UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	Docket No. 50-170
ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE	(Renewal of Facility License No. R-84)
(TRIGA-Type Research Reactor)	

LICENSEE'S ANSWERS TO

INTERVENOR'S INTERROGATORIES

Licensee submits these Answers to Intervenor's Interrogatories under the provisions of 10 C.F.R. 2.740(b). The Answers are arranged substantially in accordance with the format suggested by the Intervenor. The answer to each interrogatory is constructed as follows: the entire interrogatory is reprinted; the surname of each person with substantive input to the answer is listed; a direct answer to the question is provided (labelled "A"); and the references (if any) relied upon in formulating the answers are listed (labelled "B"). The composite list of the names of the people and their titles is: Mark Moore, Chief Supervisory Operator, AFRRI Reactor; Joseph A. Sholtis, Physicist-In-Charge, AFRRI Reactor; Leonard Allen Alt, Nuclear Physicist, AFRRI Reactor; Harry Spence, Senior Reactor Operator, AFRRI Reactor; Robert Loesch, Head, Radiation Health Physics Division, AFRRI; William R. Webber, Head, Radioanalysis and Dosimetry Division, AFRRI; John Arras, Supervisory Health Physicist, National Bureau of Standards; Frank J. Munno, Professor and Program Director, Nuclear Engineering Dept., University of Maryland; and Ronald R. Smoker, Chief, Radiation Sources Division, AFRRI

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No response has been included for the matters identified as "C" in the preface to the Interrogatories. It is unreasonable to require a complete listing of "... all documents and studies, and the particular parts thereof, known to exist but not relied upon..." since the question virtually by definition would require the listing of irrelevant material. Obviously, the production of such a listing would also be an unreasonable burden given the tremendous quantity of literature which has been published. Hence, no answer is required under 10 C.F.R. 2.740(b). Moreover, given the listing of documents that were relied upon under "B" with respect to each question, the information requested becomes as readily available to the Intervenor as it is to the Licensee.

If AFRRI is engaged or intends to engage in further research which may affect the answer to a question, it will be so stated in the answer. Future plans of the NRC staff and others are not known to AFRRI and thus no answers are provided as to the second part of "D".

As of the present the Licensee has not determined which, if any, expert witnesses will be called to testify on any question. Hence, no answers are provided for "E".

1. State the scientific basis for your assumption in the Hazard Summary Report's (HSR) analysis of a "Fuel Element Clad Failure Accident," submitted with your license renewal application, that cladding failure during a pulse operation or inadvertent transient would occur at a peak fuel element temperature of less than 100°C.

Answer to Question 1:

Answered by: Sholtis, Moore, Smoker

A. AFRRI does not make an assumption that "cladding failure during a pulse operation or inadvertent transient would occur at a peak fuel element temperature of less than 100°C." On the contrary, the assumed fission product release fraction used for analysis of the fuel element cladding failure accident (i.e., 0.1%) corresponds to the theoretical maximum release fraction at 600°C as well as the expected release fraction based on experiments for cladding failures at 1000°C.

The licensee addresses CNRS, Inc. to AFRRI's 1981 SAR, specifically Chapter VI of the SAR, "Safety Analysis," also known as the "Hazards Summary Report," already provided to CNRS, Inc., for verification of this data.

B. References relied upon:

1. Safety Analysis Report (SAR) for the AFRRI-TRIGA Reactor, Facility License R-84, Chapter VI, "Safety Analysis," 12 May 1981.

2. Simnad, M. T., The U-ZrH_x Alloy: Its Properties and Use in TRIGA Fuel, General Atomics Report No. 4314, General Atomics, San Diego, CA, February 1980.

3. Simnad, M. T., et al, "Fuel Elements for Pulsed TRIGA Research

Reactors," Nuclear Technology, Vol. 28, No. 31, January 1976, pp. 31-56.

C. See general statement.

D. See general statement.

E. See general statement.

2. State the calculations from which you derived your conclusion in the HSR that a contact configuration of the twelve elements stored in your spent fuel pool would not result in a critical mass.

Answer to Question 2:

Answered by: Sholtis, Alt

A. First, there are three errors implicit in your statement of question #2: 1) AFRRI does not have a spent fuel pool, 2) AFRRI presently does not have twelve elements stored within the storage racks, and 3) a "contact" configuration is not what is specified as the worst neutronic configuration for TRIGA fuel.

Regardless of these implicit errors in stating question #2, AFRRI's 1981 Safety Analysis Report (SAR), dated 12 May 1981, (and copy previously provided to CNRS Inc.) does cite a reference which conservatively determines that twelve AFRRI fuel elements cannot result in a critical mass under any condition. See page 4-29 of AFRRI's Safety Analysis Report (SAR), dated 12 May 1981, where it states that, "Conservative calculations show that in the event of a fully loaded storage rack failure where all 12 fuel elements fall to the bottom of the reactor tank in an optimal (i.e., worst case) neutronic geometrical configuration, a criticality excursion would not result." The reference cited in this statement appears as reference #2 on page 4-32 of the SAR and is identified as reference 1 under section B. below for your information and a copy of this analysis is attached.

B. 1. Sholtis, Joseph A., Jr., Capt, USAF, Nuclear Criticality Safety Analysis of Hypothetical AFRRI-TRIGA Fuel Element Storage Rack Accidents, AFRRI/SSD Memorandum for Record, January 19, 1981.

2. Safety Analysis Report (SAR) for the AFRRI-TRIGA Reactor, Facility License R-84, 12 May 1981, pp. 4-29 and 4-32.

3. Paxton, H. C., Thomas, J. T., Callihan, D., and Johnson, E. B, Critical Dimensions of Systems Containing U-235, Pu-239, and U-233, TID-7028, Los Alamos Scientific Laboratory and Oak Ridge National Laboratory, Oak Ridge, TN, June 1974.

C. See general statement.

D. See general statement.

E. See general statement.

Attach to question #2 response
(5 pages)

SCIENTIFIC SUPPORT DEPARTMENT

MEMORANDUM FOR RECORD: 19 January 198[illegible]

SUBJECT: Nuclear Criticality Safety Analysis of Hypothetical AFRRI TRIGA Fuel Element Storage Rack Accidents

1. An analysis was performed to substantiate that a criticality excursion would not result in the unlikely event that a fully-loaded AFRRI fuel element storage rack were to fail.

2. For the purposes of analysis, it is conservatively assumed that when the storage rack fails, all twelve fuel elements contained in the rack escape and fall to the bottom of the pool. In addition, it is conservatively assumed that the twelve fuel elements come to rest at the bottom of the pool in the most reactive neutronic configuration possible. Moreover, it is conservatively assumed that the optimum configuration of fuel elements at the bottom of the reactor tank is fully reflected by water over a complete solid angle of 4π steradians even though only 2π steradian water reflection would actually exist.

3. Fuel elements used in the AFRRI reactor are standard stainless-steel clad TRIGA elements containing $U\text{-}ZrH_{1.7}$ with 8.5 weight percent uranium at a nominal U^{235} enrichment of 20 percent (See Figure 1). Each fuel element contains a nominal maximum 38 grams of U^{235}.

4. Figure 2, reproduced from TID-7028 (1), is based on experimental and analytical data and indicates that the minimum critical mass, $m_{crit.}$, for a heterogeneous, 20% enriched, fully water reflected U^{235} system in its most reactive configuration,* is 1.1 kg of U^{235}. Since our assumed twelve element configuration contains a total of (12 fuel elements) X (38 grams U^{235}/fuel element) = 456 grams U^{235}, it would have a mass fraction critical, $m/m_{crit.}$, less than or equal to 0.456 kg U^{235}/1.1 kg U^{235} or 0.415.

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*For our assumed system, this conservative assumption not only takes into consideration an optimum reactive geometry but also neglects parasitic neutron capture in the stainless-steel clad, Sm-Al burnable poison wafers, etc., and assumes that the graphite end reflectors are replaced by water - a more effective neutron reflector.

Using: $k_{eff} = \sqrt[3]{m/m_{crit.}}$, (2) indicates that our assumed system would have a $k_{eff} \leq 0.746$. Therefore, even with the application of the most conservative assumptions, our assumed system would still not achieve criticality. In fact, if our assumed system had a $k_{eff} = 0.746$, then it would be subcritical by more than \$36.00 (assumes $\beta_{eff} = 0.007$).

Based on the minimum critical mass, $m_{crit.}$, value of 1.1 kg U^{235} obtained from Figure 2, and a U^{235} fuel loading per element of 38 gm U^{235}, a minimum of 29 AFRRI TRIGA fuel elements arranged in an optimum neutronic configuration would be required for a criticality excursion (≈\$ .09) to occur.

5. Verification of the conservatism of this analysis is provided by data in RSD 5-8[3]. That is, experience has shown that, during actual AFRRI core loading, ~69 stainless-steel TRIGA fuel elements (~2630 grams U-235) are required to achieve criticality. Therefore, since the AFRRI core lattice arrangement is very close to the optimal neutronic geometry for TRIGA fuel elements, the results of this criticality analysis are conservative by a factor of ~2.4 on a fuel element as well as U-235 mass basis for criticality.

6. In summary, a hypothetical AFRRI fuel element storage rack failure is analyzed from a nuclear criticality safety standpoint. Conservative assumptions are applied wherever possible; yet k_{eff} and $m/m_{crit.}$ for the system are found to be no greater than 0.746 and 0.415, respectively. As a result, there is no possibility of a criticality excursion in the unlikely event that a fully-loaded fuel storage rack were to fail in the AFRRI TRIGA reactor facility.

