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)	)	

AFFIDAVIT OF ROBERT E. CARTER

I, Robert E. Carter, being duly sworn depose and state:

1. I am currently the Project Manager assigned to Armed Forces Radiobiology Research Institute ("AFRRI"), TRIGA-type research reactor, Docket No. 50-170 in the Office of Nuclear Reactor Regulation, Division of Licensing of the Nuclear Regulatory Commission.

2. The purpose of this affidavit is to address the issues raised by Citizens for Nuclear Reactor Safety ("CNRS") in its contentions as admitted and numbered by the Licensing Board in the Board's Memorandum and Orders of August 31, 1981 and January 28, 1983.

Contention 1 (Accidents I)

The analysis of the "Fuel Element Clad Failure Accident", one of the two designs basis accidents (DBAs) within Applicant's Hazard Summary Report (HSR) is faulty in that:

The analysis of the "Fuel Element Clad Failure Accident" erroneously assumes that cladding failure during a pulse operation or inadvertent transient would occur at a peak fuel element temperature of less than 100°C.

Petitioner contends that such cladding failure would be much more likely to occur at elevated fuel temperatures (in excess of 400°C), resulting in far greater gap activity and fission product releases than the HSR postulates.

3. The contention is in error in that it asserts that the Licensee's Hazard Summary Report (actually Safety Analysis Report (SAR) (June 1981)) assumes that a cladding failure during pulse operation or inadvertent transient would occur at a peak fuel element temperature of less than 100°C.

4. The SAR makes no such assumption, either explicitly or by implication. Instead, for the purposes of computation, the SAR assumes that ". . . the theoretical limit of 0.1 percent gap activity for fission product noble gases and iodines, as stated in reference 2, will be used in the consequence analysis for the Design Basis Accidents (Section 6.3.4)" SAR, Section 6.3.2.2, page 6-12, 13 and Section 6.3.4.2, pages 6-17 & 6-18.

5. Figure 5-1 of reference 2 indicates that a theoretical maximum release of 0.1% (10^{-3}) of the total inventory of gaseous fission products in a standard fuel rod corresponds to an infinite irradiation time at more than 600°C.

6. Therefore, the Licensee's assumption of 0.1% release following clad failure is related to the long-term history of operation, and the actual temperature of the cladding at the instant of assumed failure is irrelevant in determining this release.

7. This contention also states that ". . . such cladding failure would be much more likely to occur at elevated fuel temperatures (in excess of 400°C). . . ".

8. Contrary to this assertion, the Licensee has assumed, for the purposes of computation, that the cladding of a fuel element has certainly failed: "Another Design Basis Accident for the AFRRI reactor is postulated to be the cladding failure of a fuel element during a pulse operation or inadvertent transient following steady state operation at 1 MW." Section 6.3.4.2 SAR, page 6-17, June 1981.

9. This postulate implies clearly that the probability (or likelihood) of clad failure is assumed to be 100% for the purposes of the ensuing computations. Because there cannot be a larger probability than certainty, the quoted part of the contention about ". . . much more likely. . ." is both irrelevant and factually incorrect.

10. Furthermore, Intervenor has quoted H. H. Hausner and F. Schumar, "Nuclear Fuel Elements", page 84 as support for the assertion that "surface cracks have been observed in fuel element cladding." The Intervenor has misused this reference. Page 84 discusses test samples of unclad zirconium hydride fuel meat, and is not relevant to fuel claddings.

Contention 2 (Accidents II)

Accidents can be expected to occur at the AFRRI reactor of a different kind and greater severity than those described in the HSR. Such accidents should be more properly designated DBA's to ensure that such accidents would not result in releases in excess of regulatory limits.

1) Fuel element storage rack failure. The HSR does not provide reasonable assurance that such an accident cannot occur in that: a) it fails to publish the calculations from which it concludes that a contact configuration of the twelve elements stored in Applicant's pool would not result in a critical mass; b) it does not cite the source for its statement that experience shows it takes approximately 67 closely packed fuel elements to achieve criticality.

2) Failure of an experiment. Applicant has failed to show that several instances of malfunctions of confinement safeguards at AFRRI could not recur during an experiment failure, resulting in the release of radiation in excess of occupational and offsite limits. Such malfunctions include: a) a breach of containment caused by missing rubber gasket sealing material on the double doors to the corridor behind the reactor control room, in violation of Applicant's Technical Specification, § 1.A.4. (See, Notice of Violation App. A, NRC Inspection Report Docket No. 50-170, 10/13/78); b) failure of the reactor room ventilation dampers to close on August 26, 1975 when the Continuous Air Monitor was alarmed (see, DNA Abnormal Occurrence Report to Directorate of Reactor Licensing, dated September 3, 1975, Docket No. 50-170, 9/10/75); c) failure of the lead shielding doors to stop opening at the fully opened position (see DNA Abnormal Occurrence Report, dated July 27, 1976, Docket No. 50-170, 8/16/76); d) reactor core position safety interlock malfunction on February 1, 1973 (not recorded in Docket No. 50-170).

