composed of fuel plates of high heat capacity undoubtedly will experience a larger total energy release, but not necessarily a higher maximum temperature, during a power excursion of given period, than a reactor having plates of low heat capacity.

From examination of the Borax results it seems clear that two distinct phases of the reactor shutdown process occur consecutively and that both may be important in determining the maximum center temperature of a fuel plate. The first phase covers the interval before an important amount of boiling occurs at the fuel-plate surface. During this interval the heat loss to the water is small and the important consideration is evidently the ratio of fuel-plate surface temperature (which determines the start of boiling) to center temperature. For periods in the range under consideration this temperature ratio is theoretically not far from unity (0.76 minimum for a 10-millisec period in Borax I). Experimentally the temperature ratio was unity for periods down to 5 millisec in the Borax I measurements. Since the total effect is small and since the temperature ratio for Borax and UCTR fuel plates should not be much different, the thinner cladding will balance the effect of the poorer "meat" conductivity. It is concluded, therefore, that will be no important difference in fuel plate performance during this initial phase of the rsion.

The second phase of the power excursion begins when a significant rate of boiling is established at the plate surface. Reactivity and consequently generation are reduced at a rate which must be a function of the rate at which heat can be transferred into the boiling water. At the same time the transfer of heat into the water removes heat from the fuel plate and limits its temperature rise. The important characteristic of the plate during this phase of the excursion is the heat flux which it can supply to the water for a given temperature difference between the plate center and surface. A figure assumed to be roughly indicative of the relative performance or merit of fuel plates during this phase is the ratio of heat flux to temperature difference under steady-state conditions. This ratio (figure of merit) will overemphasize the difference between fuel plates since the temperature distribution in the plate will be more peaked during a steady-state conduction than during conduction when the general temperature level is rising. The ratio of these figures of merit for Borax I and for the UCTR is

$$\frac{\left[\dfrac{\text{Heat Flux}}{\Delta T_{c-s}}\right]_{\text{UCTR}}}{\left[\dfrac{\text{Heat Flux}}{\Delta T_{c-s}}\right]_{\text{Borax}}} = 0.82$$

A conservative procedure would be to apply the above factor to the permissible total energy of excursion on the Borax I curve. At the same time, however, the difference in gross maximum to average power ratio for the two reactors should be taken into account since it is the temperature of the hottest point in the hottest fuel plate which is being considered. The power ratio for the two reactors is

$$\frac{\dfrac{\text{Max}}{\text{Ave}}_{\text{Borax}}}{\dfrac{\text{Max}}{\text{Ave}}_{\text{UCTR}}} = \frac{1.82}{1.63} = 1.12$$

The combination of these two factors reduces the permissible equivalent energy of the Borax-type excursion to

$$31 \times 0.82 \times 1.12 = 28.4 \text{ MW-sec.}$$

The corresponding exponential period from Figure 18 is 9.1 millisec. It is, therefore, concluded that the UCTR will tolerate a power excursion of period at least as short as 9.1 millisec without the melting of any part of any fuel plate. The excess reactivity corresponding to this period is 2.3 per cent k_{eff}.

Successive Power Excursions

It is typical of the Borax and SPERT reactors, unless the excess reactivity is removed by external means, an initial power excursion which terminates itself by expelling water from actor core will be followed by subsequent excursions as the water falls and back into the core. An exception to this behavior occurs when the initial ursion is violent enough to cause a permanent loss of reactivity by throwing large amount of water completely out of the reactor tank. In the UCTR the total quantity of water in the core is small, the submergence of the core is small, and baffles above the core are so arranged that any water splash is directed to the our de so that it cannot return to the core. Consequently, even a relatively mild power excursion (e.g., one having an exponential period of from 20 to 30 millisec) in the UCTR should result in permanent self-induced shutdown of the reactor. By these same design features the possibility of large successive power excursions, such as those studied in the SPERT project, resulting from the ramp addition of excess reactivity is eliminated. It can be anticipated that the UCTR will be safe against quite large ramp additions (larger than 2.3 per cent keff) provided only that the ramp rate is not so rapid as to add an excess reactivity of more than 2.3 per cent k_{eff} before the reactor power reaches a high level. To exceed this limit the ramp rate would need to be of the order of 1.0 per cent k_{eff} per second or larger.