3 Encls
1. Fig. 1
2. Fig. 2
3. References

JOSEPH A. SHOLTIS, Jr.
Capt, USAF
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Radiation Sources Division
Scientific Support Department

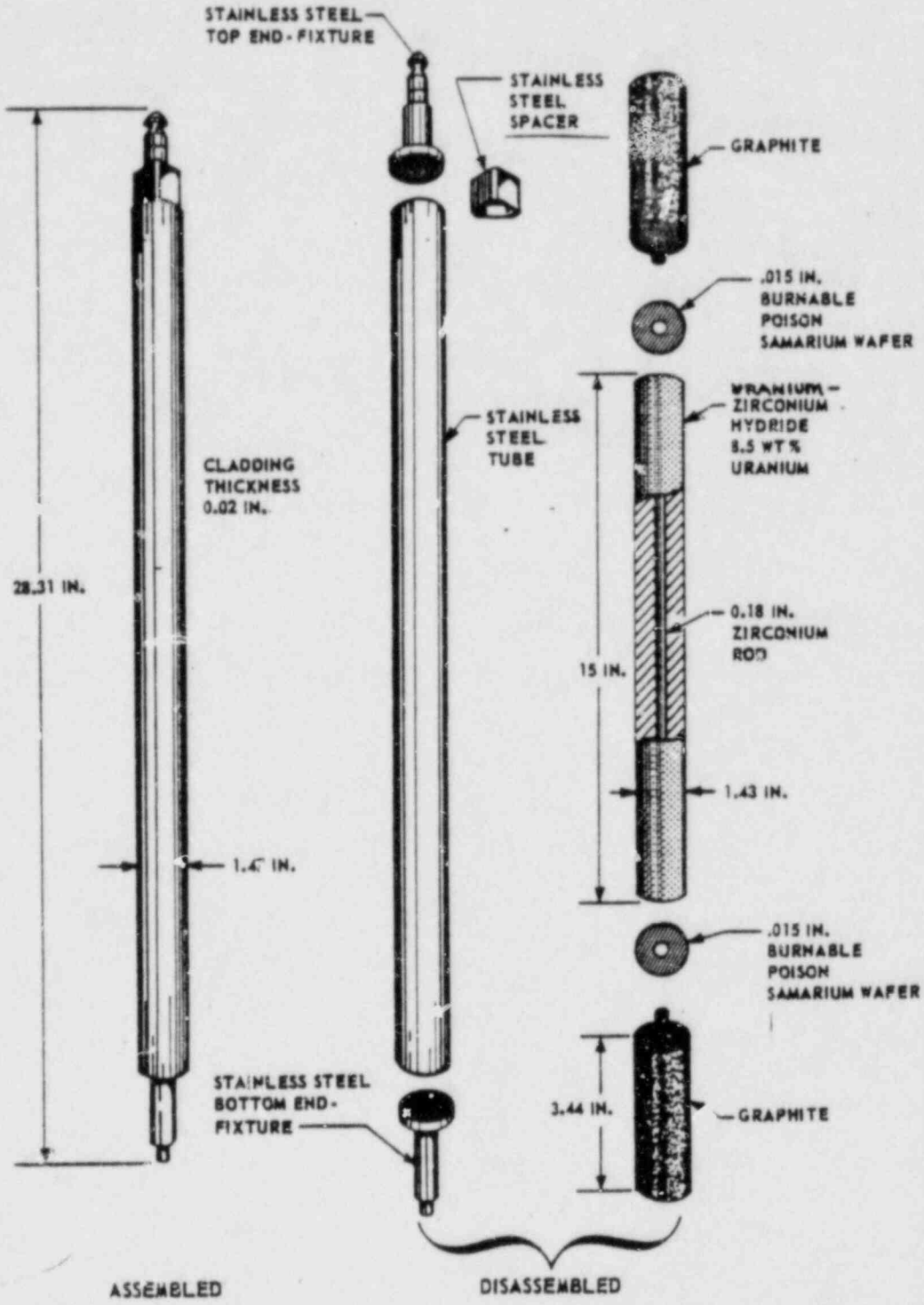

Figure 1. Standard AFRRI TRIGA Fuel Element

Encl 1

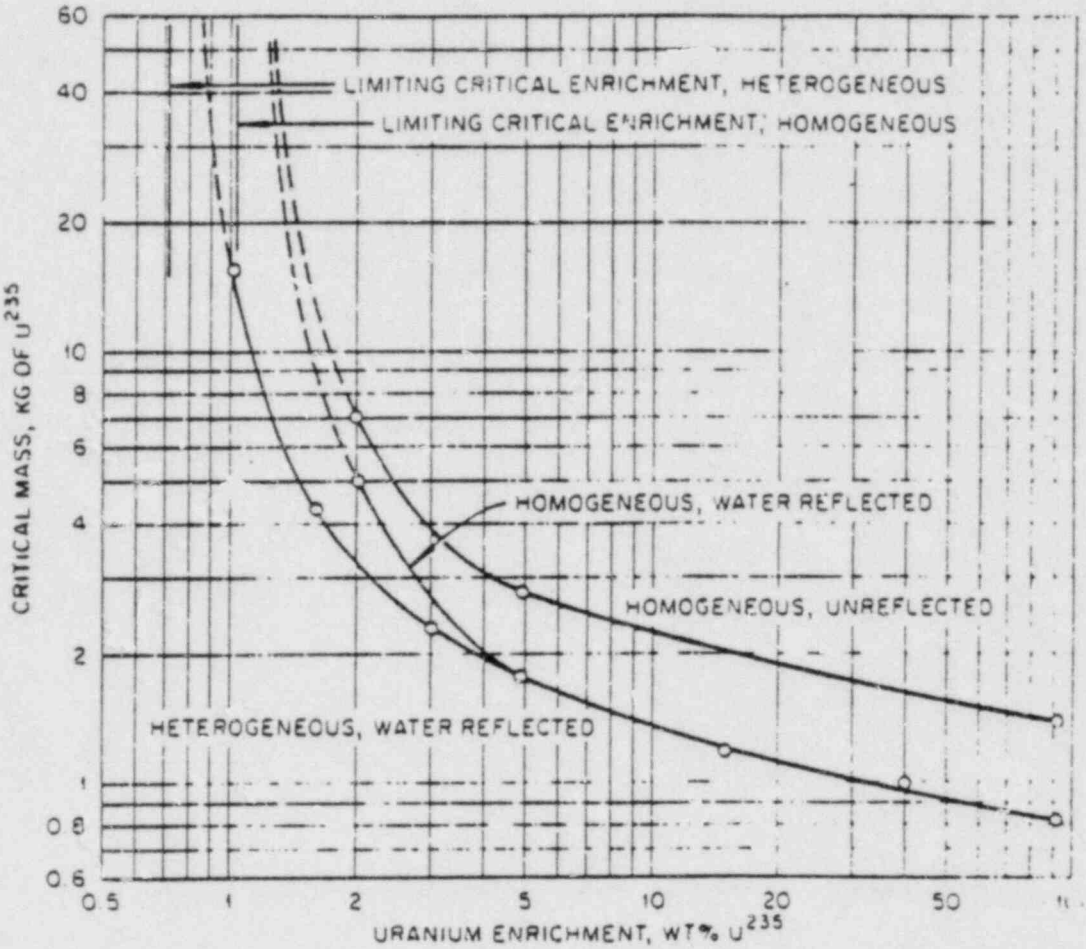

Fig. 2. Minimum critical mass as a function of U^{235} enrichment in hydrogen-moderated systems

Encl 2

REFERENCES

1. Paxton, H.C., Thomas, J.T., Callihan, D., and Johnson, E.B., Critical Dimensions of Systems Containing U^{235}, Pu^{239}, and U^{233}, TID-7028, Los Alamos Scientific Laboratory and Oak Ridge National Laboratory, Oak Ridge, TN, June 1964.

2. O'Dell, R.D. (editor), Nuclear Criticality Safety, compendium of information presented at the Biannual Nuclear Criticality Safety Short Course in Taos, NM by the University of New Mexico, May 1973, published by Technical Information Center, Office of Information Services, U.S. Atomic Energy Commission, Washington, D.C., 1973.

3. Radiation Sources Division Instruction, RSD 5-8, AFRRI/SSRS.

Encl 3

3. State the source(s) you relied on for your statement in the HSR that it takes approximately 67 closely packed fuel elements to achieve criticality.

Answer to Question 3.

Answered by: Sholtis, Moore, Smoker

A. This reference of experience is contained within AFRRI's internal Radiation Sources Division Instruction, RSD 5-8, "Reactor Core Loading and Unloading Procedures" and states that, "AFRRI-TRIGA Core II (stainless steel clad elements) attained criticality with 69 fuel elements, 2630 grams Uranium-235." This statement is based on actual core loading experience at AFRRI using the standard 1/M approach to critical procedure. The actual number of AFRRI-TRIGA fuel elements required to achieve criticality in the core may vary slightly (i.e.~1 to 2 fuel elements) depending on the loading order actually used.

B. RSD 5-8, 'Reactor Core Loading and Unloading Procedures," AFRRI/SSRS, 27 March 1981. A copy of this document is on file with the USNRC, Region I Field Office.

C. See general statement.

D. See general statement.

E. See general statement.

4. What basis, if any, do you have for believing that the following malfunctions of confinement safeguards at AFRRI could not occur during an experiment failure:

(a) a breach of containment caused by missing or inadequate rubber gasket sealing material on the double doors to the corridor behind the reactor control room;

(b) failure of the reactor room ventilation dampers to close as designed;

(c) failure of the lead shielding doors to stop opening at the fully opened position;

(d) malfunction of the reactor core position safety interlock.

Answer to Question 4:

Answered by: Moore, Sholtis

A. AFRRI does not understand what is meant by an "experiment failure" and therefore cannot provide a definitive answer to this interrogatory. Question #10a. of AFRRI's interrogatories to intervenor sought additional information on the definition for this term. Nevertheless, the first two cited malfunctions (a and b) are associated with confinement safeguards at AFRRI but would not occur during an experiment for the following reasons.

1) The operation and integrity of the safeguards are checked under the preventive maintenance program.

2) Draft gauges and visual observations allow a check of the effectiveness of gasket material and damper closure.

3) Daily operational checks of the damper system prior to performing experiments insures proper operation.

The second two cited malfunctions (c and d) have no connection with confinement safeguards at AFRRI. However, neither of the two can occur during an experiment because:

1) Any movement, past the fully open position or otherwise, of the lead shield doors will prevent the performance of an experiment or terminate (by a reactor scram) any experiment already in progress.

2) Any movement of the reactor core (necessary to engage the core position safety interlock) will prevent (by physical interlock) the performance of an experiment or terminate (by reactor scram) any experiment already in progress.

B. No references used.

C. See general statement.

D. See general statement.

E. See general statement.

5. What steps have you taken to prevent the recurrence of the following malfunctions which have occurred at AFRRI:

(a) malfunction of Safety Channel One on March 15, 1980. An NRC inspection on March 17, 1980, "revealed that Safety Channel One would not initiate a scram in accordance with [Applicant's] Technical Specifications";

(b) reactor exhaust system malfunction on August 9, 1979 caused by an electrical fire in the EF-1 cubicle of the motor control center, in turn caused by a power surge due to a faulty transformer;

(c) malfunction of the fuel element temperature sensing circuit caused by a "floating signal ground," reported by DNA on August 1, 1979;

(d) malfunction of the pool water level sensing float switch caused by wear on the jacketing around the wires leading to the switch, reported by DNA on July 31, 1979;

(e) malfunction of Radiation Monitoring System caused by two loose wires in the control box and resulting in a failure of the reactor room ventilation dampers to close (on August 26, 1975);

(f) malfunction of the Fuel Temperature - Automatic Scram System on January 29, 1974, caused by a build-up of high resistance material on the mechanical contacts of the TZ output meter;

(g) malfunction of the Reactor Core Position Safety Interlock System on February 1, 1973, caused by a faulty de-energizing relay.

