6/14/83

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

TESTIMONY OF SEAN C. HAWLEY

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

A.1. My name is Sean C. Hawley. I am employed as a development engineer in the Laboratory Safety Department at Battelle, Pacific Northwest Laboratory (PNL), Richland, Washington. A statement of my professional qualifications is attached to my testimony.

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

A.2. My testimony will explain and clarify the analytical bases for the conclusions reached in the document entitled "Analysis of Credible Accidents for Argonaut Reactors" prepared for the U.S. Nuclear Regulatory Commission in April, 1981, designated as NUREG-/ CR-2079 by the Commission and PNL-3691 by Battelle, PNL. I co-authored this document along with two colleagues at the Battelle PNL. I am thoroughly familiar with its contents and adopt it as my testimony. This document has been termed in this proceeding the "Battelle Study" and I shall reference it this way.

Q.3. What particular aspects of your Analysis do you intend to explain or clarify?

A.3. I will explain and clarify the references in the "Battelle Study" to metal-water chemical reactions, the source term, release fraction, dose calculations in the fuel handling accident analysis, graphite-stored Wigner energy and temperature coefficient, and the power excursion models used in the analysis.

Q.4. Please explain the bases for the section titled "Explosive Chemical Reactions" at pp. 28-29 of the "Battelle Study."

A.4. Certain conditions must exist in order for explosive metal-H_2O reactions to occur. Our literature review indicated that the metal must exist in a molten and highly dispersed form, or as finely divided particles. We could not conceive of a credible accident mechanism for an Argonaut-UTR that would produce finely divided aluminum particulates and a resultant metal-H_2O explosion.

The precise conditions in a reactor required to produce appreciable energy from the chemical reaction are unknown, but appear to be a combination of heat and shock wave. The heat would ensure that the metal was in a reactive state (i.e., molten), and the shock wave would disperse the metal.

This reaction cannot occur in the normal operation of the Argonaut-UTR. The post-event analyses of the SPERT destructive test and the SL-1 accident indicate that a metal-H_2O reaction

happened after, or as a result of, the combination of heat generated by the nuclear excursion and the dispersal of molten aluminum by the shock wave. This chemical reaction is not generally considered as initiating a destructive event, because the aluminum has to be molten and dispersed before the explosive reaction occurs. Thus, this reaction needs to be considered only if any of the credible accidents creates an energy release sufficient to melt and disperse the fuel and/or cladding.

The magnitude of the energy release required to initiate such an explosive reaction can be estimated from laboratory experiments on the Al-U fuel plate and water reaction. These experiments showed that the amount of damage and extent of the metal-water reaction varied with the amount of energy deposited per gram of fuel plate material. At 174 Cal/g (0.156 MWs/fuel plate) the first signs of damage and evidence that a reaction had occurred were detected. The type of fuel plate damage associated with a large degree of metal-water interaction, as in the SPERT and SL-1 cases, required about 704 cal/g (0.631 MWs/fuel plate). We could not conceive of a credible accidental power excursion whereby 0.156 MWs/fuel plate could be reached, let alone 0.631 MWs/fuel plate. Furthermore, no other credible accident was conceived that could produce temperatures approaching the melting point of the fuel or other metals in the reactor, and disperse the molten metal.

Q.5. Please explain the source term you calculated for the fuel handling accident analysis in the "Battelle Study," p. 46, Table 3.

A.5. Since typical standard operating procedures for fuel loading or unloading specify that only one fuel element be moved at a time, our source term used only one fuel element. Since the Argonaut-UTR has the capacity for 24 fully loaded fuel elements, a single fuel element would contain, on the average, 1/24 or 4.2% of the total fission products. Because the power density is higher in the central portion of the reactor core, a fuel element from this region would have the greatest uranium "burn-up" and consequently the greatest fission product accumulation. The largest peak to average power density ratio we found in the Argonaut-UTR literature was 1.5. The percent of total fission products that an average fuel element would contain was multiplied by 1.5 to give the percent of total fission product activity that a fuel element from the central portion of the core would be expected to contain, which we took at 7%.