Petitioner contends that human error coupled with failure of built-in safeguards could lead to a series of events resulting in releases of radioactivity in excess of regulatory limits and cites the following past malfunctions at AFRRI as evidence that such failures could occur there in the future: 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 (referred to in Contention 2b), Accidents II, supra); 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 T-2 output meter; g) manfunction of the Reactor Core Position Safety Interlock System on February 1, 1973, caused by a faulty de-energizing relay (referred to in Contention 2d), Accidents II, supra).

Applicant has not shown that the TRIGA reactor's negative temperature coefficient will automatically shut down the reactor in accident situations with damaged fuel elements, where the moderating effect of the hydrogen nuclei in the U-Zr-Hx alloy may be significantly reduced and the value of the negative temperature coefficient is changed.

4) Multiple fuel element cladding failure accidents have not been considered in the HSR. Such accidents could result from: a) defects in the material integrity of the fuel elements themselves; b) an uncontrolled power excursion in the reactor core; c) LOCA; d) sabotage, aircraft collision or natural ("act of God") accident.

11. (1) Fuel element storage rack failure. The contention states that ". . . the HSR fails to publish the calculations from which it concludes that a contact configuration. . .".

12. The Licensee has written and distributed and referenced a memorandum on this subject. Reference #2, page 4-32, SAR, June 1981.

13. The next part of this contention asserts that ". . . the HSR -- (b) it does not cite the source for its statement that experience shows it takes approximately 67 closely packed fuel elements to achieve criticality." It is true that the HSR (SAR) does not cite a reference. However, the relevant experience is contained in an AEC report documenting an inspection visit at AFRRI in 1965. (See AEC CO Report No. 179/65-2, Docket No. 50-170).

14. This report shows explicitly that fewer than 69 stainless steel clad fuel elements would not form a criticality condition in the AFRRI

core configuration. The data in this report are consistent with the HSR (SAR) assertion.

15. Malfunction of a confinement safeguard during an experiment failure. The contention states that the "applicant has failed to show that several instances of malfunctions of confinement safeguards at AFRRI could not recur during an experiment failure, resulting in the release of radiation in excess of occupational and offsite limits."

16. The Staff interprets this contention to mean the simultaneous malfunction of some component of the system designed to limit the release of radioactivity and failure of an experiment in progress that has generated a potentially hazardous quantity of radioactivity. Furthermore, we assume that the experiment is being performed in one of the experimental facilities to which the malfunctioning confinement safeguard is applicable.

17. The Staff position is as follows: the Licensee has not shown that the two types of events cannot occur in the future, either singly or in coincidence. On the other hand, we know of no man-made device for which it can be asserted with certainty that no future malfunction would ever occur. Therefore, it is essential to evaluate whether possible and/or credible malfunctions could put the public at risk, and to evaluate whether reasonable steps have been taken to mitigate the risks if malfunction were to occur. The Staff concludes that both facets of these considerations at AFRRI give

reasonable assurance that the public near AFRRI is not at significant risk from the simultaneous malfunction of the two independent events postulated in the contention.

18. A "confinement" must be a system or a device that could intercept radioactive material which may be somewhere on a route from where it is intended to be retained towards points where it is not intended to be.

19. Based on the wording of the contention, only the possible transport of radioactivity in the air need be considered.

20. In considering the definition of a confinement safeguard, and comparing that with the properties and functions of the systems listed in this contention, we delete two from consideration as follows: (a) the lead doors in the pool serve no confinement function. They are used to decrease the level of direct radiation in one of the exposure rooms when the reactor core is positioned at the far end of the pool. The primary purpose is to decrease radiation exposure to operations personnel while they are working in the exposure room. This function has zero potential impact on the external environment or the public. (b) the reactor core position interlock is simply a device to prevent the operator from inadvertently attempting to operate the reactor while the core dolly is in motion, or to move the dolly across the pool while the lead doors are closed. This interlock has no confinement function.

21. Of the four systems referred to in the contention, two are related to confinement of radioactivity, namely the reactor room ventilation dampers and the rubber gaskets on the double doors between the reactor room and the hallway on the control room level.

22. Reviewing the annual reports and event reports from the licensee, and the inspection reports from the AEC/NRC inspectors, we find in the record that the room dampers malfunctioned once, and the rubber gaskets were not in place at one time between 1962 and 1982. This does not provide enough data to support a prediction of frequency of future similar events.

23. The second facet of the contention deals with the failure of an experiment. As noted above, in order to be relevant to the contention, the experiment must contain a sufficient quantity of radioactivity to be capable of producing releases "exceeding occupational and off-site limits".

24. In order to be in a location that could be influenced by the above two confinement safeguards, the experiment would have to be in the pool, the in-pool experiment tube, or possibly in one of the pneumatic transport tubes.