Answer to Question 5:

Answered by: Moore, Sholtis

A. One cannot prevent with absolute certainty the malfunction of any physical system; one can only reduce the likelihood of malfunction and

mitigate any adverse effects should they occur. AFRRI has an extensive program of preventive maintenance and checkout procedures to insure malfunctions are detected and corrected prior to reactor operations. In each of the cited examples, the malfunctions were detected either during pre-operational checks by the reactor staff or were observed by the operator such that there were no adverse effects from any of the malfunctions. Each of the cited examples will be treated separately to explain the source of detection and the action taken.

Example (a) Malfunction discovered during routine pre-operational checkout. The channel is one of two redundant and one of four safety channels provided. The channel was repaired. Your example is wrong in fact. The malfunction was reported to the NRC, as are all malfunctions, by AFRRI. It was not revealed by an NRC inspection.

Example (b) Malfunction occurred prior to operation and discovered during a daily system checkout. A new power breaker (electrical) was installed, the system was then checked and returned to service.

Example (c) Malfunction was observed by the reactor operator; the floating ground was corrected by relocating a ground strap. The channel was returned to service.

Example (d) Malfunction discovered during routine maintenance check. A new unit was installed such that no wear would occur on the leads.

Example (e) Malfunction discovered during daily pre-operational checkout procedures. The wires were tightened, system checked, and returned to service.

Example (f) Malfunction discovered during daily pre-operational startup checkout. (The correct designation for the meter is T2 not TZ and the buildup of material occurred on a set of relay contacts not the meter.) At the time of the malfunction, the relay contacts were cleaned, checked, and returned to service. More recently (1978) this system was upgraded.

Example (g) Discovered by operator, the relay system was replaced and new operational procedures were effected to perform a redundant check of this system.

B. AFRRI Annual Reports 1975-1980, available in Docket 50-170.

AFRRI to NRC Malfunction Reports, various dates, available in Docket 50-170.

C. See general statement.

D. See general statement.

E. See general statement.

6. What are the scientific and mathematical bases for your assumption that the TRIGA's negative temperature coefficient will automatically shut down the reactor in accident situations with damaged fuel elements?

Answer to Question 6:

Answered by: Sholtis, Munno, Moore, Smoker, Alt

A. First, this is not an assumption; it is fact, based on physical laws of nature!

In layman's terms, a negative temperature coefficient of reactivity simply means that as the fuel temperature increases, negative reactivity is inserted causing the neutron population and, thus, the reactor power level to decrease. For pulsing operations in a TRIGA reactor, prompt heating of the U-ZrH_x fuel causes the introduction of an overwhelming amount of negative reactivity which automatically terminates the power excursion. This effect is intrinsic to the U-ZrH_x fuel and acts automatically primarily due to the presence of hydrogen in the U-ZrH_x fuel-moderator matrix material. The hydrogen in the U-ZrH_x acts to thermalize (i.e. slow down) neutrons so that fission is more likely to take place. Basically, as the fuel heats up, the hydrogen becomes thermally excited and neutrons which are undergoing thermalization cannot reach an energy less than the equilibrium energy of the thermally excited hydrogen. As a result, the neutron energy spectrum becomes harder with increasing fuel temperature and the fission rate decreases—a negative reactivity effect.

If hydrogen is somehow removed from the fuel elements, then neutron thermalization is reduced and the neutron energy spectrum also becomes harder—a negative reactivity effect. In fact, if all of the hydrogen

somehow were removed from the TRIGA fuel elements, achievement of criticality itself would be impossible since ~80% of the neutron thermalization that normally occurs does so by virtue of elastic collisions of the epithermal neutrons with the hydrogen in the U-ZrH_x fuel.

Therefore, regardless of what is meant by "damaged" fuel elements in the statement of this question, the reactor's negative temperature coefficient of reactivity will either remain unchanged, at worst, or become even stronger in a negative sense.

Numerous references describing reactor kinetics, the negative temperature coefficient of reactivity, and its effect in TRIGA fuel are available in the open literature. Some are identified under B below. Moreover, everytime a TRIGA reactor is pulsed and the power excursion terminates itself constitutes proof of its intrinsic efficacy.

A more detailed description of the prompt negative temperature coefficient of reactivity in TRIGA fuel is attached for your information.

B. 1. Hetrick, D. L. (ED), Dynamics of Nuclear Systems, University of Arizona Press, Tucson, AZ, 1972.

2. Scalettar, R. and West, G. B., "Calculations of the Temperature Coefficient and Kinetic Behavior of TRIGA," General Atomics Report No. GA-4474, 1963.

3. Sholtis, J. A., and Moore, M. L., "Reactor Facility, Armed Forces Radiobiology Research Institute," AFRRI Technical Report No. AFRRI TR81-2, May 1981.

4. Profio, A. E., Experimental Reactor Physics, John Wiley and Sons, Inc. Publishers, New York, NY, 1976.

5. Keepin, G. R., Physics of Nuclear Kinetics, Addison-Wesley Publishing, Reading, MA, 1965.

6. Glasstone, S. and Sesonske, A., Nuclear Reactor Engineering, Van Nostrand Publishing, Princeton, NJ, 3rd Edition, 1967.

7. Lamarsh, J. R., Introduction to Nuclear Reactor Theory, Addison-Wesley Publishing, Reading, MA, 1966.

8. El-Wakil, M. M., Nuclear Power Engineering, McGraw-Hill, New York, NY, 1962.

9. Duderstadt, J. J., and Hamilton, L. J., Nuclear Reactor Analysis, Wiley and Sons Publishing, New York, NY, 1976.

10. Chastain, J. W., Jr. (ED), U.S. Research Reactor Operation and Use, Addison-Wesley, Reading, MA, 1958.

11. Simnad, M. T., The U-ZrH_x Alloy: Its Properties and Use in TRIGA Fuel, General Atomics Report No. 4314, February 1980.

12. Simnad, M. T., et al, "Fuel Elements for Pulsed TRIGA Research Reactors," Nuclear Technology, Vol. 28, No. 31, January 1976, pp. 31-56.

13. Whittemore, W. L., et al, "Stability of U-$ZrH_{1.7}$ TRIGA Fuel Subjected to Large Reactivity Insertions," General Atomics Report No. GA-6874, 1965.

14. Lillie, A. F., et al, "Zirconium Hydride Fuel Element Performance Characteristics," AI-AEC-13084, Atomics International, 1973.

15. Leadon, B. M., et al, "Measurements and Calculations of Hydrogen Loss from Hydrided Zirconium-Uranium Fuel Elements During Transient Heating to Temperatures Near the Melting Point," Transactions American Nuclear Society, Vol 8, No. 547, 1965.

16. Beyster, J. R., et al, "Neutron Thermalization in Zirconium Hydride," General Atomics Report No. GA-4581, 1963.

17. Beyster, J. R., et al, "Measurements of Neutron Spectra in Water, Polyethylene, and Zirconium Hydride," Nuclear Science and Engineering, Vol. 9, No. 168, 1961.

18. West, G. B., et al, "Kinetic Behavior of TRIGA Reactors," General Atomics Report No. GA-7882, 1967.

19. Vasil'ev, G. A., et al, "Space Energy Distribution of Reactor Neutrons in Metal Hydrides," Vopr. Fiz. Zashch. Reacktorov, Vol. 5, No. 91, 1972.

20. Gietzen, A. T., "Developments in TRIGA Reactors and Fuel," Paper presented at the Sixth European Conference of TRIGA Reactor Users, Mainz, Germany, 16-18 September 1980.

21. Stone, R. S., et al, "Transient Behavoir of TRIGA, A Zirconium-Hydride, Water-Moderated Reactor," Nuclear Science and Engineering, Vol. 6, 1959, pp 255-259.

22. McReynolds, A. W. et al, "Neutron Thermalization by Chemically Bound Hydrogen and Carbon," Proceedings U.N. 2nd Int'l Conf. on Peaceful Uses of Atomic Energy, Geneva, Switzerland, 1958, p. 1540.

23. Merten, U. et al, "Uranium-Zirconium Hydride Fuel Elements," Paper presented at the 1st Int'l Symp. on Nuc. Fuel Elements, Columbia Univ., New York, 1959.

24. Coffer, C. O., et al, "Characteristics of Large Reactivity Insertions in a High Performance TRIGA U-ZrH Core," General Atomics Report No. GA-6216, April 1965.

25. Scalettar, R., "Kinetics of TRIGA, Part I: Fundamentals," General Atomics Report No. GA-2599, January 1962.

26. AFRRI Final Safeguards Report, Chapter V, Nuclear Analysis, March 1962.

27. AFRRI-TRIGA Reactor Safety Analysis Report, Facility License R-84, 12 May 1981.

C. See general statement.

D. See general statement.

E. See general statement.

Attach to questions re license (3 pages)

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APPENDIX C

A BRIEF DISCUSSION OF THE TRIGA PROMPT NEGATIVE TEMPERATURE COEFFICIENT OF REACTIVITY

Reactors fueled with TRIGA U-ZrH fuel-moderator elements exhibit a strong prompt negative temperature coefficient of reactivity. For the stainless steel clad $U\text{-}ZrH_{1.7}$ fuel, the temperature coefficient is -1.26×10^{-4} $\delta k/k$ per ^{o}C. There are several factors contributing to the prompt coefficient as noted below:

RELATIVE MAGNITUDE OF CONTRIBUTING COMPONENTS OF THE PROMPT NEGATIVE TEMPERATURE COEFFICIENT OF TRIGA REACTORS

	$U\text{-}ZrH_{1.0}$, Al Clad (%)	$U\text{-}ZrH_{1.7}$, SS Clad (%)
1. Cell increased disadvantage factor with increased fuel temperature leading to a decrease in neutron economy	40	60
2. Irregularities in the fuel lattice due to control rod positions—essentially same effect as 1 above	10	10
3. Doppler broadening of U^{238} resonances—increased resonance capture with increased fuel temperature	20	15
4. Leakage—increased loss of thermal neutrons from the core when the fuel is heated[a]	30	15

[a] The low-hydride core is assumed to be reflected by graphite radially, whereas the high hydride core is water reflected radially. The graphite reflector gives ~30% more negative contribution to the leakage component for either core.

From the above it is seen that the dominant contribution to the TRIGA temperature coefficient is the cell effect. It should be noted that the cell effect is also referred to as the warm-neutron effect, the anti-moderation effect, and the Dyson effect. The cell effect is associated with the change in the thermal spectrum caused by heating of the zirconium hydride moderator. This effect can be explained by assuming that the hydrogen-atom lattice vibrations can be described by an Einstein model with a characteristic energy $h\nu = 0.140$ ev. This description is consistent with the theory that the hydrogen atom occupies a lattice site at the center of a regular tetrahedron of zirconium atoms.