Beam Tube Reactivity Effects

The UCTR has two 6-inch diameter beam tubes which extend to within 11 inches of the fuel-graphite interfaces. The maximum change on the core reactivity which can be effected by these two beam tube facilities was calculated to be 0.18% $\Delta K/K$ or 0.09% $\Delta K/K$ per beam tube. The calculation is based upon the effect of a black absorber 6 inches in diameter placed in the same position as the beam tubes. The reduction in the reflector savings due to the black absorber was calculated using the following equation.

$$\delta\,(\text{reflector savings}) = \frac{D(\text{core})}{D(\text{reflector})}\cdot L(\text{reflector})\cdot \tanh\left(\frac{T(\text{reflector thickness})}{L(\text{reflector})}\right)$$

Reference: "Elements of Nuclear Reactor Theory" Glasstone and Edlund.

The reflector savings for the 49.5 cm and 28.0 cm of graphite were calculated to be 7.83 cm and 5.26 cm respectively. The area of the black absorber is 13% of the adjacent core face area. Using the reflector savings given above and the area weighting factors, the reflector savings with and without the 6" diameter black absorber were calculated to be 7.33 cm and 7.83 cm respectively.

The reactivity effect of the single 6" diameter black absorber was then determined calculating the critical buckling with and without the black absorber.

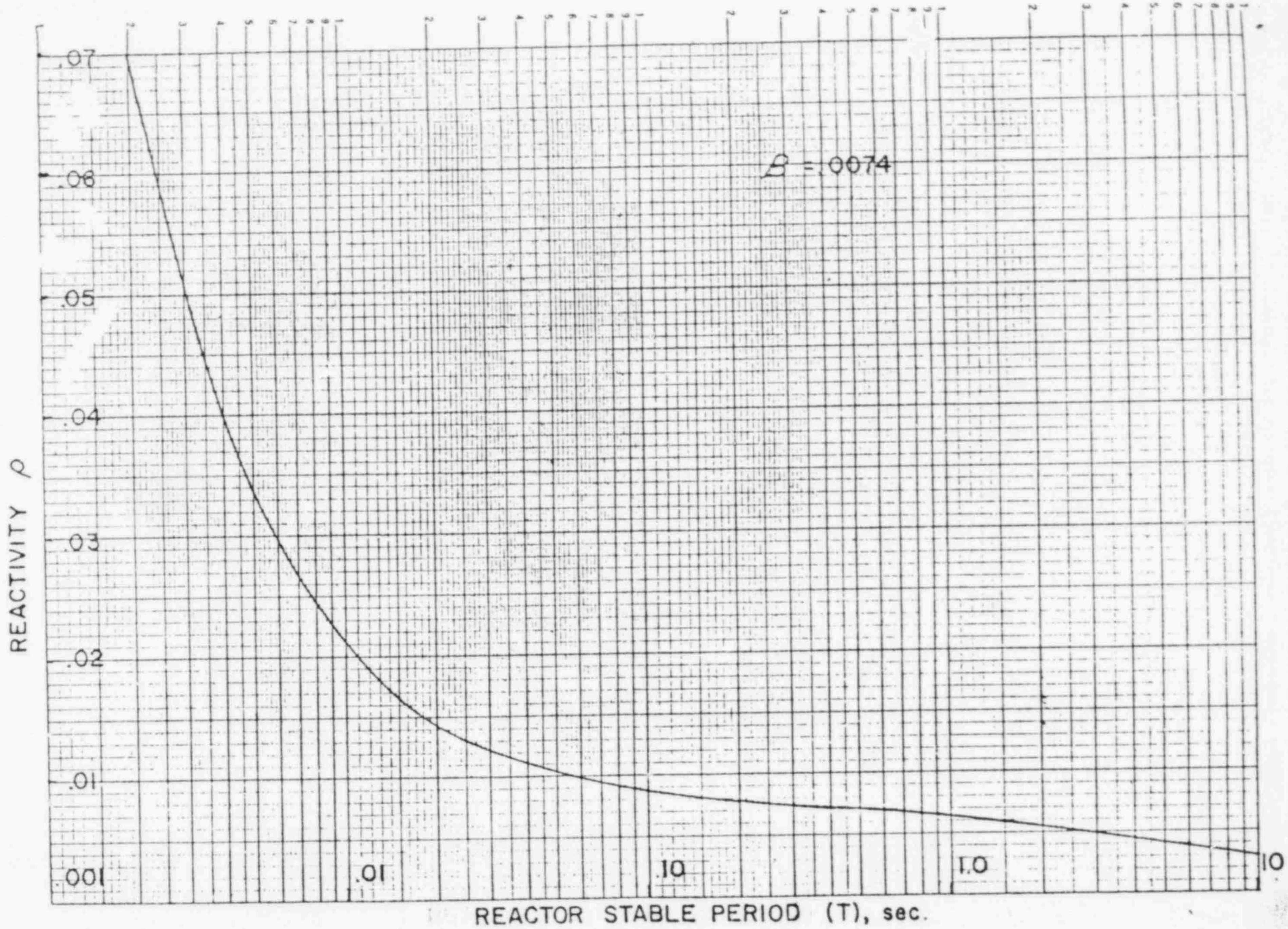
Using the value of -0.09% $\Delta K/K$, for a single beam tube the shortest period which the reactor could go on, due to the sudden withdrawal of a black absorber from the 6" beam tube, would be approximately 80 seconds. Therefore, the reactivity change which can be effected by the beam tubes does not represent a hazard to reactor operation.