25. An experiment in one of the two exposure rooms could not lead to airborne radioactivity in the reactor room itself, or in the hallway behind the control room.

26. The types of experiments authorized at AFRRI by the Technical Specifications (Section D, Experiments, currently applicable, and section II, 3.9 Experiments, proposed revision) would not lead to the release of radioactivity to unrestricted areas that would exceed 10 C.F.R. Part 20 Appendix B guidelines, even if total failure of an experiment were to occur and all contained gaseous and aerosol radioactivity were released.

27. The Staff has determined that the Technical Specifications provide reasonable assurance that almost exactly the scenario proposed by the Intervenor has been reviewed, and that there is reasonable assurance that if any authorizable experiment were to fail, off-site exposures would not exceed 10 C.F.R Part 20 limits. This conclusion is supported by the experience record at AFRRI, indicating that no potentially hazardous experiment has failed, between initial licensing in 1962, and 1982.

28. The second part of this contention alleges that ". . . human error, coupled with failure of built-in safeguards could lead to a series of events resulting in releases of radioactivity in excess of regulatory limits. . .". In order to respond to this contention, the Staff relies on its SER (NUREG-0882) and references therein. In summary, the SER concluded: (a) there are redundant systems on all of the important safety channels (§ 4.8.4); (b) a transient induced by the instantaneous insertion of all available excess reactivity does not exceed the safety limits of the fuel (§ 4.7.1); (c) a loss of all

coolant in a reasonable time interval, including the rate of loss of coolant proposed by the Intervenor, even following a very long steady-state run and pulse, will not lead to fuel-clad failure (§ 14.2.2); (d) loss of electrical power causes control rods to be released for gravity-fall into the core, leading to benign reactor shutdown (§ 4.8); (e) there are no high pressures or high temperatures during normal operation that can lead to disruptive disassembly of the reactor or dispersion of contained radioactivity (§§ 5.2, 14.2.1, 14.2.7).

The specific events that the Intervenor cites as "built-in safeguards" are discussed individually in the following:

29. "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.'" Review of the record shows that this statement is not supported by the facts. The Licensee submitted a report to NRC dated March 25, 1980. There was no NRC inspector or inspection involved. Furthermore, the malfunction of channel one was discovered by the Licensee during a routine check-out of instruments following an inadvertent electrical power outage over a week-end while the reactor was not operating. The reactor was not brought to power while the safety channel was inoperable. Therefore, no Technical Specifications were violated.

30. "Reactor exhaust system malfunction on August 9, 1979. This malfunction was reported to NRC by the Licensee in a letter dated August 15, 1979." The malfunction was apparently caused by a voltage surge on the facility electrical power system during an "electrical storm". The reactor was not in operation at the time of the occurrence, and the malfunction was discovered during the next routine pre-start check-out. This is exactly a major function of such check-outs.

31. "Malfunction of the fuel element temperature sensing circuit caused by a 'floating signal ground.' The Licensee discovered this malfunction, and reported it to NRC by letter dated August 1, 1979. As indicated by the letter, this temperature measuring system constitutes one of the primary safety channels. If the operator should malfunction, and both of the redundant safety channels should malfunction, the fuel temperature safety channel would be relied on to scram the reactor. The Licensee discovered this malfunction by observing the instruments on the control console prior to bringing the reactor to power. Apparently, the failure mode in this instance was such as to limit reactor operation to some power level less than the licensed power level. The Staff does not consider that there was a significant probability of the reactor exceeding any authorized or safety limits.

32. "Malfunction of the pool water level sensing float switch caused by wear on the jacketing around the wires leading to a switch." This

malfunction was reported to NRC by letter dated July 31, 1979. The malfunction stemmed from degraded insulation on a signal wire. While a loss of a large fraction of the water from the pool is considered to be significant enough to warrant a shut down of the reactor, the potential impact is primarily one of direct radiation exposure to the AFRRI staff. Therefore, the pool level switch is wired in to cause a reactor scram. Loss of switch function could result in loss of a scram function, but would not precipitate an emergency or lead to secondary unsafe reactor responses. The open pool top can be observed from the reactor control room (through a window) so the operator does not rely solely on the switch to be functional to provide evidence that sufficient pool water is present.

33. "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." This is the same incident that is the subject of contention 2.2) b) above. This event was reported to NRC by the Licensee by letter dated September 3, 1975. The Licensee discovered the malfunction while performing a routine, required pre-start-up check-out of the reactor, which precluded operation with the malfunction condition. The primary purpose of the check-out was fulfilled. In this event, preventative maintenance might have detected the loosening of the wires before loss of electrical contact occurred. On the other hand, the method of discovery (pre-start check-out) indicates that this check-out procedure was an acceptable adjunct to a less than perfect preventive maintenance procedure.