The basic consequences of this model, which have been experimentally verified, are:

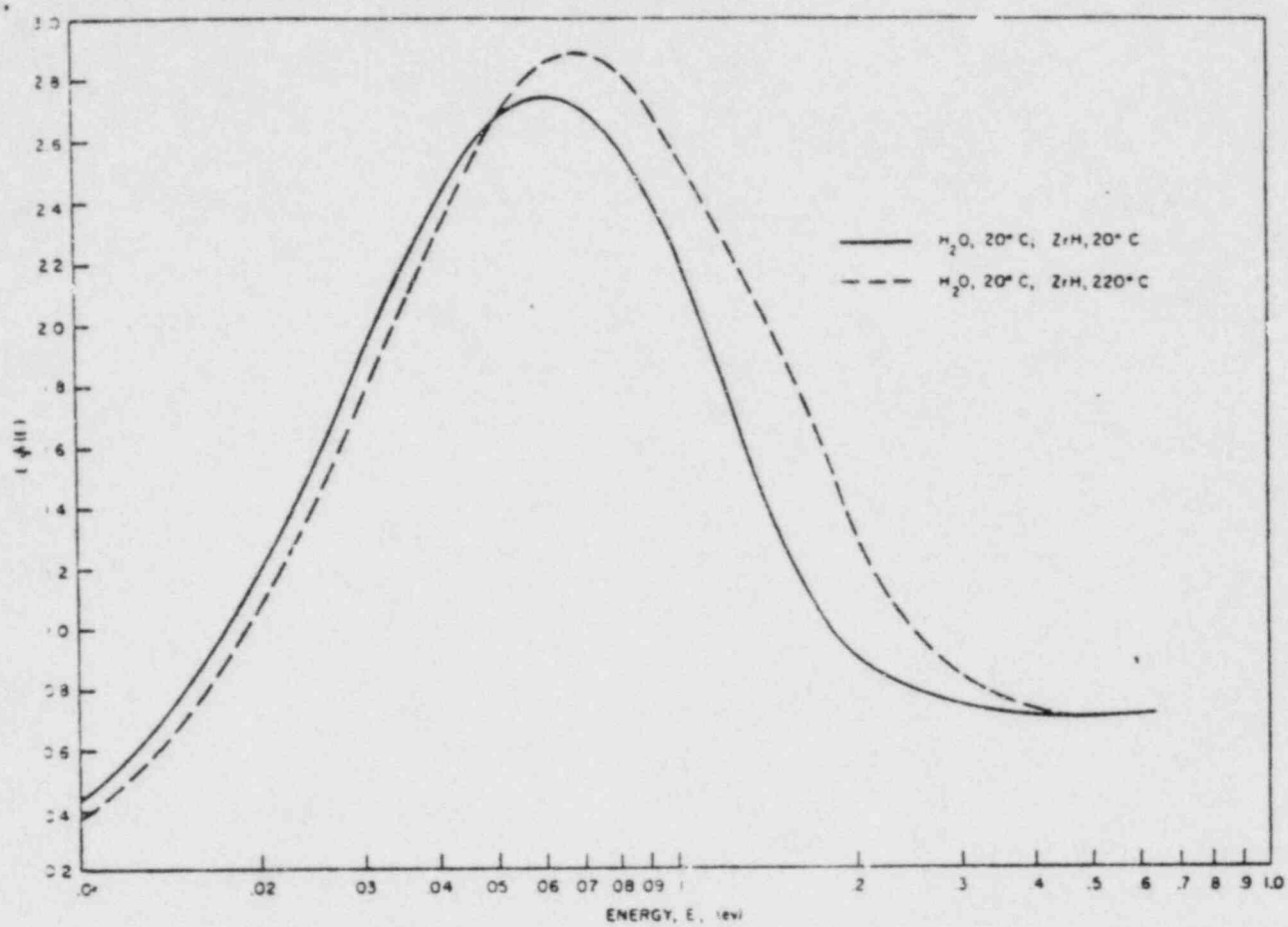
1. Neutrons with energies less than $h\nu$ cannot lose energy in collisions with zirconium hydride;

2. A slow neutron can gain energy $h\nu$ in a collision with zirconium hydride with a probability $\exp(-h\nu/hT)$ which increases very rapidly with temperature.

It is seen that the basis for the strong TRIGA coefficient is the incorporation of a large fraction of hydrogen as moderator in the fuel element itself. Since the fuel is a homogeneous alloy with a large portion of the moderator, the energy deposited by fission fragments is immediately manifested in increased moderator molecular mean velocity. In the core of $U\text{-}ZrH_{1.7}$ this increased molecular velocity is very effectively translated into an increased average thermal neutron velocity. The result is an essentially instantaneous shift in the neutron spectrum and a resulting shift in the balance among fissions, absorption, and leakage. A rise in the temperature of the hydride increases the fraction of hydrogen atoms in higher excited states and increases the probability that a thermal neutron in the fuel element will gain energy (hardening the thermal neutron spectrum, as shown in the attached figure), and escape to be captured in the water rather than in the fuel.

Many papers have been written on the TRIGA temperature coefficient, both from a theoretical and an experimental point of view. Technical Foundations of TRIGA[1] reviews the original experiments in measuring the moderating properties of zirconium hydride and confirming the TRIGA coefficient. Nelkin[2] reviews the methods of calculating the various contributions to the temperature coefficient and the discussion by West[3] summarizes experimental work and reactor physics calculations for TRIGA cores, and provides extensive references to the literature on this subject. A paper by Scalettar[4] reviews the fundamental TRIGA kinetic theory.

In summary it can be said that extensive theoretical work established that the TRIGA reactor has self-limiting properties that make it inherently safe. These safety characteristics were then proved by tests and experiments, and finally, have been demonstrated by more than 200 cumulative reactor years of operation.



Neutron flux spectrum in TRIGA for hydride temperatures of 20°C and 220°C

References:

1. "Technical Foundations of TRIGA," General Dynamics, General Atomic Division Report GA-471, August 27, 1958. 119 p.

2. Nelkin, M. S., and G. B. West, "Calculations of the Prompt Temperature Coefficient for TRIGA," private communication.

3. Dee, J. B., and G. B. West, "TRIGA Nuclear Analysis," General Dynamics, General Atomic Division Report GA-6025, February 1, 1965.

4. Scalettar, R., "Kinetics of TRIGA--Part I: Fundamentals," General Dynamics, General Atomic Division Report GA-2599, October 29, 1961.

7. Do you believe the following postulated events could occur in the AFRRI reactor and, if not, what are your bases for so believing:

(a) defects in the material integrity of the fuel elements;

(b) an uncontrolled power excursion in the reactor core;

(c) a loss-of-coolant accident;

(d) sabotage, aircraft collision or natural ("act of God") accident.

Answer to Question 7:

Answered by: Moore. Sholtis

A. Each section of the question is independently answered as follows:

(a) No. The TRIGA fuel elements at AFRRI have sufficient operational history (power) such that material defects, were they present, would have become apparent. Additionally, AFRRI fuel elements have less power history than identical fuel elements at other TRIGA reactors, which have maintained their integrity.

(b) No. An "uncontrolled" power excursion is not possible in the TRIGA core because the same negative temperature coefficient that controls a planned power transient would control an unplanned transient (see answer to question 6).

(c) Yes. Although remote, loss of coolant is considered as a possible accident and is therefore considered in the AFRRI SAR on file with the NRC and already provided to CNRS, Inc.

(d) (Sabotage) Yes. Although remote, sabotage is possible and therefore addressed in appropriate documents.

(Aircraft collision) Yes. However, the extreme unlikelihood of aircraft collision precludes major treatment, regardless, the results of such an

accident (e.g. loss of coolant, etc.) are treated in the AFRRI SAR.

(Natural "Act of God" Accident) Yes. However the low probability of such an occurrence precludes major treatment and, in any event, the consequences of such accidents are treated in the AFRRI SAR.

B. No references used.

C. See general statement.

D. See general statement.

E. See general statement.

8. Do you believe a multiple fuel element cladding failure could result from any or all of the events described in question 7 and, if not, what are your bases for so believing?

Answer to Question 8:

Answered by: Moore, Sholtis

A. a,b. No. These events cannot occur.

c. No. There are insufficient temperatures for clad failure to occur. See answer to question 9.

d. Yes. Although remote, multiple fuel element clad failures could occur and are referred to in the SAR.

B. No references used.

C. See general statement.

D. See general statement.

E. See general statement.

9. State the scientific and mathematical bases for your belief that, should an accident such as that described in your HSR as a "Loss of Shielding and Cooling Water" accident occur while the reactor core is in the pulse mode, air convection cooling would be sufficient to prevent cladding failures resulting in fission product releases in excess of 10 C.F.R. Part 20 limits.

Answer to Question 9:

Answered by: Moore, Sholtis

A. Since the water surrounding the core is necessary to initiate and sustain a chain reaction (20% of moderation), the loss of water would prevent the initiation of a pulse or terminate a pulse already in progress. In any condition whereby the stainless steel cladding temperature increases above 500°C, such as during a LOCA, the cladding ultimate strength (1000°C) would be decreased. To establish the ultimate strength in this case, one must assume, based on the second law of thermodynamics, that the cladding temperature can never exceed that of the fuel meat, since the fuel meat is the source of heat. An analysis of this condition indicates that the equilibrium hydrogen pressure produces a stress on the clad equal to its "ultimate strength" at approximately 950°C.

The current and proposed AFRRI Technical Specification limit for excess reactivity is $5.00. If we assume that AFRRI had this amount of excess reactivity (a physical impossibility at this time) and it was somehow inserted in a single step function, the resultant fuel temperature would be approximately 800°C. Even if we somehow postulate a LOCA precisely at the moment the pulse terminated and assume worst case adiabatic conditions, the resultant temperatures and pressures are still well below (by 150°C) the

rupture point of the stainless steel clad. Moreover, the temperature differential between the air and the cladding would result in a transfer of heat, thereby rapidly decreasing the temperature across the cladding and fuel.

B. 1. "The U-ZrH_X Alloy: Its Properties and Use in TRIGA Fuel," M. T. Simnad, Feb 1980, G. A. Project 4314 G. A. Report E117-833

2. Principles of Heat Transfer, Frank Kreith, 2nd edition July 1968, International Text Book Co., Scranton, Penn.

C. See general statement.

D. See general statement.

E. See general statement.

10. State the scientific and mathematical bases for your position that the following accidents could not occur in the AFRRI reactor:

(a) power excursion accident (PEA) resulting in multiple cladding failures at an elevated temperature with reduction in the thermalizing effect of hydrogen, followed by an explosive zirconium-steam interaction;

(b) a loss-of-coolant accident (LOCA) resulting in multiple cladding failures at an elevated temperature, followed by an explosive zirconium-air interaction.