REFERENCES

B.1 Dietrich, J.R., and D.C. Layman, "Transient and Steady State Characteristics of a Boiling Reactor. The Borax Experiments, 1953," AECD-3840, Argonne National Laboratory, February, 1954.

B.2 Dietrich, J.R., "Experimental Determinations of the Self-Regulation and Safety of Operating Water-Moderated Reactors," A/Conf. 8/P/481, International Conference on the Peaceful Uses of Atomic Energy, June 30, 1955.

B.3 Dietrich, J.R., "Experimental Investigation of the Self-Limitation of Power During Reactivity Transients in a Subcooled, Water-Moderated Reactor. Borax-I Experiments, 1954," AECD-3668, Argonne National Laboratory, August 17, 1955.



β = .0074
REACTIVITY ρ
.07
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.03
.02
.01
.001
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.10
1.0
10
REACTOR STABLE PERIOD (T), sec.

UNIVERSITY OF FLORIDA TRAINING REACTOR

HAZARDS SUMMARY REPORT

Prepared by

J. M. Duncan

$5.00 per copy

A Report to
The United States Atomic Energy Commission
Division of Civilian Application

From the
Department of Nuclear Engineering
College of Engineering
University of Florida
Gainesville, Florida
October, 1958

APPENDIX F

ESTIMATION OF EFFECTS OF ASSUMED LARGE REACTIVITY ADDITIONS

It has been demonstrated repeatedly in the Borax and SPERT reactors that water-cooled, water-moderated reactors of suitable design may have a very substantial self-protection against the effects of reactivity accidents, even in the absence of corrective action by the reactor control system. This self-protection is provided by the negative steam-void coefficient of reactivity and the negative temperature coefficient of reactivity, both of which can result in important reactivity reductions as the reactor power rises. The UFTR has been designed with a high degree of self-protection of this type. In this appendix estimates are made of the behavior of the reactor under various hypothetical conditions of excess reactivity addition with no corrective action by the control system.

The characteristics of the UFTR 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. Its behavior can be predicted most reliably by utilizing the Borax I data with simple correction factors to convert them to the UFTR conditions.

The significant quantitative characteristics of the UFTR and the Borax I reactor are compared in Table F-1.

TABLE F-1

COMPARISON OF UFTR AND BORAX I CHARACTERISTICS

Characteristic	UFTR	Borax I
Fuel plate "meat"	46 w/o U-Al alloy (20 per cent enriched)	18 w/o U-Al alloy (fully-enriched)
Fuel plate cladding	1100 aluminum	1100 aluminum
"Meat" thickness	0.040 in.	0.020 in.
Cladding thickness	0.015 in	0.020 in.
Coolant-channel thickness	0.137 in.	0.117 in.
Core volume (approx.)	71 liters	106 liters
Void coefficient of reactivity (calculated	-0.21 per cent k/ per cent coolant void	0.24 per cent k/ per cent coolant void
Temperature coefficient of reactivity (room temperature)	-0.01 per cent k/°C (estimated)	-0.01 per cent k/°C
Effective prompt-neutron lifetime	1.4×10^{-4} sec (calculated)	0.65×10^{-4} sec
Power ratio in core, maximum average	1.63	1.82