34. "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 T-2 thermocouple output meter." This malfunction was discovered by the Licensee during a routine pre-start-up- check-out of the reactor instrumentation. Because of the method of discovery, the Licensee decided that it did not qualify as an abnormal occurrence during operation, so it was not reported to AEC. Subsequently, a military inspection team recommended reporting the event, so the Licensee did. AEC/NRC policy was that the incident should have been reported promptly. This event also indicated a less than perfect preventive maintenance program but the pre-start-up check-out supplemented it, and precluded operation with a malfunctioning thermocouple safety channel.

35. "Malfunction of the Reactor Core Position Safety Interlock System on February 1, 1973, caused by a faulty de-energizing relay (referred to in Contention 2d), Accidents, II supra)." As discussed above, even though the Staff concurs that this was an equipment malfunction, the interlock would not qualify as a safety-related item. Therefore, this part of the contention is not germane to protecting the health and safety of the public, nor has the Intervenor proposed a relevant scenario.

36. This part of the contention asserts that "Applicant has not shown that the TRIGA reactor's negative temperature coefficient will automatically shut down the reactor in accident situations with

damaged fuel elements, where the moderating effect of the hydrogen nuclei in the U-Zr-H alloy may be significantly reduced and the value of the negative temperature coefficient is changed." The Staff agrees that the Licensee has not shown that the negative temperature coefficient of reactivity will automatically shut down the reactor in the accident situations postulated by the Intervenor. Nor does the Staff consider that such a demonstration is necessary. There is no requirement in reactor licensing that the temperature coefficient itself be capable of actually "shutting down" the reactor under all accident conditions that would be accompanied by fuel damage.

37. Under normal steady state operating conditions, it is not necessary that the negative temperature coefficient be large enough to cause reactor shutdown. It is conservatively sufficient that the coefficient be negative so that a reactor transient is mitigated rather than amplified by increasing temperature. Under all normal operating conditions and most accident conditions in research reactors, it is expected that the control elements will be operable or already inserted, so these will serve to shut down the reactor. Furthermore, for a reactor designed and loaded to operate on thermal neutrons, most of the material in the core that thermalizes, or moderates the neutron energy, is necessary for the chain reaction to continue. Thus, if a significant fraction of the hydrogen is removed from the core region of a TRIGA reactor, either by loss of coolant water or from the fuel meat itself, the fission chain

reaction will most likely no longer be self sustaining. That is, the reactor will shut down without reliance on the control elements. Hence, if a primary assumption of the contention occurs, namely ". . . the moderating effect of the hydrogen nuclei in the U-Zr-H alloy may be significantly reduced. . .", the effective reactivity of the reactor will decrease. Because this is also the result of a temperature increase if the negative temperature coefficient were functional, the end result is qualitatively the same.

38. In summary, while the contention is approximately correct (Applicant has not shown") the proposed scenario consists of two physical mechanisms whose results are similar, not opposite, as implied by the remainder of the contention. This contention is apparently based on an inadequate understanding of reactor physics by the Intervenor.

39. It is true that the HSR (SAR) does not explicitly consider multiple fuel element cladding failure. However, if the concern of the Intervenor is the release of fission product radioactivity, the consequences of multiple clad failure are simply derivable from the single clad failure treated. Furthermore, the Staff's SER (Sec. 14.2.7.1) does draw conclusions about the potential impact of multiple clad failures. In the following, the four proposed initiating mechansims for multiple clad failure (Contention No. 4) are discussed:

a) Defects in the material integrity of the fuel elements themselves.

The history of TRIGA fuel and fuel clad failures indicates that design, fabrication, and quality control are fully adequate to preclude such an incredible event. The Staff knows of no instance of multiple clad failures of TRIGA fuel in a single operational reactor attributed to defects in material integrity.

b) an uncontrolled power excursion in the reactor core.

The Staff's SER (§§ 4.7, 14.2.1), with references, concludes that this is not a credible scenario with the AFRRI reactor, as authorized, for causing multiple clad failures.

c) LOCA

The Staff's SER (§ 14.2.2), with references, concludes that this is not a credible scenario with the AFRRI reactor, as authorized, for causing multiple clad failures. Furthermore, loss of coolant by itself would shut the reactor down, because coolant is also a necessary component of the neutron moderator. When the reactor is subcritical, the principal source of heat energy is the beta-decay in the fuel. This source could not raise the fuel temperature rapidly (SER § 14.2.2).

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

Althouth small, the likelihood is not zero that any of these postulated events could occur in such a way as to cause multiple fuel clad failures. Hence, the Staff's SER (§ 14.2.7, 14.2.7.1) addressed the potential consequences to the public in unrestricted areas. Both the Licensee's (SAR, Sec. 6.3.4.2) and the Staff's analyses include specifically the loss of cladding integrity of a single fuel element. There is no basis in the Intervenor's contention for considering any scenario of multiple clad failures that requires more than simply multiplying the consequences of a postulated single-clad failure by the number of fuel-clads assumed to fail. To be conservative, this can be up to 90 elements, the total core loading. Based on these assumptions, the Staff's conservative analyses lead to hypothetical upper limits of both whole body dose and thyroid committed dose equivalent that lie within 10 CFR Part 20 Appendix B guidelines.