Answer to Question 10a:

Answered by: Moore, Sholtis

A. A power excursion accident cannot happen at elevated temperatures with a reduction in the thermalizing effect of hydrogen, resulting in multiple cladding failures, followed by an explosive zirconium-steam interaction because:

1. AFRRI's TRIGA reactor is a thermal reactor; therefore, it requires thermal neutrons to sustain (or increase) a chain reaction. With a reduction in the thermalizing effect of hydrogen (which you presuppose), the neutron energy spectrum would become hardened (i.e., the neutrons will remain at epithermal energies). This hardening of the neutron spectrum is a negative reactivity effect for a thermal reactor. Therefore, this would reduce the fission rate and automatically terminate the power excursion more effectively than normal and shut the reactor down completely.

2. An explosive Zirconium-steam interaction cannot occur because the conditions necessary for such a reaction to occur are not present in the TRIGA core. The source of heat within a fuel element is the Uranium

Zirconium hydride fuel matrix. The $UZrH_x$ has been shown to have a benign response to water or steam via quench tests of fuel samples from temperatures as high as 1200°C (This temperature is well above the maximum temperature possible in the AFRRI-TRIGA).

B. 1. "TRIGA Low-Enriched Uranium Fuel Quench Test," General Atomic Project 4314, G.A. Report GA-A15384, July 1980.

2. See references cited under section B response to question #6.

Answer to Question 10b:

Answered by: Moore, Sholtis

A. A loss of coolant accident (LOCA) resulting in multiple cladding failures at an elevated temperature followed by an explosive zirconium-air interaction cannot occur in the AFRRI reactor because:

The basic constituents for a zirconium-air interaction are not present; namely, elemental zirconium is not present in the fuel meat and adequate temperatures for the interaction are not present.

The TRIGA fuel is $U\text{-}ZrH_x$ (i.e., uranium-zirconium hydride) which has been shown experimentally to be relatively chemically inert in air at temperatures up to 850°C. See reference 1 under B below. For the postulated LOCA situation, the fuel cladding temperatures would be in the range of 550°C to 700°C—insufficient for an air-zirconium hydride reaction to occur. This 550-700°C temperature range was determined analytically using conservative assumptions and only takes account of heat removal via air convection cooling. See reference 2 under B below.

B. 1. Simnad, M. T., et al, "Fuel Elements for Pulsed TRIGA Research Reactors," Nuclear Technology, Vol 28, January 1976, pp 31-56.

2. Final Safeguards Report, AFRRI-TRIGA Reactor, Chapter VI, Hazards Analyses for Loss of Coolant, March 1962.

C. See general statement.

D. See general statement.

E. See general statement.

11. For each of the seven components (A-G) and sub-components of emergency planning set forth in Attachment A, Contention 3, "Emergency Plan," of the Stipulation signed by AFRRI, NRC, and Intervenor on March 31, 1981, you have brought your emergency plan into conformity with the requirements of 10 C.F.R. Part 50, Appendix E. State the identity of persons, agencies, and organizations where identification of same is called for in the sub-component, and attach schematics, agency directives, correspondences, and cooperative agreements between yourself and other persons, agencies, and organizations,and any other documents that pertain to the emergency planning for the AFRRI facility.

Answer to Question 11:

Answered by: Smoker, Spence, Alt, Moore

A. The AFRRI Emergency Plan is not required to meet the full requirements of 10CFR50, Appendix E.* As a research reactor, we fall under the requirements of NRC Regulatory Guide 2.6. Our emergency plan was developed, in close coordination with the NRC staff, along the guidelines of Reg Guide 2.6, and we believe that we have fully met all requirements.

B. 1. 10CFR50, Appendix E, Section I, third paragraph and footnote 3.

2. USNRC Regulatory Guide 2.6, Emergency Planning for Research Reactors, January 1979.

Both are available in NRC reading room.

C. See general statement.

D. See general statement.

E. See general statement.

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*Under Appendix E, research reactors are treated under different guidelines than power reactors.

12. If you have not met the requirements of 10 C.F.R., Part 50, Appendix E, with respect to any of the components and sub-components listed in question 11, state why not, whether you plan to comply, and if so, when and how.

Answer to Question 12.

Answered by: Smoker, Spence, Alt, Moore

A. Since AFRRI has met necessary requirements, this question is not applicable.

B. Not applicable.

C. See general statement.

D. See general statement.

E. See general statement.

13. What emergency planning requirements which apply to the AFRRI facility, other than those set forth in 10 C.F.R. Part 50, Appendix E, have been proposed or adopted by the Nuclear Regulatory Commission, Federal Emergency Management Agency, Department of Energy, Environmental Protection Agency, Food and Drug Administration, Division of Radiological Control of The Maryland Department of Health, Montgomery County Civil Defense Office, and any other Federal, state, county, or municipal agency?

Answer to Question 13:

Answered by: Smoker, Alt, Moore

A. Emergency planning requirements applicable to the preparation of AFRRI's "Emergency Plan for TRIGA Reactor," for the purpose of USNRC license renewal, are predicated on the guidance set forth in USNRC letter dated 3 Apr 81 with accompanying "License Renewal Review Items," and on direct coordination with the USNRC during the preparation of the plan. Since the license renewal procedure is under the jurisdiction of the USNRC, any additional, proposed or adopted, requirements by any of the other agencies cited in the above question are not applicable. Additionally, to date no recommendations have been received by AFRRI from the aforementioned agencies. Should any additional requirements be identified as a result of USNRC review, AFRRI will take appropriate action.

B. References:

1. USNRC letter, dated 3 Apr 80, Subject: Facility Operating License No. R-84, with enclosure "License Renewal Review Items."

C. See general statement.

D. See general statement.

E. See general statement.

14. For each requirement referred to in question 13, describe the extent to which and how you have brought your emergency planning into compliance. State the identity you relied on for your statement in the HSR that it takes approximately 67 closely packed fuel elements to achieve criticality.

Answer to Question 14:

Answered by: Smoker, Alt, Moore, Sholtis

A. Based on the answer to question number 13, the first portion of this statement is not applicable. The second portion of this question, concerning the number of fuel elements to achieve criticality, is answered in our response to question number 3.

B. References:

1. USNRC letter, dated 3 Apr 80, Subject: Facility Operating License No. R-84, with enclosure "License Renewal Review Items."

2. RSD 5-8, "Reactor Core Loading and Unloading Procedures," AFRRI/SSRS, 27 Mar 81.

See general statement.

D. See general statement.

E. See general statement.

15. If you have not met the requirements referred to in question 5, state why not, whether you plan to comply, and if so, when and how.

Answer to Question 15:

Answered by: Smoker

A. We can find no requirements in question 5 or 15 and are unsure specifically to what requirements the question is referring to.

B. No references used.

C. See general statement.

D. See general statement.

E. See general statement.

16. Describe how you developed the classification system for emergencies (2.1-2.4) in your Emergency Plan submitted with your license renewal application.

Answer to Question 16:

Answered by: Smoker, Alt, Moore

A. The classification system for emergencies (2.1-2.4) in AFRRI's Emergency Plan submitted with its license renewal application is that system specifically elucidated by paragraphs 2.1 through 2.1.4 in Annex A of USNRC Regulatory Guide 2.6, dated January 1979.

B. References: USNRC Regulatory guide 2.6, dated January 1979.

C. Reference general statement.

D. Reference general statement.

E. Reference general statement.

17. Give an example of each class of emergency referred in your Emergency Plan (2.1-2.4), including a description of the chain of events leading to such an accident and the steps you would take to mitigate or terminate the emergency condition.

Answer to Question 17:

Answered by: Alt, Smoker, Moore

A. Answer:

(1) One example of a "Personnel Emergency" condition, as defined in the AFRRI Emergency Plan, is a possible scenario where an AFRRI staff member trips while carrying an activated isotope, fractures a leg, and spills the radionuclide on himself. Mitigation/termination of this condition would involve providing medical and decontamination assistance to the injured staff member and decontaminating the incident site.

(2) One example of an "Emergency Alert" condition, as defined in the AFRRI Emergency Plan, is a severe storm warning. Mitigation/termination of this condition would involve securing the reactor, experimental facilities, and related equipment.

(3) One example of a "Reactor Emergency" condition, as defined in the AFRRI Emergency Plan, is a possible break in the reactor primary water system. Mitigation/termination of this condition would involve securing the reactor, experimental facilities and related equipment, and performing actions to restore and maintain the water level in the reactor tank. These actions could include valving off selected sections of the reactor primary water system, use of the reactor tank repair kit, or employment of the reactor tank auxiliary fill lines.

(4) The "Facility Emergency" condition is, as explained in the AFRRI Emergency Plan, not deemed possible for the AFRRI TRIGA reactor.

B. References:

(1) AFRRI Emergency Plan, paragraphs 2.0 through 2.4

(2) USNRC Regulatory Guide 2.6, dated January 1979

(3) AFRRI Safety Analysis Report, Chapter 6, 12 May 1981.

C. See general statement.

D. See general statement.

E. See general statement.

18. For each class of emergency (2.1-2.4) state whether during AFRRI's operating history any such emergencies have occurred and describe each such emergency, including the precipitating event, individuals exposed to radiation, injuries sustained, mitigating steps taken, resolution of the emergency situation, citations and/or notices of violations from the NRC and any other Federal, state, county, or municipal agency, and steps you have taken to preclude or reduce the probability of recurrence of such emergencies.

Answer to Question 18:

Answered by: Moore, Smoker

A. 1980 was the first year in which an Emergency Plan was required to be submitted by AFRRI to the USNRC. Prior to that date, such classification of emergencies did not exist; therefore, this question can only be answered from this date. That answer is, no such emergencies have occurred. In addition, to the best of the current Reactor Staff's knowledge, there have never been any incidents associated with the reactor facility that would have required invoking the AFRRI TRIGA Reactor Emergency Plan.

B. No references used.

C. See general statement.

D. See general statement.

E. See general statement.

19. In 4.2.1 through 4.3.3 of the Emergency Plan, describe the means by which you plan to notify AFRRI Security Officer, supervisors of the laboratories, AFRRI Director, Head of Radiation Sources Division, and Radiological Safety Department Head, AFRRI staff, NNMC personnel, weather officials, police, and civil defense personnel in the event of an emergency, including a description of the back-up means of notifying the said individuals, officials, and personnel if the planned means is not feasible and the back-up persons who will be notified if those designated are not available.

Answer to Question 19:

Answered by: Smoker, Alt, Moore, Sholtis

A. (1) The primary means to notify the AFRRI Security Officer, supervisors of the laboratories, AFRRI Director, Head of Radiation Sources Division, Radiological Safety Department Head, AFRRI Staff, NNMC personnel, weather officials, police, and civil defense personnel is by telephone or public address, where appropriate.

(2) The back-up means to notify these individuals is by use of automobile messenger or by message relayed by the local police department.