The U-238 in the 20 per cent enriched fuel of the UFTR introduces a negative Doppler coefficient of reactivity estimated to be of the order of 4×10^{-7} k/°C equivalent to .004 per cent reduction in k per 100°C rise in fuel temperature. Although the Doppler coefficient acts instantaneously and would cause the shutdown of the reactor in case of a reactivity accident, its effect is not expected to be important because expulsion of the water moderator will terminate an excursion before the fuel temperature has risen appreciably.

In addition to the quantitative differences, the UFTR differs from Borax I in that the maximum coolant water level is only a few inches above the upper ends of the fuel plates (instead of about 4 ft) and the coolant water, once it has been ejected forcibly from the core by a power excursion, cannot fall or flow back into the core.

Effect of 0.6 Per Cent Excess Reactivity

An excess reactivity of 0.6 per cent k_{eff} will be available in the reactor if its temperature is abnormally low (nearly freezing).

The addition of all this excess reactivity will cause the reactor to operate at a power such that the reactivity losses associated with the temperature increase and the voids formed will equal the excess reactivity.

If the reactivity is added slowly, after the reactor is critical, the power will approach such an equilibrium level slowly as the reactivity is added. If the reactivity is added suddenly when the reactor is initially subcritical or at very low power, the power will at first rise exponentially with a period not shorter than 0.3 sec which is the asymptotic period corresponding to the full excess reactivity of 0.6 per cent k_{eff}. Many experiments with the Borax reactors have demonstrated that for periods of this order of magnitude the transition from the exponential power rise to the equilibrium power level (in which excess reactivity is balanced by temperature and steam void coefficients) is a smooth one involving little or no power overshoot. On the basis of this experience, it can be said that the magnitude of the power excursion which results from the 0.6 per cent reactivity addition will not depend greatly on whether the reactivity is added suddenly or relatively slowly and in neither case will it approach a level which would cause a fuel plate to burn out.

In order to compute the power level at which the reactor will operate after the addition of the 0.6 per cent excess reactivity discussed in the foregoing it is necessary to know the water-temperature coefficient of reactivity. The relative importance of the two moderators, graphite and water, in determining the effective neutron temperature introduce uncertainties in the theoretical computation of this coefficient. The coefficient cannot, however, have an absolute magnitude less than that of the water-density coefficient of reactivity referred to a temperature scale, i.e., the coefficient computed on the assumption that:

$$\frac{d\ k_{eff}}{dT} = \frac{\delta\ k_{eff}}{\delta\ \rho} \cdot \frac{\delta\ \rho}{\delta\ T}$$

where ρ refers to the water density and T to the temperature. Note that $\frac{\delta \rho}{\delta T}$ is just the negative of the void coefficient of reactivity. On the assumption that this minimum value is the true value, a rise of water temperature from near 0°C to 80°C would reduce reactivity by 0.6 per cent k_{eff}.

The capacity of the reactor-coolant system is such that if the outside air temperature were 0°C and the average water temperature in the reactor were 80°C, energy would be removed at the rate of 365,000 BTU/hr or 107 kw. Under these conditions the reactor water-inlet temperature would be 60°C and the exit temperature, coincidentally, would be 100°C. It is, therefore, concluded that if the full available excess reactivity of 0.6 per cent k_{eff} were added to the reactor on a cold day with the coolant system operating, the reactor would operate at an equilibrium power level about 10 times higher (100 kw) than its normal maximum with little or no net steam production. Before reaching the equilibrium power, when the water in the coolant system would be heated to the equilibrium value, the reactor would operate at a somewhat higher power level and some net steam production might occur. If the coolant were not flowing during the time of excess reactivity addition, the equilibrium power level would be quite low and equal to the heat losses. In no case would the power level approach a value high enough to justify any fear of fuel-plate burnout.