Contention 4 (Routine Emissions I)

Applicant has not demonstrated that airborne and waterborne radioactive emissions from routine operations and disposal of solid wastes will be maintained within the limits of 10 CFR Part 20 in that actual and probable violations of these regulatory limits have taken place on the occasions listed below and Applicant's radiation monitoring methods and corrective actions are inadequate to detect and prevent their recurrence.

1) Applicant's equipment, methods, and reporting system for measuring releases into the Montgomery County sanitary sewerage system and at its perimeter and offsite monitoring stations do not provide reasonable assurance that violations of regulatory limits have in all instances been or will be detected.

Environmental monitoring is inadequate to determine radiation doses to the public due to inhalation or ingestion because:

a) film dosimetry detects only external gamma radiation.

b) the particulate radioactivity monitor for airborne effluents (i.e. a pancake-probe CM 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) Applicant was cited by the NRC for a violation of environmental sampling and analysis procedures. 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.) Moreover, Applicant's "Environmental Sampling and Analysis" program does not provide adequate information on how quarterly environmental samples of water, soil and vegetation are prepared and analyzed, nor does it provide the raw data collected over the past ten years.

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.

2) An NRC inspection conducted January 10-12, 1979 revealed that, contrary to Applicant's Technical Specifications governing discharge of airborne radionuclides, Argon-41 and other radionuclides were discharged at ground level outside the reactor building for several months through a lead in the ventilation exhaust stack drain line (see NRC Inspection Report No. 50-170/79-01). It is highly probable that this resulted in releases in excess of the maximum permissible concentrations set forth at 10 C.F.R. Part 20, Appendix B.

3) Applicant's Airborne Release Reports for 1962, 1963, and 1964 and AEC Inspection Reports for the same years (Docket No. 50-170) reveal that releases of Argon-41 from Applicant's stack exceeded the maximum permissible concentration for unrestricted areas listed at 10 C.F.R. Part 20, Appendix B, during those years. (Also see letter from AEC to National Naval Medical Center (NNMC) dated October 6, 1961, Docket No. 50-170).

4) Applicant's Environmental Releas Data and Perimeter Monitoring Reports, Docket No. 50-170 (5/27/66 report and 9/20/66 report), show that

omissions from the AFRRI facility in 1962 and 1963 resulted in annual whole body doese in unrestricted areas in excess of the NRC's regulatory limit of 0.5 rem.

The specific subparts of the contention follow:

a) "Film dosimetry detects only external gamma radiation"

40. The Staff's SER (§ 11.3.3) and the Licensee's various reports (Docket No. 50-170) including annual reports and a special report dated 12/14/71 conclude that there are no measurable quantities of airborne radioactive particulates routinely released from the AFRRI reactor building. The only airborne reactor-related radioactive material released in measurable quantity is ^{41}AR. This is a typical emitter of beta rays accompanied by a gamma ray. It is also a noble gas, exhibiting the chemical inertness of a noble gas.

41. Committee II of ICRP (International Commission on Radiological Protection) has published recommendations related to the control of radiation exposures. These recommendations, on the whole, have been codified in 10 C.F.R. Part 20. The recommendations include the one that personnel exposures from immersion in a large cloud of a radioactive noble gas be based on the external whole body exposure, and exposure due to internal betas be considered relatively insignificant. Therefore, gamma ray environmental monitoring at AFRRI is consistent with 10 C.F.R. Part 20 guidelines and best international expert recommendations.

b) "the particulate radioactivity monitor for airborne effluents (i.e., a pancake-probe G-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 No. 50-170/77-01-03)."

42. The Licensee states, in a report transmitted to AEC by letter dated 12/14/71, Docket No. 50-170, that no detectable quantity of airborne radioactive particulates due to routine reactor operations is released to the unrestricted environment. Therefore, the detailed operating characteristics of the particulate monitor are not of crucial importance for routine operations. Furthermore, even though the Licensee has stated (in its 12/14/71 report) that the detector was not isokinetic, this was apparently based on a reluctance to claim something not actually evaluated. In the meantime, the Licensee has evaluated this detector system, and concluded that it meets the established criteria for being isokinetic. Licensee's Answers to Intervenor's Interrogatories dated October 30, 1981, Question No. 28(b). (For isokinetic criteria, see Appendix C, ANSI N13.1-1969 "Guide to Sampling Airborne Radioactive materials in Nuclear Facilities").