(3) The back-up personnel who will be notified if those designated are not available are as follows:

a. AFRRI Security Officer - AFRRI Security NCO

b. Supervisors of the laboratories - Senior laboratory staff member as listed on the AFRRI Emergency Notification Roster

c. AFRRI Director - AFRRI Deputy Director

d. Head of Radiation Sources Division - Reactor PIC, Reactor CSO, or Chairman of Scientific Support Department

e. Radiological Safety Department Head - Head, Radiation Health Physics Division

f. AFRRI staff, NNMC personnel, weather officials, police, and civil defense personnel - back-up personnel on duty as determined by individual organizations.

B. References:

(1) AFRRI Emergency Plan, Chapter 4.

(2) AFRRI Emergency Notification Roster

C. Reference general statement.

D. Reference general statement.

E. Reference general statement.

20. Describe the accident that occurred in your cobalt facility between April 22 and May 16, 1981, including a statement of the class of emergency it began as and escalated to, the precipitating event(s), the mitigating steps taken, the extent to which the emergency plan operated as planned, who the decision-makers were (including the person(s) who acted in the Director's absence), the individuals who were exposed to radiation as a result of the accident, their levels of exposure and whether the same exceeded Federal limits, the concentration levels of radiation in the cobalt storage room, AFRRI building, and outside the buidling (in restricted and non-restricted areas), and whether these exceeded Federal levels, final resolution of the accident, steps you have taken to preclude its recurrence, citations and notices of violation from the NRC,and correspondences between AFRRI and other agencies pertaining to the accident.

Answer to Question 20:

Answered by: Smoker

A. This question is irrelevant to these proceedings.

B. Not applicable.

C. Not applicable.

D. Not applicable.

E. Not applicable.

21. Describe how this accident affected the operation of your reactor.

Answer to Question 21:

Answered by: Smoker, Sholes, Moore, Alt

A. There was no effect on reactor operations.

B. No references used.

C. See general statement.

D. See general statement.

E. See general statement.

22. Describe how, in a "worst-case scenario" of the cobalt accident, the operation of your reactor would have been affected.

Answer to Question 22:

Answered by: Smoker, Sholtis, Moore, Alt

A. There would be no effect on reactor operations from a "worst-case" cobalt accident.

B. No references used.

C. See general statement.

D. See general statement.

E. See general statement.

23. Describe the evacuation and other emergency plans, both within the AFRRI facility and in conjunction with other agencies and the public, that were put into a state of readiness and/or were actually carried out in the course of the cobalt accident.

Answer to Question 23:

Answered by: Smoker

A. This question is irrelevant to these proceedings.

B. Not applicable.

C. See general statement.

D. See general statement

E. See general statement.

24. Describe the instructions AFRRI personnel were given during the cobalt accident regarding protective and mitigative measures they should take, evacuation, and the possibility that they could not return to work if the emergency situation escalated or continued unabated.

Answer to Question 24:

Answered by: Smoker

A. This question is irrelevant to these proceedings.

B. Not applicable.

C. See general statement.

D. See general statement.

E. See general statement.

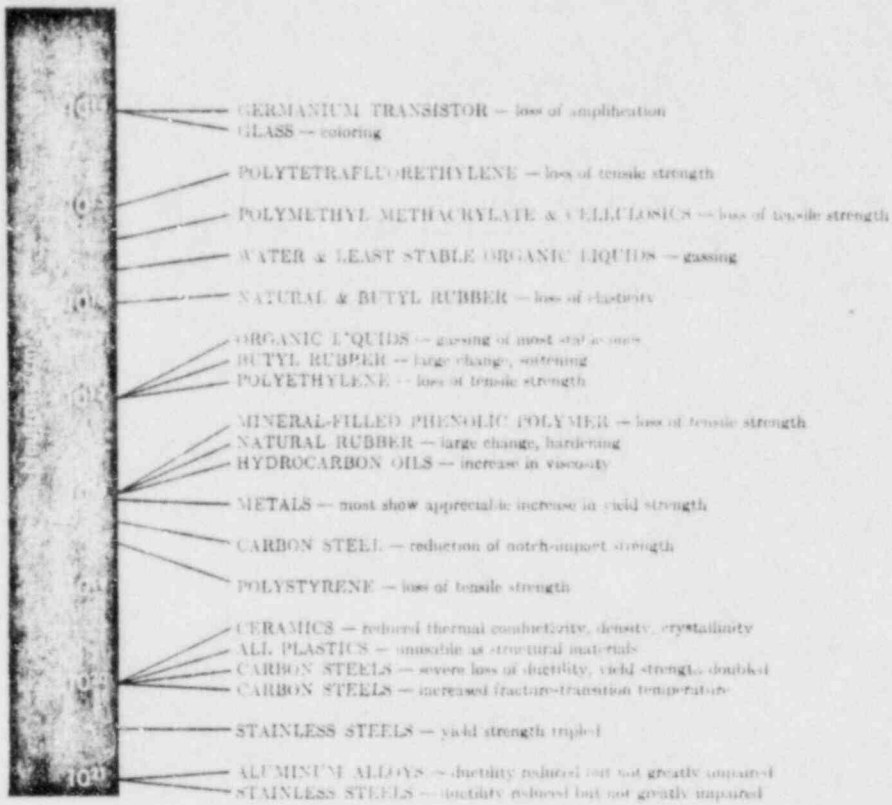
25. What scientific and mathematical bases, if any, do you have and what empirical data and evidence can you cite, if any, to support the position that the effects of aging including increasing brittleness and metal fatigue on the reactor parts will not reduce the margin of operating safety and increase the risk of malfunctions and accidents during the AFRRI reactor's proposed third and fourth decades of operation?

Answer to Question 25:

Answered by: Sholtis, Munno

A. First, it should be emphasized that a reduction in a safety margin pertaining to mechanical properties does not imply that the risk of malfunctions and accidents associated with structural failure is increased unless the safety margin completely disappears and the environment or conditions for reaching or exceeding the mechanical failure limits (i.e., properties) of the structure are present.

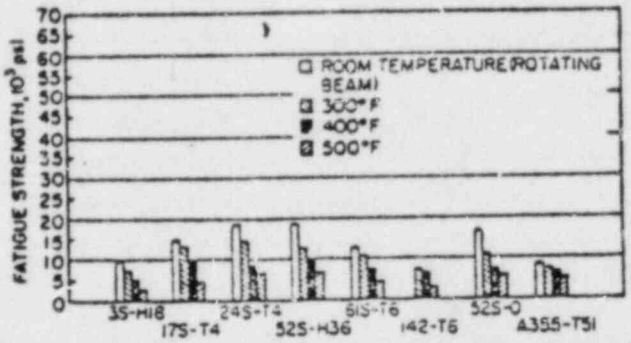
With respect to radiation embrittlement, almost every source of information dealing with radiation damage to materials that was reviewed in connection with this issue, including several basic texts, indicates that below a fast neutron fluence of 1.0×10^{20} n/cm^2 the mechanical properties of stainless steels and aluminum (the materials that provide structural support within the AFRRI TRIGA reactor) are essentially unaffected. In fact, at a fast neutron fluence level of 1.0×10^{21} n/cm^2, both aluminum and stainless steel have ductilities that are said to be "reduced but not greatly impaired." Moreover, the yield strength of stainless steel actually increases at an NVT of approximately 3.0×10^{20} fast n/cm^2. See figure immediately below taken from: Introduction to Nuclear Engineering, R. L. Murray, Prentice-Hall, Inc., Englewood Cliffs, second edition, 1961, p. 178.



Variation of damage with dosage.

Using a conservative value for fast neutron fluence, at which the mechanical strength properties of both aluminum and stainless steel are assumed to begin degrading, of 1.0×10^{20} n/cm^2 and using an average annual burnup of 25.8 MW-hours per year determined from AFRRI's actual operational records, and finally using a conservative value for the AFRRI reactor fast flux of 8.0×10^{12} n/cm^2/sec while operating at 1.0 MW(t), indicates that the AFRRI reactor facility structures exposed to fast neutron irradiation would not accumulate 1.0×10^{20}, fast neutrons/cm^2 until the year 2097 AD, i.e., not for another 116 years, based on initial criticality of the AFRRI reactor in 1962.

With respect to the metal fatigue, again almost every reference dealing with materials, metallurgy, and fracture mechanics that was reviewed in connection with this issue, again including several basic texts, indicates that the fatigue strength of stainless steel is well above that for aluminum and that degradation in the fatigue strength of aluminum is essentially unaffected for 100°C thermal cycling. In addition, for 150°C thermal cycling, the fatigue strength of aluminum is only reduced by about 33% after 5×10^8 such thermal cycles. See figure below taken from: Nuclear Engineering Handbook, H. Etherington (ED), McGraw-Hill, New York, NY, first edition, 1958, p 10-50.



Comparison of the fatigue strengths of various aluminum alloys at 5×10^8 cycles as affected by temperature. Tested after stabilization periods of 0.5 to 2 hr at testing temperature. (*Constructed from aluminum producers' data by C. M. Craighead et al., Battelle Memorial Institute. Aug. 1, 1952.*)

The structural support components of the AFRRI reactor core can never exceed the cladding surface temperature which is typically 100°C for full power steady state as well as pulse operations.

Using 100°C as a conservative baseline thermal cycling range for the onset of degradation of fatigue strength in aluminum after 5×10^8 cycles, as per the figure illustrated above, and using 1000 such cycles per year as an average based on AFRRI's actual operating records, indicates that the fatigue strength of the aluminum structural support components of the core would be essentially unaffected for 500,000 years. (Note: The aluminum and stainless steel structural support components experience, as a maximum, ~100°C temperature variations during reactor operations.) Moreover, stainless steel is

much more durable than aluminum, and requires thermal cycling over a much greater temperature range (~400°C) before its fatigue strength begins to degrade. As a result, the useful life of stainless steel structures in the core would also not be limited by fatigue.

B. 1. Etherington, H. (ED), Nuclear Engineering Handbook, First Edition, McGraw-Hill, New York, NY, 1958.

2. Murray, R. L, Introduction to Nuclear Engineering, Second Edition, Prentice-Hall, Englewood Cliffs, NJ, 1961.

3. Tipton, C. R. (ED), Reactor Handbook 1, Materials, Interscience Publishers, New York, NY, 1960.

4. Hausner, H. H. and Roboff, S. B., Materials for Nuclear Power Reactors, Reinhold Publishing Corp., New York, NY, 1955.

5. Kopelman, B. (ED), Materials for Nuclear Reactors, McGraw-Hill, New York, NY, 1959.

6. Bonilla, C. F. (ED), Nuclear Engineering, McGraw-Hill, New York, NY, 1957.

7. Brooks, H., "Nuclear Radiation Effects in Solids." Annual Review of Nuclear Science, 6, 1956.

8. Vineyard, G. H, et al, "The Effects of Irradiation," Chapter 8, Progress in Nuclear Energy, Series V, Metallurgy and Fuels, Pergamon Press, London, 1956.