Maximum Tolerable Sudden Reactivity Addition

In order to assess the safety factor which exists between the normal excess reactivity available in the reactor and the excess reactivity necessary for a serious power excursion, 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. Such an excursion would damage the reactor core but would not result in any substantial release of fission products.

The first step in the procedure is the estimation of the exponential period corresponding to the excess reactivity which would have characterized a power excursion of similar effect in Borax I. The estimate requires that (1) a relationship be established between the maximum temperature of the fuel plate and the energy release of the excursion and (2) the energy release be related to the period of the excursion.

For the case of power excursions of short period, with reactor water at saturation temperature, it is shown in Reference 1 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.* Measurements of the same type with cold reactor water (the case directly applicable to the UFTR) showed a similar relationship but with a proportionality constant of only about 10°F per MW-sec (Reference 3). The difference is not an unreasonable one since the subcooled water represents a more effective heat sink than the saturated water. However, the experiments with the saturated water were carried to short periods in the range of interest whereas the subcooled experiments were limited to longer periods. Therefore, more conservative saturated water data will be used. To raise the maximum temperature of the fuel plate from the temperature of boiling water to the melting point of aluminum, a temperature change of approximately 1000°F, would require a power excursion with a total energy release of $\frac{1000^{\circ}F}{24.4^{\circ}F/MW\text{-}sec}$ or 41 MW-sec.

According to the data of Reference 3, replotted in Figure 6.1A, a "subcooled" power excursion of reciprocal period 150 sec^{-1} would give an energy release of 41 MW-sec in addition to the energy necessary to raise the fuel plate temperature to the saturation temperature of water. It is therefore concluded that a power excursion of period at least as short as 1/150 sec (6.7 millisec) could have been tolerated by Borax I with subcooled water without melting at the hottest point in the fuel plates.

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 of the three related variables--neutron lifetime, excess reactivity, and exponential period--which characterize the neutron physics of a power excursion, it is the exponential period which determines the total energy release and the temperatures attained. The excess reactivity and neutron lifetime have large effects only as they jointly determine the period. This supposi-

*Actually, the energy data of Reference 1 were revised in Reference 2 because of later and better calibrations of the instrumentation. The numbers above are taken from the later (more pessimistic) data.

tion is consistent, for example, with the observations that the total energy transferred to the coolant water during a power excursion is many times the amount which would vaporize enough water to compensate for the excess reactivity, and that the actual reactivity reduction which occurs during the excursion is much larger than the initial excess reactivity. The extension of the Borax results to the UFTR is made on the basis of this evidence.

It is convenient, first, to treat only the effects of the slightly greater fuel-plate spacing and the slightly lower void coefficient of reactivity of the UFTR relative to the Borax I. Information will also be drawn from the Borax II experiments. The Borax II reactor differed from Borax I in that the coolant-channel thickness was greater in the ratio $\frac{0.264 \text{ in.}}{0.117 \text{ in.}} = 2.26$ and that the calculated void coefficient of reactivity was lower in the ratio

$$\frac{0.10\% \; k_{eff}/\% \text{ void}}{0.24\% \; k_{eff}/\% \text{ void}} = \frac{1}{2.4} = 0.416$$

Both of these differences would be expected to cause a higher energy release per fuel plate in Borax II than in Borax I for a power excursion of given period. The measurements made with subcooled water at periods down to 23 millisec showed that the energy release per fuel plate in Borax II was between 1.7 and 2.0 times that of Borax I, with the smaller ratio applying to the shorter periods (Reference 2). Therefore, it seems quite conservative to assume, in the case of any two reactors, (1) and (2), of the Borax type having a ratio of fuel plate spacings, S_1/S_2, and a ratio of void coefficients of reactivity, C_1/C_2, that the ratio of energy release per fuel plate for a subcooled power excursion of given period will be no greater than $E_2/E_1 = S_2/S_1$ or $E_2/E_1 = C_1/C_2$ whichever is the larger. For the UFTR and Borax I the ratios are:

$$\frac{S_{UF}}{S_{Bo}} = \frac{0.137}{0.117} = 1.17 \qquad \frac{C_{Bo}}{C_{UF}} = \frac{0.24}{0.21} = 1.14$$

It is concluded, therefore, that a Borax reactor having a coolant-channel thickness and a void coefficient of reactivity equal to those of the UFTR would release not more than 1.17 times as much energy per fuel plate as Borax I. The limiting non-melting period for such a reactor would be that which in Borax I gave an energy release of 41/1.17 = 35 MW-sec. The period obtained from Figure 6.1A, corresponding to a total energy release of 35 MW-sec, is 7.7 millisec.