43. The unresolved status of the effluent monitor systems listed in Intervenor's reference: Inspection Report No. 50-170/77-01-03 was

resolved in a subsequent inspection and is discussed in Inspection Report No. 50-170/78-01.

c) "Applicant was cited by the NRC for a violation of environmental sampling and analysis procedures... 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.) Moreover, Applicant's 'Environmental Sampling and Analysis' program does not provide adequate information on how quarterly environmental samples of water, soil, and vegetation are prepared and analyzed, nor does it provide the raw data collected over the past ten years".

44. The contention is correct in that the Licensee was cited for not making a beta absorption correction as a part of the counting procedure to assess the concentration of radionuclides in liquid waste before release. This citation was categorized as a deficiency. However, the Licensee made the necessary measurements and evaluations, and determined that a correction of 10% was appropriate. The Licensee committed to make that correction in a letter to NRC dated March 7, 1977, and the open item was resolved by NRC as noted in Inspection Report 50-170/78-01 (item 77-01-02).

45. For the second part of this contention, the NRC staff accepted the Licensee's "environmental sampling and analysis program" as outlined in the enclosure with the letter to AEC, dated 12/14/71.

Furthermore, the inspector of the facility who reported in Inspection Report No. 50-170/77-1 found the program acceptable.

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

46. The Staff has seen no evidence that the Licensee uses the quoted model "... to derive its dose assessments to the environment...". This conclusion of the Staff is supported by the following: Licensee letter to AEC, with enclosure, 12/14/71; recent I&E inspection reports 50-170/70-1; 50-170/77-01; 50-170/78-01; 50-170/79-01, for examples.

The next part of the contention states "An NRC inspection conducted January 10-12, 1979 revealed that, contrary to Applicant's Technical Specifications governing discharge of airborne radionuclides, Argon-41 and other radionuclides were discharged at ground level outside the reactor building for several months through a leak in the ventilation exhaust stack drain line (see NRC Inspection Report No. 50-170/79-01). It is highly probable that this resulted in release in excess of the maximum permissible concentrations set forth at 10 C.F.R. Part 20, Appendix B."

47. The Staff concurs that there was a deviation from the Technical Specifications and, therefore, a violation of the AFRRI license. This violation was classified as an infraction by the NRC inspector (see Inspection Report cited by Intervenor). The referenced inspection report also states that the Licensee took action that resolved the problem before the inspector's report was submitted. Furthermore, the Licensee has stated that the referenced trap has since been mechanically capped to prevent future occurrence of this event (see Licensee's response to Intervenor's first set of interrogatories, question 31, dated November 3, 1981).

"Applicant's Airborne Release Reports for 1962, 1963, and 1964 and AEC Inspection Reports for the same years (Docket No. 50-170) reveal the releases of Argon-41 from Applicant's stack exceeded the maximum permissible concentration for unrestricted areas listed at 10 C.F.R. Part 20, Appendix B, during those years. (Also see letter from AEC to National Naval Medical Center (NNMC) dated October 6, 1961, docket No. 50-170)."

48. The Staff has reviewed the documentation in Docket No. 50-170 related to the release of ^{41}AR from the AFRRI facility. In summary, we have determined that the references cited by the Intervenor do indicate that AFRRI either did or potentially would (letter of October 6, 1961) sometimes release ^{41}AR in concentration's at the stack exit that exceeded the limits of 10 C.F.R. Part 20, Appendix B. However, improved methods of measurements, coupled with changes in

the AFRRI Building complex and using actual rather than hypothetical maximum operating schedules show that AFRRI's release of ^{41}AR currently falls well below the guidelines of 10 C.F.R. Part 20, Appendix B.

49. Current evaluation indicates that it is very likely that AFRRI's releases of ^{41}AR, averaged over a year, did not exceed 10 C.F.R. Part 20, Appendix B concentrations in unrestricted areas at any time in the past. SER §§ 12.9, 12.10.

50. October 6, 1961 letter. This letter referred to the Final Safeguards Report (FSR) for the DASA-TRIGA reactor, filed in support of the application for construction permit on September 19, 1961. Among other items, this letter requested that AFRRI provide a revised evaluation of potential exposures to persons in unrestricted areas resulting from anticipated releases of ^{41}AR. In March 1962, the AFRRI submitted a "Final Safeguards Report, Revised Edition" that contained updated and revised information, and superseded the previously submitted documentation. Appendix C of the revised FSR provides a revised evauation of the hypothetical release of ^{41}AR, and this new evaluation was acceptable to the Staff who reviewed it at the time. In the meantime, the actual operating experience supersedes those projected operating conditions. Thus, the reference quoted by the Intervenor constitutes only one step in the initial licensing of the applicant's facility, and the subsequent steps, which it is essential to include, were acceptable to the Staff.