9. Lyman, T. (ED), Metals Handbook, American Society for Metals, 1948 ed; 1954 supplement.

10. Mantell, C. L. (ED), Engineering Materials Handbook, McGraw-Hill, New York, NY, 1958.

11. Billington, D. S., "Radiation Damage in Reactor Materials," Proceedings, Int'l. Conference on Peaceful Uses of AtomicEnergy, August 1955.

12. Weinstein, R. (ED), Nuclear Engineering Fundamentals, Book IV, Nuclear Materials, McGraw-Hill,New York, NY, 1964.

13. Dienes, G. J,and Vineyard, G. H., Radiation Effects in Solids, Volume II, Interscience Publishers, New York, NY, 1957.

14. Dienes, G. J., "A Theoretical Estimate of the Effect of Radiation on the Elastic Constants of Simple Metals," Phys. Rev., 86, 228, 1952.

15. Fraser, A. S., et al, "High-Temperature Embrittlement of Stainless Steel Irradiated in Fast Fluxes," Nature, Vol. 211, 1966, pp. 291-292.

16. Rosenbaum, H. S., "Microstructures of Irradiated Materials," General Electric, NEDO-12356, 1973.

17. Claudson, T. T., et al, "The Effects of Fast Flux Irradiation on the Mechanical Properties and Dimensional Stability of Steel," Nuclear Applications and Technology, Vol 9, July 1970, pp 10-23.

18. Conway, C., et al, "Fatigue and Tensile Behavior of Irradiated and Unirradiated SS 304 and SS 316," Nuclear Applications and Technology, 1970.

C. See general statement.

D. See general statement.

E. See general statement.

26. Do you believe your environmental monitoring system (i.e., your equipment, methods, and reporting system for measuring releases into the Montgomery County sanitary sewerage system and at your perimeter and offsite monitoring stations) is adequate to determine radiation dose to the public due to inhalation or ingestion?

Answer to Question 26:

Answered by: Loesch, Arras

A. Although no AFRRI USNRC licenses require any environmental monitoring, the present AFRRI environmental surveillance program is both more comprehensive and more restrictive than current regulatory requirements and is adequate to determine radiation dose to the general public.

B. No references used.

C. See general statement.

D. See general statement.

E. See general statement.

27. If your answer to question 26 is "No," explain why not and describe the steps you have taken to make the system adequate.

Answer to Question 27:

Answered by: Loesch

A. No answer required.

B. No references used.

C. See general statement.

D. See general statement.

E. See general statement.

28. If your answer to question 26 is "Yes," explain why the following inadequacies in your system, alleged by the Intervenor and cited by the NRC, do not now nor in the future, will in fact detract from your system's ability to fully and accurately determine radiation doses to the public.

(a) film dosimetry detects only external gamma radiation.

(b) the particulate radioactivity monitor for airborne effluents (i.e., a pancake-probe C-M counter) is not isokinetic, and therefore cannot be used for meaningful evaluations. Applicant's only other stack effluent monitoring system, the radioactive gas monitor, is likewise not reliable for particulate sampling. (See Environmental Release Report issued 12/14/71, covering period 1/1/70-9/30/71, and Inspection Report No. 50-170/77-01-03.)

(c) The Violation Notice of Gross Beta Effluent Analysis, based on an NRC Inspection conducted January 12-14, 1977, cited Applicant for calculational omissions, methods for preparing and analyzing samples, and instrumentation used. The gross beta measurements were made without the use of a beta self-absorption correction in the presence of significant amounts of suspended solid material. (See NRC Inspection Reports No. 50-170/77-01-02 and 50-170/77-01-03.)

(d) The "concentric cylinder set model" used by Applicant to derive its dose assessments to the environment, and from which it concludes its effluents are within regulatory limits, is an unrealistic model.

Answer to Question 28:

Answered by: (a) Arras, Loesch; (b) Webber; (c) Loesch; (d) Arras

A. (a) This is an error in fact. The beta capability of our environmental monitors has never been questioned by the NRC. The monitoring of

external gamma emitters is more than adequate to determine the dose to unrestricted areas due to air activation products produced by reactor operations. Both the NRC and American National Standards Institute have set standards for the application of TLD's to environmental monitoring. They specify response criteria for x and gamma photons only.

(b) The stack particulate monitor (pancake probe G-M detector), although not required by our reactor license, was installed by AFRRI and has been maintained and utilized for more than ten years. The flow rate in the stack effluent duct at the point of sampling is typically 12500 cfm which results in a linear flow velocity of 1770 ft/min in the three foot diameter duct. The flow rate in the probe is nominally 8.5 cfm through the 15/16" diameter sampling tube resulting in a linear flow velocity of 1770 ft/min. This results in reasonable isokinetic sampling. It should be noted that the air released from AFRRI first flows through an absolute filter, tested and maintained at >99.98% efficiency for particles greater than .3 microns. Over the past ten years no indication of long-lived fission or activation products from the reactor have been detected on the stack particulate monitor filter.

(c) This question is misleading. The cited NRC violation stated that the measurements were inadequate in that the gross beta measurements were made without the use of a beta self-absorption correction factor. Since this inspection, a beta self-absorption correction factor has been applied to all analyses of liquid waste samples. Even with the correction factor, all releases were well within all regulatory requirements. At no time was there a significant possibility of exceeding regulatory limits, since the standard procedures require specific radionuclide analysis if concentrations are greater

than 10% of regulatory limits. No items of non-compliance were found in either the methods for preparing and analyzing samples or the instrumentation used.

(d) This question is an error in fact. The "Concentric Cylinder Set Model" only supplements environmental TLD's; it is not in itself used to determine compliance with regulatory limits. No responsible organization, including the NRC, has found the model to be unrealistic.

B. References.

1. NRC Regulatory Guide 4.13
2. ANSI Standard N545-1975
3. NRC Inspection Report No. 50-170/77-01-02
4. NRC Inspection Report No. 50-170/77-01-03

C. See general statement.

D. See general statement.

E. See general statement.

29. Describe the system you have used and use to prepare and analyze quarterly environmental samples of water, soil, and vegetation in your "Environmental Sampling and Analysis" program referred to in your Environmental Impact Appraisal Data Report.

Answer to Question 29:

Answered by: Loesch, Webber

A. The quarterly environmental samples are prepared and analyzed as follows:

Surface Water: A one liter sample is obtained downstream from the radioactive waste tank storage facility and filtered to remove suspended solids. The liquid is then evaporated and both the filter and planchett are counted for gross alpha and beta. Specific samples can be analyzed further on a multichannel analyzer if the previous gross count indicates activity above normal background.

Soil: Surface soil, within top six inches, is collected and dried for approximately three days. One gram is placed in a stainless steel planchett and counted on a proportional counter. In addition, a 3.5 liter marinelli flask is filled and analyzed on a gamma spectrometer.

Vegetation: A mixture of vegetation is collected from various areas around AFRRI. The vegetation is first washed. The vegetation is then analyzed in a 3.5 liter marinelli flask with a multichannel analyzer. In addition, the rinse water is evaporated and any matter contained within is counted on a proportional counter.

B. Health Physics Procedure 2-2.

C. See general statement.

D. See general statement.

E. See general statement.

30. What raw data have you collected from 1970 to the present in your "Environmental Sampling and Analysis" program?

Answer to Question 30:

Answered by: Loesch

A. Voluminous data have been collected, since the inception of the program, in the following areas:

1. Vegetation analysis
2. Soil analysis
3. Surface water analysis
4. Quarterly TLD reports from supplier (65 stations)
5. Stack gas effluent monitor outputs
6. Liquid radioeffluent analysis

In addition, a good deal of information has been collected about the actual weather characteristics in the local area.

B. No references used.

C. See general statement.

D. See general statement.

E. See general statement.

31. What steps have you taken to prevent the recurrence of discharge on January 10-12, 1979 of Argon-41 and other radionuclides at ground level outside the reactor building through a leak in the ventilation exhaust stack drain pipe, referred to in NRC Inspection Report No. 50-170/79-01?

Answer to Question 31:

Answered by: Moore

A. Promptly upon discovery of the dry water trap, the drain line was removed and the exit from the stack capped and sealed.

B. No references used.

C. See general statement.

D. See general statement.

E. See general statement.

32. Has the incident referred to in question 30 or similar incident occurred on any other occasion at AFRRI?

Answer to Question 32:

Answered by: Moore, Smoker, Sholtis, Arras

A. It is presumed that your reference in the statement of this question is to question #31 and not #30. The answer to this question, therefore, is "no."

B. No references used.

C. See general statement.

D. See general statement.

E. See general statement.

33. In view of your statement in your Environmental Impact Appraisal Data Report, p. 5, that the efficiency of the reactor's cooling tower "is determined by the temperature and humidity of the outside ambient air," what are the operating parameters of said tower's efficiency?

Answer to Question 33:

Answered by: Smoker, Spence, Moore

A. No tests have been conducted to determine the specific efficiencies for various ambient temperature and humidity conditions. However, it is known that as the temperature differential between the outside air and secondary cooling water increases, the tower efficiency increases. Likewise, as the outside relative humidity decreases, the tower efficiency increases.

B. No references used.

C. See general statement.

D. See general statement.

E. See general statement.

34. State the names and addresses of all suppliers of the fuel for your reactor.

Answer to Question 34:

Answered by: Moore

A. 1. General Atomics Corp.

Box 81608

San Diego, CA 92138

2. Harry Diamond Labs - U.S. Army

Powder Mill Road

Adelphi, MD

B. AFRRI Reactor staff members' personal address index.

C. See general statement.

D. See general statement.

E. See general statement.

35. How and from where is said fuel transported to your facility?

Answer to Question 35:

Answered by: Moore

A. All but three of the fuel elements were delivered in 1964-5 by General Atomics in San Diego, CA; how they were physically delivered is not known by the current AFRRI staff.

The remaining fuel elements were transported from Forest Glenn, MD to AFRRI by truck in early 1978.

B. No references used.

C. See general statement.

D. See general statement.

E. See general statement.

36. Describe your procedures for interim on-site and off-site disposal of the reactor's spent fuel elements.

Answer to Question 36:

Answered by: Moore, Smoker, Sholtis

A. Since the reactor fuel currently available and in use at AFRRI (since 1964) is expected to last through the requested licensing period (20 years), there are presently no plans for interim disposal of any spent fuel elements, either on- or off-site.

B. No references used.

C. See general statement.

D. See general statement.

E. See general statement.

37. Describe any occasions on which you have incinerated or buried nuclear waste, or directed another to do the same, including the dates and locations of said activities and the amounts of waste involved.