The remaining difference between Borax I and the UFTR is in the composition of the fuel plates. The UFTR plates are thicker; their uranium-aluminum alloy has a somewhat lower conductivity because of the higher uranium concentration, and their aluminum cladding is thinner.

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 because a large fraction of the total energy released is stored in the fuel plate during the important stage of the reactor shut-down. For example, a reactor composed of fuel plates of high heat capacity undoubtedly will experience a larger total energy release, but not necessarily a higher maximum temperature, during a power excursion of given period, than a reactor having plates of low heat capacity.

From examination of the Borax results it seems clear that two distinct phases of the reactor shutdown process occur consecutively and that both may be important in determining the maximum center temperature of a fuel plate. The first phase covers the interval before an important amount of boiling occurs at the fuel-plate surface. During this interval the heat loss to the water is small and the important consideration is evidently the ratio of fuel-plate surface temperature (which determines the start of boiling) to center temperature. For periods in the range under consideration this temperature ratio is theoretically not far from unity (0.76 minimum for a 10-millisec period in Borax I). Experimentally the temperature ratio was unity for periods down to 5 millisec in the Borax I measurements. Since the total effect is small and since the temperature ratio for Borax and UFTR fuel plates should not be much different, the thinner cladding will tend to balance the effect of the poorer "meat" conductivity. It is concluded, therefore, that there will be no important difference in fuel plate performance during this initial phase of the excursion.

The second phase of the power excursion begins when a significant rate of boiling is established at the plate surface. R activity and consequently generation are reduced at a rate which must be a function of the rate at which heat can be transferred into the boiling water. At the same time the transfer of heat into the water removes heat from the fuel plate and limits its temperature rise. The important characteristic of the plate during this phase of the excursion is the heat flux which it can supply to the water for a given temperature difference between the plate center and surface. A figure assumed to be roughly indicative of the relative performance or merit of fuel plates during this phase is the ratio of heat flux to temperature difference under steady-state conditions. This ratio (figure of merit) will overemphasize the difference between fuel plates since the temperature distribution in the plate will be more peaked during a steady-state conduction than during conduction when the general temperature level is rising. The ratio of these figures of merit for Borax I and for the UFTR is

$$\frac{\left[\dfrac{\text{Heat Flux}}{\Delta T_{c-s}}\right]_{\text{UFTR}}}{\left[\dfrac{\text{Heat Flux}}{\Delta T_{c-s}}\right]_{\text{Borax}}} = 0.82$$

A conservative procedure would be to apply the above factor to the permissible total energy of excursion on the Borax I curve. At the same time, however, the difference in gross maximum to average power ratio for the two reactors should be taken into account since it is the temperature of the hottest point in the hottest fuel plate which is being considered. The power ratio for the two reactors is

$$\frac{\dfrac{\text{Max}}{\text{Ave}}_{\text{Borax}}}{\dfrac{\text{Max}}{\text{Ave}}_{\text{UFTR}}} = \frac{1.82}{1.63} = 1.12$$

The combination of these two factors reduces the permissible equivalent energy of the Borax-type excursion to

$$35 \times 0.82 \times 1.12 = 32 \text{ MW-sec.}$$

The corresponding exponential period from Figure 6.1A is 8.3 millisec. It is, therefore, concluded that the UFTR will tolerate a power excursion of period at least as short as 8.3 millisec without the melting of any part of any fuel plate. The excess reactivity corresponding to this period is 2.4 per cent k_{eff}.