To determine the actual fission product concentration, we needed to assume a value for the length of operation since all other variables, constants and conversion factors, such as power level, fission product yield factors and decay constants for each fission product of interest, were fixed and known. We assumed the reactor operated at 100 kW (full power) for 24 hours per day for 365 days (36.5 MWd) and calculated the fission product activities at the end of this period. For this long period of continuous operation the saturation factor (which combines the production of fission products with their subsequent decay) for all of the fission products in

Table 3, except ^{85}Kr, reaches the maximum value (i.e., one). Readily available conversion factors (see CRC Handbook of Chemistry and Physics, 53rd Ed.; Radiological Health Handbook, 1970) allow transforming the power level into a fission rate which is then multiplied by the appropriate yield fraction and saturation factor to produce the amount of the (particular) fission product. Seven percent of these values (i.e., for the radioiodines and noble gases) will produce the values (labeled Curies at Shutdown) in Table 3 of the Battelle study. Obviously, our assumed power history of full power, continuous operation for one year was very conservative. A typical weekly operation schedule at a small research reactor would consist of one or two long (less than 8 hours) full power runs and one or two runs of shorter duration (e.g., about 2 hours) or at less than full power. Therefore, the typical schedule is intermittent, and accumulates a maximum of about 20 full power hours per week. Over a year then, the reactor would accumulate only 4.33 MWd, and this figure is probably an upper bound given the normal variations in demand for reactor operations over the school year. For ease of calculation (and conservatism) the saturated activities of the radionuclides were used in our calculations. For example, continuous operation of the reactor would have to exceed thirty days to approach 95% of the saturated activity for ^{131}I. The typical intermittent operating schedule is characterized by more time shutdown than operating and this reduces the nominal radioiodine inventory (and the corresponding thyroid dose) below that which we used in NUREG/CR-2079.

As an example of the effect of intermittent operation on the radioiodine inventory consider a regular cycle of 4 hours full power operation at 100 kW followed by 20 hours shutdown. This intermittent cycle would produce an equilibrium value of about 26 curies of ^{131}I (after about 125 days) as compared to the saturated activity of 164 curies from continuous operation. Since the half lives of the other radioiodines are short in comparison to the 20 hour downtime, the maximum activities would be at the end of the operating portion of the cycle. These activities would be about 172 Ci ^{132}I, 50 Ci ^{133}I, 406 Ci ^{134}I and 128 Ci of ^{135}I. This example is intended to emphasize that our assumption of continuous full power operation produces an extremely conservative value for the (radioiodine) source term.

Q.6. Explain the derivation of the 2.7% release fraction you used.

A.6. For the fuel handling accident, we needed a release fraction for a solid piece of metal at room temperature. In most fuel elements a gap getween fuel and cladding creates a space where gaseous fission products will accumulate. In the plate type fuel, where the cladding is in direct contact with the fuel, there is no manufactured gap. However, we postulated that such a gap could exist. The following quote from Nuclear Reactor Safety (edited by F. R. Farmer, Academic Press 1977, page 14) describes the mechanism we used for fission product accumulation.

> In reactors with metallic fuels the principal modes of release of fission products from the fuel material are related to the fission process itself. The considerable kinetic energy of recoil initially possessed by a fission

> fragment results in a proportion of those formed within range of a free surface being released. The majority of these recoil atoms are normally reembedded in the cladding but some will be retained in any fuel/clad interspace. An associated phenomenon known as "knock-out" occurs when a fission fragment displaces atoms of both fuel material and previously formed fission products from the fuel surface. The range of fission fragments, like that of alpha particles, is very small and it is clear that "recoil" and "knock-out" will normally release only a small fraction of the fission products formed in the fuel material. Such release is usually only of interest as the basis of methods for detecting and locating fuel elements with failed cladding.
>
> For modern power reactors fuel with porous ceramic materials operating at higher temperatures than metallic fuels, increased mobility of the fission products within the fuel material subsequent to their formation, either by diffusion or by recrystallization of the fuel material, leads to releases many times larger than those due to recoil and knock-out.