Answer to Question 37:

Answered by: Loesch, Arras

A. Up until 1970, some of AFRRI's low level irradiated biological waste and various burnable wastes were transferred to the National Naval Medical Center for incineration under their license. During the period 1963 to 1970, between 160 and 405 (average was 274) 3-cu. ft. boxes per year were transferred to NNMC. At no time did AFRRI hold a license for or incinerate any radioactive waste. In addition to these wastes, liquid scintillation vials and non-burnable waste materials were shipped to Edgewood Arsenal who handled shipment transfer to a commercial burial site. Since 1970, no materials have been transferred for incineration to any location. Currently, all waste is containerized in 55-gal drums and shipped by Southwest Nuclear to a commercial low-level waste burial site at Richland, Washington. Our waste contract is coordinted through the U.S. Army Armament Materiel Readiness Command, Rock Island, Illinois. Radioactive waste shipments are made approximately three times per year and average 20 to 30 drums per shipment.

B. References.

1. AFRRI Radioactive Material Shipment Records, AFRRI Form 116, for period CY70.

2. Contract with Rock Island, Ill.

C. See general statement.

D. See general statement.

E. See general statement.

38. Is there now any nuclear waste buried on AFRRI or NNMC grounds?

Answer to Question 38:

Answered by: Arras, Loesch

A. At no time has any radiological waste of any kind been buried on AFRRI or NNMC grounds, nor are there any plans to do so.

B. No references used.

C. See general statement.

D. See general Statement.

E. See general Statement.

39. Is it still your practice to routinely discharge radioactive effluents into the Montgomery County sewerage system?

Answer to Question 39.

Answered by: Loesch

A. All liquid wastes are accumulated by AFRRI in its liquid waste storage tank facility and routinely discharged to the Montgomery County sewerage system. Prior to discharging any liquids from the waste tank facility, the contents of the appropriate tank are sampled, analyzed, and the results reviewed by a member of the professional Health Physics staff to insure compliance with all current regulatory requirements. It should be noted that although the NRC specifies an accumulated limit of 1 curie/yr, AFRRI has set its own internal administrative limit of 100 mCi/yr (10% of federal limits). The levels released are extremely low, even before considering dilution factors; AFRRI releases typically are less than 10% of the limits specified in 10 CFR 20.303. Note: No liquid radioactive wastes are generated by routine reactor operations.

B. References.

1. 10 CFR 20.303

2. AFRRI Health Physics Procedure 6-4, "Waste Tank Facility"

C. See general statement.

D. See general statement.

E. See general statement.

40. What radioisotopes and in what concentrations and absolute amounts have you discharged into said sewerage system every year from 1961 to the present?

Answer to Question 40:

Answered by: Loesch, Arras

A. Attached is a summary of AFRRI's radioeffluent releases from 1963 to present.

It should be noted that all releases to the present have been less than 10% of regulatory limits and typically less than 1%. Also, the vast majority of the radionuclides present in AFRRI's waste liquid are not generated by reactor operations, but by biomedical research performed under other licenses.

B. References.

1. AFRRI Quarterly Radioeffluent Summary Reports

2. AFRRI Waste Tank Log

3. DF, dtd 16 Aug 79, "Radioeffluent Summary: 1974 through mid-1979"

C. See general statement.

D. See general statement.

E. See general statement.

Attach to question #10 response
(2 pages)

ATTACHMENT

(Page 1)

Year	Gross α,β,γ	Fraction of Regulatory Limit
1963 µc total	243.3	<.0003
µc/ml	4.2E-9	
1964 µc total	44.7	<.00005
µc/ml	1.7E-9	
1965 µc total	65.0	<.00007
µc/ml	1.8E-9	
1966 µc total	2870	<.003
µc/ml	6.2E-8	
1967 µc total	35495	<.04
µc/ml	3.9E-7	
1968 µc total	598	<.0006
µc/ml	9.9E-9	
1969 µc total	602	<.0007
µc/ml	8.1E-9	
1970 µc total	1128	<.002
µc/ml	7.0E-9	
1971 µc total	394.8	<.0004
µc/ml	2.3E-9	
1972 µc total	92000	<.1
µc/ml	3.9E-7	
1973 µc total	3400	<.004
µc/ml	1.2E-8	

ATTACHMENT

(Page 2)

Year	Gross α	Gross β	Gross γ	Fraction of Regulatory Limit
1974 µc total	2.13	1435.6	2669	<.005
µc/ml	3.6E-12	2.4E-9	4.5E-9	
1975 µc total	0.81	147.9	956	<.002
µc/ml	1.4E-12	2.5E-10	1.6E-10	
1976 µc total	1.55	343.5	2770	<.004
µc/ml	2.6E-12	5.7E-10	4.5E-9	
1977 µc total	1.17	145.0	502	<.0007
µc/ml	2.0E-12	2.4E-10	8.4E-10	
1978 µc total	1.11	595.4	3188	<.004
µc/ml	1.9E-12	9.9E-10	5.3E-9	
1979 µc total	0.66	828.9	2846	<.004
µc/ml	4.0E-12	5.1E-9	1.7E-8	
1980 µc total	3.3	364.3	2113	<.003
µc/ml	2.8E-11	3.1E-9	1.8E-8	
* 1981 µc total	21.4	376.9	2084.1	<.003
µc/ml	2.1E-10	3.8E-9	2.1E-8	

NOTE: Gross α, β, and γ are totals for unidentified mixtures of radionuclides.
µc/ml: Average (over the year) release point concentrations.
Regulatory limit: Based on NRC license limit of 1 Ci/year.
*Year to date.

41. What measures have you taken to prevent the recurrence of the following security and management violations that have occurred at the AFRRI facility:

(a) Eighteen activations of the facility alarm system during a 34-day period, caused by personnel leaving after normal duty hours from unauthorized exits. Auditors were told by AFRRI security personnel and other AFRRI officials that investigations were not made of the activations and that not enough security people were on duty to investigate each time the alarm went off;

(b) unauthorized people entering the facility by following employees in who used their magnetic cards to unlock the door;

(c) failure to escort visitors attending weekly seminars and provide them with dosimeters;

(d) failure of employees entering and exiting the building after hours to sign a log showing their time of arrival and departure;

(e) violations of Applicant's accounting and dispensing procedures for controlled substances such as narcotics.

Answer to Question 41:

Answered by: Smoker

A. None of the cited instances of alleged violations involved the Reactor Controlled Access Area (CAA), which has separate controlled access functions from the rest of the AFRRI complex. At no time has the physical security of the AFRRI reactor controlled access area been questioned or cited as being inadequate by USNRC or any other applicable agency. Since the cited alleged violations have no impact on the integrity of the reactor CAA, your

questions are irrelevant to this proceeding. Nevertheless, the following is provided for your information. Both the AFRRI Complex Physical Security Plan and the AFRRI Reactor Physical Security Plan, as well as associated procedures, have been revised and implemented. Logically, both of these plans are for "Official Use Only" and their contents are not for public disclosure. The AFRRI staff has been made aware of the correct procedures dealing with these plans and specific AFRRI staff personnel have the responsibility to insure compliance of the same. Additionally, part (e) of this question has no bearing on security at all and is totally irrelevant to this proceeding.

B. No references used.

C. See general statement.

D. See general statement.

E. See general statement.

42. State your reasons for believing that the reactor and related operations in the AFRRI facility are or are not susceptible to sabotage and/or terrorism.

Answer to Question 42.

Answered by: Sholtis, Smoker

A. On 3 October 1980, AFRRI submitted its Physical Security Plan to the USNRC. This Physical Security Plan was reviewed and approved as written by USNRC as meeting all of the requirements under 10CFR 73.67 for the protection of special nuclear material of low strategic significance. The AFRRI Physical Security Plan was fully implemented on 10 March 1981.

B. 1. Letter from Mr. James R. Miller/USNRC to Captain Paul Tyler/AFRRI, dated 10 February 1981, Subject: AFRRI Physical Security Plan.

C. See general statement.

D. See general statement.

E. See general statement.

43. List the names and locations of all other nuclear research and testing reactors in the United States where experiments such as or similar to those performed at AFRRI are or could be carried out.

44. State your reasons for believing that all or some of the research and experiments performed at AFRRI could or could not be carried out at other reactors such as those referred to in question 43.

Answer to Questions 43 and 44:

Interrogatories 43 and 44 appear to be related to the issues encompassed by Intervenor's contention entitled "Siting." Presumably, Intervenor seeks evidence upon which to base an argument that the geographic location of the TRIGA reactor could (and therefore should) be changed. That contention was rejected by the Board in its August 31, 1981 Memorandum and Order. Alternatively, these Interrogatories were propounded to elicit information related to Intervenor's contentions entitled "NEPA I" and "NEPA II." If so, since those contentions are more properly within the province of the NRC Staff, these Interrogatories should be directed to the Staff. Regardless of who should provide the answers, they are premature since they address matters which are not yet in issue (and indeed may never become issues if the NRC Staff prepares the appropriate environmental documentation in compliance with NEPA). Whether based on "Siting" or "NEPA," these Interrogatories are not relevant at this stage of the proceeding. Licensee therefore objects to them.

45. For each of the years 1975 to the present, state what percentage of AFRRI staff time, reactor operating time, and annual operating budget has been spent on research and experiments unrelated to AFRRI's chartered mission, as stated below, including but not limited to ballistics and forensic testing for the FBI and materials testing for private industry.

DOD Directive No. 5105.33, May 11, 1972:

II. Mission

The mission of AFRRI shall be to conduct scientific research in the field of radiobiology and related matters that are essential to the medical support of the Department of Defense.

IV. Functions

...AFRRI shall

A. Operate facilities for conducting research on the biological effects of ionizing radiation and disseminate the results.

Answer to Question 45:

Licensee objects to this Interrogatory because the particular projects that have been accomplished at AFRRI over the last six years are not relevant to this proceeding. The information that would be produced in response to this interrogatory would have evidentiary value in some sort of intra-governmental budget process but has no application to the present proceeding.

DAVID C. RICKARD
Counsel for Licensee

AFFIDAVIT

Ronald R. Smoker, being duly sworn according to law, deposes and says that he is the Chief, Radiation Sources Division, Armed Forces Radiobiology Research Institute, and as such is responsible for the operation of AFRRI's TRIGA reactor and that he supervised the preparation of the answers to these Interrogatories and that those answers are true and correct to the best of his knowledge, information and belief.

Ronald R. Smoker

Ronald R. Smoker

State of Virginia) ss:

County of Fairfax)

Sworn to and subscribed before me this 30th day of October, 1981.

Norma N. Saville

NOTARY PUBLIC

NORMA N. SAVILLE

My commission expires Nov. 12, 1984.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE (TRIGA-Type Research Reactor)	Docket No. 50-170 (Renewal of Facility License No. R-84)

CERTIFICATE OF SERVICE OF DUPLICATE SIGNED
COPIES OF 30 OCTOBER 1981 FILING

I hereby certify that true and correct copies of the foregoing "LICENSEE'S ANSWERS TO INTERVENOR'S INTERROGATORIES" were mailed this 30th day of October, 1981, by United States Mail, First Class, to the following:

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