51. Regarding the Licensee and inspection reports for 1962, 1963, and 1964 referenced by the Intervenor, I&E report No. 50-170/64-1 states that the Licensee's operations were found to be in noncompliance with 10 C.F.R. 20.106(b). This is the only one of these early inspection reports to cite a noncompliance, which is in contradiction to the Intervenor's contention. On the other hand, I&E report 50-170/64-2 discusses the perimeter monitoring program at AFRRI and concludes that the facility was operated in compliance with regulations. It is more relevant to AFRRI's current and expected future operations to consider more recent reports by the Licensee, and more recent I&E inspection reports. Examples of I&E inspection reports that provide broader coverage and more in-depth review of the environmental monitoring program related to airborne releases are 50-170/66-1, /68-2, /69-1, /70-1. The last report is especially thorough, and gives good reasons for concluding that AFRRI's airborne releases, which are principally ^{41}AR, have in all likelihood never exceeded 10 C.F.R. Part 20, Appendix B concentrations averaged over a year at ground level outside of the AFRRI buildings.

"Applicant's Environmental Release Data and Perimeter Monitoring Reports, Docket No. 50-170 (5/27/66 report and 9/20/66 report), show that emissions from the AFRRI facility in 1962 and 1963 resulted in annual whole body doses in unrestricted areas in excess of the NRC's regulatory limit of 0.5 rem."

52. I&E inspection reports for the period referenced, namely 1966, i.e., 50-170/66-1 and 50-170/66-2 review the environmental monitoring program, including monitor data from 1962 and 1963. The conclusion is reached that "The environmental monitoring to date shows the average yearly ambient radiation levels to be less tha 0.5 rem per year for unrestricted areas." (page 2, Appendix A, 17C/66-1).

Contention 8 (Accidents III)

Accidents can be expected to occur at the AFRRI reactor of a different kind and greater severity than those described in the HSR. Such accidents would result in significant offsite releases and include:

Two maximum credible accidents (MCAs) beyond the design basis of the reactor (Class 9 accidents): 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; and b) LOCA resulting in multiple cladding failures at an elevated temperature, followed by an explosive zirconium-air interaction.

53. The Staff has reviewed the extensive literature on the question of metal-steam and metal-air explosive chemical interactions. The Staff relied primarily on the literature referenced directly or indirectly in its SER as being the most relevant to a zirconium hydride fueled research reactor, and summarized its evaluations in Section 14.2.3 of the SER.

54. Very special conditions of metal droplet size, rate of formation, mixture ratio of metal and water molecules, and temperatures are necessary. Because of these considerations, it is very unlikely that an uncontrolled heating and dispersion of the components of a reactor core would lead to both the necessary and sufficient conditions to support an explosive chemical reaction.

55. Among the detailed considerations are the following:

a) It is necessary that the metal be finely divided, and hot.

b) It appears that this is not sufficient. The metal must be in the form of molten droplets of optimum sizes.

c) A nearly stoichiometric mixture of a significant fraction of the metal and water vapor molecules is necessary.

d) The rate of formation and dispersion of the metal droplets is important. If that rate is too slow, the rate of reaction of the mixture would be limited, and it would not explode.

e) The high temperature of the molten metal is necessary to initiate the rapid chemical reaction, because zirconium hydride is relatively inactive in water at temperatures below approximately 1000°C.

56. In addition to the factors listed above, there is another physical consideration. This is, not only must energy be supplied rapidly to raise the temperature of the solid metal to its melting point, but the latent heat of fusion must be supplied to cause melting. For zirconium hydride, this is equivalent to raising the temperature approximately an additional 500°C. Thus, to just cause melting requires the energy equivalent to raise the solid fuel temperature to (500 + 1800 =) 2300°C. No credible reactivity insertion in the AFRRI reactor could produce that much energy in a pulse. SER § 14.2.3.

Contention 9 (Accidents IV)

The analysis of the loss of coolant accident (LOCA) and the two design basis accidents (DBAs) within Applicant's Hazard Summary Report (HSR) is faulty in that:

1) It erroneously concludes that in event of an accident described therein as "Loss of Shielding and Cooling Water", air convection cooling would be sufficient to prevent cladding failure and significant fission product release.

Petitioner contends that in the event of a rapid loss of coolant while the reactor core is in the pulse mode, there could be a sudden temperature elevation sufficient to cause multiple cladding failures and fission product releases in excess of the limits provided in 10 C.F.R. Part 20.

57. The Staff asserts that this contention invokes a physically impossible scenario for the AFRRI reactor. The Staff's evaluation of a LOCA following long steady-state operation of the reactor followed immediately by a pulse of the maximum size possible with the actual conditions of the reactor would not lead to overheating of fuel sufficient to cause clad failure (see section 14.2.2 SER, and references cited). In that evaluation, the Staff assumed a rate of loss of water twice that accepted by the intervenor in his definition of "rapid" (see Intervenor CNRS's Response to NRC Staff's First Set of Interrogatories, December 3, 1981, Question No. 28.

58. Because the AFRRI reactor is designed and loaded to operate on thermal neutrons, loss of most of the coolant water from the core would reduce reactivity sufficiently that the reactor becomes subcritical, and any operation, either pulsing or steady state, is impossible. Therefore, either the reactor can be pulsable with water in and around the core, or the core can be devoid of water

and lacking the excess reactivity required for pulsing. These are mutually exclusive conditions for the AFRRI reactor.