For the purposes of the Battelle study, we used the reported range of fission fragment recoil in aluminum as the maximum depth from which fission fragments could escape. This small distance, when applied to the fuel plate shape, produced a thin shell from which we assumed all the gaseous fission products would escape and accumulate in the fuel/cladding interface.

This condition implicitly assumes that both fission fragments within the recoil distance of the surface accumulate in the gap whereas statistically the actual number would be closer to one.

We also assumed that fission products would be produced uniformly throughout each fuel plate of the fuel element. Although the distribution would not be perfectly uniform any variation would be

minimized since we were also assuming this would be a centermost fuel element. A fuel element from this region should have not only the most uniform fission product distribution but also the highest.

The amount of activity released would depend upon the circumstances of the accident, primarily on the temperature and exposed surface area of the fuel. Even assuming fuel handling immediately following normal operation at full power, the fuel temperature would be low and diffusion would be essentially zero. Hence, any release of fission products would be from the surface of the fuel, or for the purposes of the Battelle study, from a postulated gap.

Estimating the amount of surface area exposed depends on the characteristics of the forces impinging on the fuel element. Jagged breaks or guillotine breaks, cladding delamination that exposes more fuel meat, though not visible, are among the factors to consider. Rather than produce a single estimate or even a range of the amount of surface area likely to be exposed in an accident, all the surface area of the entire fuel element (eleven fuel plates) was used. The nominal dimensions of the fuel meat are 7.23 cm x 0.102 cm x 65.0 cm and assuming an accident in which the entire surface area of all 11 plates in a fuel element was denuded of cladding, the total surface area exposed would be 10,500 cm^2. The volume of fuel from which the radioactivity could escape would be

$$1.37 \times 10^{-3} \text{ cm} \times 10{,}500 \text{ cm}^2 = 14 \text{ cm}^3$$

The total fuel volume for the 11 plates is 525.7 cm^3.

The 2.7% release fraction we used is simply the fraction of the total fuel element volume that 14 cm^3 represents.

This postulates 100% release of the gaseous activity (i.e., radioisotopes of krypton, xenon, and iodine) produced within fission fragment range of the surface.

This is an upper limit value and is equivalent to a 100% gap activity release or 100% cladding failure or postulating that through some unstated mechanism 2.7% of the gaseous activity in a single fuel element would be released.

Because of the low temperature, any release of other fission products, including the semivolatiles, such as strontium, would be negligible. Furthermore, and in contrast to measured release fractions from molten Al-U fuel where only about 5% iodines and 10% noble gases were released from a given amount of fuel, we postulated a 100% release fraction for iodines and noble gases from a given amount of non-molten fuel. Had we assumed the surface "volume" used in our calculations to be molten, the experimental evidence would require that only 5% of 2.7% of iodines and 10% of 2.7% of noble gases would be released. Therefore, our release fraction is very conservative, especially for non-molten fuel.

In a core-crushing accident, more than one fuel element is likely to be damaged. The extent of damage can be qualitatively stated as

being partial. In other words, some fraction (i.e., less than 100%) of the total fuel plate surface area would be exposed. The sum of the exposed fuel meat surface area for damaged elements would most likely not exceed the value used for the single fuel element, i.e., 10,500 cm^2.

This is a most conservative assumption for a single fuel element, but is equivalent to all the fuel elements sustaining partial damage. As demonstration of the extreme conservatism used I will compare the amount of damage, expressed as guillotine, transverse cuts assumed to be sustained by one fuel element (eleven fuel plates) to 24 fuel elements (264 plates) to expose 10,500 cm^2 of fuel meat. The nominal dimensions of the fuel meat give a transverse cross-sectional surface area of (7.23) (0.102) = 0.737 cm^2. Each guillotine cut across a fuel plate would expose 2 surfaces or 1.48 cm^2, and each cut across a fuel element would expose 16.3 cm^2. Dividing the postulated total surface area exposed (10,500 cm^2) by 16.3 cm^2 yields the number of cuts across the fuel element, which is 644. Over the 65.0 cm length of the fuel meat this is equivalent to cutting completely through all eleven fuel plates essentially every millimeter (0.040" or about 3/64"). Thus, the fuel element would have to be completely shredded into strips as wide as the fuel meat is thick, <u>or turned into confetti</u>.