Successive Power Excursions

It is typical of the Borax and SPERT reactors, unless the excess reactivity is removed by external means, an initial power excursion which terminates itself by expelling water from the reactor core will be followed by subsequent excursions as the water falls and flows back into the core. An exception to this behavior occurs when the initial excursion is violent enough to cause a permanent loss of reactivity by throwing a large amount of water completely out of the reactor tank. In the UFTR the total quantity of water in the core is small, the submergence of the core is small, and baffles above the core are so arranged that any water splash is directed to the outside so that it cannot return to the core. Consequently, even a relatively mild power excursion (e.g., one having an exponential period of from 20 to 30 millisec) in the UFTR should result in permanent self-induced shutdown of the reactor. By these same design features the possibility of large successive power excursions, such as those studied in the SPERT project, resulting from the ramp addition of excess reactivity is eliminated. It can be anticipated that the UFTR will be safe against quite large ramp additions (larger than 2.4 per cent k_{eff}) provided only that the ramp rate is not so rapid as to add an excess reactivity of more than 2.4 per cent k_{eff} before the reactor power reaches a high level. To exceed this limit the ramp rate would need to be of the order of 1.0 per cent k_{eff} per second or larger.

REFERENCES

F.1 Dietrich, J.R., and D.C. Layman, "Transient and Steady State Characteristics of a Boiling Reactor. The Borax Experiments, 1953," AECD-3840, Argonne National Laboratory, February, 1954.

F.2 Dietrich, J.R., "Experimental Determinations of the Self-Regulation and Safety of Operating Water-Moderated Reactors," A/Conf. 8/P/481, International Conference on the Peaceful Uses of Atomic Energy, June 30, 1955.

F.3 Dietrich, J.R., "Experimental Investigation of the Self-Limitation of Power During Reactivity Transients in a Subcooled, Water-Moderated Reactor. Borax-I Experiments, 1954," AECD-3668, Argonne National Laboratory, August 17, 1955.

CONTENTION VI

RESPONSE TO NRC STAFF'S ASSERTED MATERIAL FACTS

1. "The 10 C.F.R. Part 20 Appendix B release limit for unrestricted areas for ^{41}Ar is 4×10^{-8} µCi/ml."

NOT DISPUTED

2. "The ^{41}Ar releases from the UCLA reactor into unrestricted areas are 3.8×10^{-9} µCi/ml."

DISPUTED

(Foster declaration for VI, P4-8; Lyon declaration for VI, P 20; Pulido declaration for XV, P 4)

3. "The UCLA radiation monitoring system data has been verified by an environmental monitoring program."

DISPUTED

(Foster declaration, P3-26; Lyon declaration, P3-20)

4. "The most conservative interpretation of the UCLA environmental monitoring program is 30 mrem/yr from reactor radiological releases into unrestricted areas."

DISPUTED

(Foster declaration, P 10-11, 13-15, 24-26)

5. "A dose of 30 mrem/yr. is 6% of the permissible level in 10 CFR 20.105(a)."

NOT DISPUTED

Counterfact: A dose of 30 mrem/yr in unrestricted areas of UCLA would violate 10 CFR 20.1(c) and would be the equivalent of members of the public, without their permission or knowledge, receiving the equivalent of an additional chest X-ray per year without any medical need therefor

(Monosson declaration P 18; Lyon declaration, P 16-17; 10 CFR 20.1(c); ICRP Publication 22)

6. "The radioactive emissions from the UCLA research reactor could not be significantly reduced by additional stack height."

DISPUTED

(Pulido declaration for XV, P3-5,7-8,34; March 13, 1975, Response by UCLA to Notice of Violation, second-to-last page, indicating raising stack the 17 feet to the originally required height would increase dispersion and decrease radioactive concentrations in public areas by a factor of five).

RESPONSE TO UCLA'S ASSERTED MATERIAL FACTS

27. "Based on conservative assumptions the maximum argon-41 concentration seen in the past five years at the Mathematical Sciences air intake is 2.6×10^{-9} μCi cm^3."

DISPUTED.

(Foster declaration, ¶8; SAR to Amendment 10 to UCLA license; Reg Guide 1.111)

28. "The highest radiation level on the unrestricted rooftop areas adjacent to the reactor building exhaust stack does not exceed 22 mrem per year above background."

DISPUTED

(Foster declaration, ¶ 10-21; plus the TLD raw data)

29. "The radioactive emissions from the UCLA reactor have been reduced to a level that is as low as reasonably achievable."

DISPUTED

(Foster declaration, ¶3-26; Pulido declaration for XV, ¶ 4-11,34; Lyon declaration, ¶4-5,8,16-17,20)