59. Given the size of the AFRRI reactor pool, a leak of 250 gallons per minute would cause the surface of the water to fall at the rate of approximately 1 foot in 3 minutes. Thus, from the time the reactor would be pulsable to the time the surface of the pool would be below fuel level would require approximately 4 minutes. This is sufficient time to allow the water to remove a large fraction of the energy generated by the fissions during a large pulse, and, therefore, to lower the fuel temperature by several hundred degrees celsius. Thus, by the time all of the water had left the core, the fuel temperatures would be approximately as assumed for the LOCA analyses. SER §§ 14.2.1, 14.2.2.

60. Because no more than one pulse of significant size can be generated "during a LOCA", a single pulse is the credible scenario to be considered. The Staff's SER concludes that one pulse could not cause clad rupture. SER § 14.2.1.

Contention 10 (Routine Emissions II)

Applicant has not demonstrated that airborne and waterborne radioactive emissions from routine operations and disposal of solid wastes will be maintained within the limits of 10 C.F.R. Part 20 in that actual and probable violations of these regulatory limits have taken place on the occasions listed below and Applicant's radiation monitoring methods and corrective actions are inadequate to detect and prevent their recurrence.

3) Since Applicant's Environmental Impact Appraisal (EIA), submitted in conjunction with its license renewal application, admits that the highest average unrestricted area exposure rate from

airborne releases (set forth in the EIA) extends to residential areas, it is highly probable that such exposures have resulted and continue to result in doses to the public in excess of 0.5 rem and, violate the principle that emissions from Applicant's operation be kept as low as is reasonably achievable (the ALARA principle).

Petitioner bases this conclusion on (1) the AFRRI Environmental Release Data and Perimeter Monitoring Reports, Docket No. 50-170 (including but not limited to the 5/27/66 report, 9/20/66 report, and 12/14/77 report), and (2) AFRRI's written response to Mr. Joe Miller's (from Citizens for Nuclear Reactor Safety, Inc.) question #11, Autumn 1979. (The written questions and responses are in Petitioner's possession.)

4) Applicant's Environmental Release Report, issued 12/14/71, indicate that between 1/1/70 and 1/1/71 exposure rates in several unrestricted areas were as high as 1-5 mRad/hr. At this rate, any person who lived or worked in these areas 500 hours in a year, or about 10 hours a week, would receive an annual whole body dose in excess of the NRC's limit of 0.5 rem/hr. Since 50-60% of the area within a one mile radius of the AFRRI stack is residential, it is highly probable that the population dose limit was exceeded during this period. This is a violation of the ALARA principle. Because these measurements are taken only a few times a year, it must be assumed, in the absence of more complete data, that these dose rates represent the average dose rates to unrestricted areas over long durations of time.

61. Careful reading of the quoted reports of item 1 (apparently the 12/14/77 should read 12/14/71) does not support the allegation that "exposures <u>have</u> and <u>continue</u> to result in doses to the public in excess of 0.5 rem. This is not a substantively different contention from No. 4, Routine Emissions I, and is disposed of by the Staff's arguments in that contention.

62. The Intervenor's use of AFRRI's response to question No. 11 to Mr. Joe Miller falls in the same category. The highest measurement on any monitor station, totalled for the subject year is 30 millirem. This is clearly at least a factor of ten below the Intervenor's contention. Thus, the contention is refuted rather than supported by the Intervenor's references.

63. The Intervenor has misunderstood the information in the cited AFRRI report in item 4). The highest integrated exposure for one year measured in an unrestricted area was 76 mrem, accumulated during calendar year 1970 (page 3, question 3 b.(1)). This environmental station was close to and influenced by the Maxitron 250 x-ray facility. The highest short term exposure rate at this location, measured by the Licensee with ionization chambers, is quoted by the Licensee as approximately 5.0 mrem/hr during maxitron operation. Both the Licensee's reports, and AEC/NRC inspections have confirmed the source of this high exposure rate (see I&E report No. 50-170/70-1, dated 2/27/70, especially page 6, item (3)).

64. To quote the referenced AFRRI report, "the maximum dose recorded, potentially resulting from AFRRI radioactive releases is 8 millirem in calendar 1970, and 8 millirem in 1971 (through 9/30/71)." These data are quoted on page 3, question 3.f, and are consistent with the I&E inspection report referenced above. (50-170/70-1).

Robert E. Carter

Robert E. Carter

Subscribed and sworn to before me
this 24th day of February 1983

Notary Public

My Commission Expires: 7-1-86

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)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF MOTION FOR SUMMARY DISPOSITION" in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class, or, as indicated by an asterisk, through deposit in the Nuclear Regulatory Commission's internal mail system, this 25th day of February, 1983:

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