A similar process applied to 24 fuel elements (264 fuel plates) would have (16.3 cm^2 exposed per cut per fuel element times 24 fuel

elements) 391 cm^2 exposed per cut and require 27 cuts (or their equivalent length of 1.03 kilometers) completely through all the fuel elements (all 264 fuel plates) every 2.41 cm (about 1 inch).

Estimating the surface area exposed due to a jagged cut is beyond the scope of this discussion. However, once an acceptable value for the surface exposed per unit length cut is established, a process similar to that used above could then determine the length of the cut, rip or tear. The foregoing discussion is not intended to provide a realistic description of damage to fuel elements, but to show how large 10,500 cm^2 really is when compared to the fuel plate dimension.

Q.7. Please explain the conservatisms in the dose calculations in the "Battelle Study," pp. 47-49.

A.7. The dose calculation derived from the damaged fuel element analysis used several conservative factors. One such factor associated with the source term was the lack of any decay time between removing the fuel element from the reactor and subsequent damage. This assumption is, for a fuel handling accident, essentially incredible and thus very conservative, since some decay of the radionuclides would occur during the time required to remove the fuel element from the reactor. As a research reactor with low demands on operating time, such delays often extend into weeks. A delay time of only one week would reduce the radioiodine inventory considerably. For

example, the ^{131} I would be reduced to 0.55, ^{133}I to 3.2×10^{-3}, and ^{135}I to 2.8×10^{-8} of the original source terms. All the ^{132}I and ^{134}I would have decayed away entirely. The source term itself was conservative since a continuous mode of operation was assumed and the consequent saturation activity of ^{131}I was used in the dose calculation.

The release fraction for radioiodine was actually 100% from the affected fuel volume whereas a more realistic value for non-molten fuel would be much less than the 5% fraction for molten fuel.

In postulating a release to the environment, no credit was taken for any filtration, forced dilution or plate out. Although such factors can be estimated, it would be best to factor these in on a site specific basis. However, such processes will substantially reduce the airborn concentration escaping the reactor room. In addition, if the fuel element were damaged while covered with water, the radioiodine release would be reduced to only a few percent of the dry case.

As discussed earlier, 10,500 cm^2 is a truly conservative number for the area of fuel exposed by damage to a single fuel element. As an example, consider a single fuel element to undergo damage in such a way as to expose 1/24 of the total or 438 cm^2. If some sort of jagged cuts are made, which expose not 1.48 cm^2 per cut, but 14.8 cm^2, then to expose 438 cm^2 would require that each of the eleven fuel plates

in the element be cut into three or four pieces or the equivalent of 296 cm (about 9'8") of guillotine cuts. Therefore, assuming the inventory from an intermittent cycle, the release fraction from molten fuel, a 168 hour decay time and 438 cm^2 of exposed fuel meat, which yields a "volume" release fraction of 0.11% instead of 2.7%, the amount of radioiodines released to the reactor room (from the fuel element with 7% of the core inventory) would be 840 mCi ^{131}I, 12 μCi ^{130}I and 0.24 nCi ^{135}I. With no credit taken for plate out, filtration and dilution, the thyroid dose equivalent is calculated (using the method in NUREG/CR-2079) to be 4.1 mrem, most of which is due to the ^{131}I.

Q.8. How did you account for stored Wigner energy in graphite in your analysis?

A.8. In estimating the amount of stored energy in the graphite, the value of 0.3 cal/gram at 50°C was determined by comparing the slopes of the lines (one for 30°C, one for 50°C) using figure 12-2 on page 329 of Nightingale's book. (Nightingale, R.E. 1962, Nuclear Graphite, Academic Press, New York. Referenced on p. 54 of NUREG/CR-2079).

If equation 12-1 (also on page 329) is used instead and necessary interpolations of the data in Table 12-1 (page 330) are made, the value of stored energy at 50°C for 12 MWd is 7.7 cal/g. This value will increase with additional MWd of exposure and at 57 MWd would reach 22 cal/g.

The above values used a thermal flux of 1×10^{12} n cm^{-2} S^{-1} and the fluence to MWd/aT conversion factor given in Nightingale for use with equation 12-1 and table 12-1.

A thermal flux of 1×10^{12} for a 100 kw reactor represents in my experience, a reasonable maximum value.

The exact number of accumulated MWd as well as the maximum flux will always be site specific and NUREG/CR-2079 used a reasonable approach using what were the most conservative numbers available.

Q.9. Please explain the positive graphite temperature coefficient.

A.9. Based on the information available at the time NUREG/CR-2079 was prepared, the positive graphite temperature coefficient was substantially smaller than the negative water coefficients. Furthermore, during the power excursion or inadvertent transient, the majority of the heat received by the graphite would have to be transferred from fuel to the moderator/coolant to the graphite. The heating would be delayed and consequently the graphite temperature effect on the power excursion would be negligible. Any "prompt" heat due to the fission neutrons and gamma rays that would be delivered to the graphite, would also be delivered to the water moderator/coolant, thus manifesting the negative water coefficients at the same time.

Q.10. Explain your use of data from the SPERT series of transients in your analysis of power excursions in Argonaut-UTRs.

A.10. When reviewing the available literature for prior power excursion or transient accident analyses, the wide variation in methods and results was noted. However, there was often insufficient information presented to analyze or evaluate the methods used. Simply taking ratios of various coefficients, as done in some previous analyses, was not sufficient justification to repeat the method for our report.

The approach used in NUREG/CR-2079 to model the power excursion and energy release was what appeared to be a generic method. This simple, generic method predicts that the total energy released in the excursion is twice the maximum power divided by the reciprocal of the period.

The extensive data collection and analysis of the SPERT series of transients was the best available data on the excursion behavior (i.e., predicting maximum power from a given period or reactivity insertion) for Al-U plate type cores. The particular core with the most extensive analyses was the SPERT 1D core, which was the most similar to the Argonaut-UTR design. The 1A core did have a water channel thickness closer to the Argonaut - UTR than the D core but the fuel element assemblies were more similar between the UTR and D core than between UTR and A core.

Therefore, our method used a simple, generic model and the best available data (i.e., most likely maximum power or energy release for a given period) on excursion behavior for Al-U plate type HEU cores.

In addition, a hot spot factor was scaled linearly from the SPERT destructive test to conservatively estimate such a hot spot for the Argonaut-UTR. The difference in peak to average flux or power values were not factored into the comparison. If these differences are taken into account, the scaled hot spot becomes a factor of 0.625 (1.5/2.4) lower since the SPERT core had a peak to average value of 2.4 while the Argonaut-UTR is a maximum of 1.5. Therefore, instead of a 586°C hot spot for a 12 MWs energy release, it would be 366°C. Since preparing NUREG/CR-2079 we have received other documents detailing the SPERT studies. The following quote is from IDO-16528, 1959, which reported results on the SPERT A, B and P cores. (The A core had fixed fuel plates, 17 per element with plate stiffeners; the B core had fuel elements constructed to allow variation in number of plates per element and the P core was stainless steel fuel plates with 18 per element).

> "In summarizing the transient behavior of the SPERT cores following a step addition of reactivity, the important results can be stated as follows: first, the behavior of all cores on the basis of (reciprocal period) is remarkably similar, notwithstanding the inclusion of both aluminum and stainless steel cores and the wide range of static characteristics involved; and second, those changes that do exist are predicted by an elementary theory which incorporates two experimentally determined quantities -- the average void coefficient and the neutron lifetime. The power - or energy - ratio at a constant (reciprocal period) has been shown to vary inversely as the

> square root of the void coefficient, divided by the (reduced prompt) neutron lifetime. Also, on the basis of reactivity, increasing the void coefficient will result in a core that releases less energy."

Simply following the process given in IDO-16528, the ratio of energy release for the SPERT 1D core to the Argonaut-UTR core is 0.582. Thus, for a period of 7.2 msec, the SPERT 1D core produced 9.32 MWS total energy release and the Argonaut-UTR core, by the above process, would produce 16.0 MWs.

Using the same method in NUREG/CR-2079 to calculate the adiabatic temperature rise, 16 MWs would produce an average increase of 320°C and an estimated maximum increase of 528°C in the center most fuel plates. Even with an initial temperature of 60°C, this would yield a final temperature of 588°C, still well below the fuel meat melting point.

It should be remembered that these temperatures are calculated assuming that all the energy generated in the fuel plates remains there to cause the (maximum) temperature increase. There will be heat flow to the coolant/moderator and thus actual temperatures will be lower than those calculated for a no heat flow or adiabatic case.

In conclusion, consideration of void coefficient differences between the SPERT and Argonaut-UTR produces a somewhat higher energy release than previously calculated, but still well below the fuel melting point.

Q.11. Please explain the statement on p. 26 of The Battelle Study which states that consequences of core crushing would be some multiple of the consequences of the fuel handling accident.

A.11 By that statement we were comparing realistic assessments of accidents. The sentence on p. 26 should be clarified to read: "The similar consequences from a core-crushing accident would be some multiple of the consequence of the realistic fuel-handling accident."

Q.12 Describe the differences between the Argonaut and the SPERT reactors.

A.12 The Argonaut is a research reactor, licensed for steady state operation, using graphite and water moderation/reflection with essentially fixed core parameters. The SPERT reactors were test reactors, capable of pulsing, using only water moderation/reflection and had various cores.

Q.13 Does the rabbit tube create a danger of rapid insertion and removal of reactivity?

A.13 I do not believe it creates a <u>danger</u> although the reactivity can certainly be removed or inserted rapidly. Given the design of typical rabbit systems, the entire sample would have to be filled with elements such as boron, cadmium, rare earths, or uranium (235) to produce a significant, but not dangerous, effect.

Q.14 Has the increase in excess reactivity and power limits since the original Argonaut development decreased the margin of safety?

A.14 I do not consider the margin significantly decreased because a) I do not believe there is a credible means to accidentally insert 2.6% Δ k/k stepwise and b) although the power increase will generally increase the amount of fission products present at any given time, it is not sufficient to cause fuel melt due to decay heat.

SEAN C. HAWLEY, Research Scientist, Health Physics Technology Section, Radiological Sciences Department

Education

B. A., Chemistry, Reed College	1978
Washington State University	1979-1980
Joint Center for Graduate Studies (University of Washington)	1980-Present

Experience

Mr. Hawley has eight years of experience working with research reactors and one year of experience evaluating emergency preparedness at nuclear power plants.

- Reactor Operation and Supervision. Mr. Hawley received his first Senior Operator's Permit in 1973 for the Reed College Reactor Facility. In his six years there, he was a senior reactor operator, assistant health physicist, reactor supervisor, and training supervisor. In 1979, he received his second Senior Operator's Permit for the Washington State University reactor. As reactor supervisor, he was responsible for the safe operation and maintenance of the reactor, and also advised and instructed researchers in the methodology of neutron activation analysis.

- Accident Analysis and Emergency Preparedness. Since joining Battelle, Mr. Hawley has been analyzing credible accidents for research reactors and participating in emergency prepardness appraisals and exercise observations at nuclear power plants. He is also reviewing emergency plans for other NRC-licensed facilities.

Mr. Hawley is a member of the American Chemical Society and the Health Physics Society.

Publications

Hawley, S. C., R. L. Kathren and M. A. Robkin. 1981. Analysis of Credible Accidents for Argonaut Reactors. NUREG/CR-2079, PNL-3691, Pacific Northwest Laboratory, Richland, Washington.

Hawley, S. C., and R. L. Kathren. 1982. Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors. NUREG/CR-2387, PNL-4028, Pacific Northwest Laboratory, Richland, Washington.