UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

MEMORANDUM IN SUPPORT OF MOTION TO COMPEL ANSWERS TO LICENSEE'S FIRST INTERROGATORIES TO THE CITIZENS FOR NUCLEAR REACTOR SAFETY, INC.

INTRODUCTION

On August 31, 1981 this Board entered an Order which provided, *inter alia*, that the parties would file their first sets of interrogatories on September 30, 1981 and the answers to the first sets of interrogatories on October 30, 1981. All parties filed their first sets of interrogatories on September 30, 1981. The Licensee responded to the Intervenor's first set of interrogatories on October 30, 1981. The Intervenor, however, on October 31, 1981, requested a 30 day enlargement in which to respond to Licensee's first set of interrogatories. A purported response to Licensee's first set of interrogatories was ultimately filed by the Intervenor on December 29, 1981 and received by the Licensee on January 4, 1982. Comparison of the Licensee's First Interrogatories to the Citizens for Nuclear Reactor Safety, Inc. (Exhibit A) with Intervenor CNRS's Response to Licensee's First Set of Interrogatories (Exhibit B) discloses that many of the purported answers are evasive, incomplete, or both. Accordingly, Licensee has submitted herewith its Motion to Compel Answers to Licensee's First Interrogatories to the Citizens for Nuclear Reactor Safety, Inc.

QUESTION PRESENTED

Although 10 C.F.R. § 2.740(f) provides that "failure to answer or respond shall not be excused on the ground that the discovery sought is objectionable unless the person or party failing to answer or respond has applied for a protective order," a condition not here present, it may be helpful to discuss the Licensee's purpose in submitting its first set of interrogatories before addressing the responsiveness of the Intervenor's answers to specific interrogatories. The August 31, 1981 Order issued by this Board clearly contemplated submission of motions for summary disposition upon the conclusion of the second round of discovery and it is the intent of the Licensee to submit such a motion. Many of the contentions involved in these proceedings are, in the opinion of the Licensee, either irrelevant to the safety of reactor operations or totally without scientific merit. Because the contentions are broadly stated and set forth only the ultimate conclusions of the Intervenor, their relevancy and scientific merit, if any, can only be determined by examining the underlying hypotheses, calculations, and assumptions. The intent of the Licensee, throughout the first set of interrogatories, was to obtain additional information and more clearly focus the issues. As will be demonstrated below, the non-responsive answers submitted by the Intervenor completely frustrate this effort.

Answers which are incomplete or evasive must, under 10 C.F.R. § 2.740(f), be treated as failures to answer or respond. In determining whether an answer is incomplete or evasive, the Board should consider the fact that these proceedings have now been pending for more than a year and are rapidly approaching the originally established summary disposition phase. If the Intervenor has any bases for its contentions, such bases must now be disclosed. If, on the other hand, the Intervenor's contentions have no foundation, both the Licensee and the Board are entitled to be informed of that fact.

For the convenience of the Board and the parties, the following discussion is keyed to the number of the interrogatory and purported answer concerned.

SPECIFIC FAILURES TO RESPOND WHICH ARE THE SUBJECT OF THE ATTACHED MOTION

INTERROGATORY 6. The Intervenor refused to respond to this interrogatory, asserting that it did not yet know what experts will be called. At this stage of the proceedings at least some information regarding anticipated expert testimony must be available to the Intervenor. Moreover, the Intervenor, in refusing to respond to interrogatories numbers 27, 28, 29, 30, and 34, specifically identified "experts" by name and stated that the named individuals would testify as to those matters at the hearing. Licensee submits that the Intervenor's responses to the substantive interrogatories are utterly inconsistent with the position taken by the Intervenor with regard to interrogatory number 6.

INTERROGATORY 7b. The Intervenor failed to answer the interrogatory as stated. Although the Licensee specifically requested the basis and supporting documentation for the Intervenor's inference that the Licensee's "HSR erroneously assumes that during an inadvertent transient a peak fuel temperature would be below 100°C," the Intervenor in its response commented upon 600°C rather than 100°C. Moreover, the Intervenor refused to provide any supporting documentation. The Intervenor's response is irrelevant to the interrogatory and must be considered evasive. If the Licensee and this

Board are to be expected to assess and evaluate the Intervenor's inference, the underlying basis and references must be provided.

INTERROGATORY 7c. The interrogatory was not answered as stated. The Licensee specifically requested "references that state the cladding temperature accompanying the 'elevated fuel temperature' postulated by the petitioners." The Intervenors responded by stating an assumption and referring to any elementary thermodynamics text. Although the Licensee does not expect or require specific references defining the Adiabatic Model for Heat Transfer, the interrogatory clearly requires the Intervenor to cite the references, if any, which support their assumption that the "cladding temperature will essentially mirror the fuel temperature" during a pulse operation or inadvertent transient. Absent such information, neither the Licensee nor the Board can test the validity of the Intervenor's assumption.

INTERROGATORY 7d. The Intervenor failed to answer this interrogatory as stated. The Licensee requested a specific temperature associated with the Intervenor's alleged "elevated fuel temperature." The Intervenor's response merely cited a rise in fuel temperature of "several hundred degrees" without even specifying the units of measurements (types of degrees). Although the Intervenor provided a list of variables in an effort to justify its vague, non-specific response, the variables cited are irrelevant to both the interrogatory and the Licensee's reactor. The "elevated fuel temperature" was postulated by the Intervenor in its contention. The Licensee submits that the Intervenor can and should be required to provide specific information regarding its contention. Otherwise, neither the Licensee nor the Board can assess the likelihood of occurrence or probable effects.

INTERROGATORY 7e1. The Licensee agrees with the statement made by the Intervenor in response to this interrogatory. Unfortunately, however, the response does not answer the interrogatory as stated because the Licensee asked the Intervenor to provide a specific reactivity insertion which the Intervenor considers to be associated with its definition of inadvertent transient. The Intervenor's response is evasive and fails to provide either the Licensee or the Board with any specific information regarding its contention.

INTERROGATORY 7e2. The Licensee agrees with the statement made by the Intervenor in response to this interrogatory. Unfortunately, however, the response does not answer the interrogatory as stated because the Licensee asked the Intervenor to provide a specific maximum power level which the Intervenor considers to be associated with its definition of inadvertent transient. The Intervenor's response is evasive and fails to provide either the Licensee or the Board with any specific information regarding its contention.

INTERROGATORY 7e4. Although the Intervenor attempted to answer the interrogatory by asserting that multiple human and mechanical errors can, taken collectively, render some system safeguards ineffective, the response is totally irrelevant because the Intervenor never even attempted to explain why the behavior of the reactor would differ.

INTERROGATORY 7j. Although the Licensee specifically requested that the Intervenor provide references, the Intervenor failed to do so. Accordingly, the response is incomplete and provides no documentary basis upon which to evaluate the Intervenor's contention.

INTERROGATORY 7n. Although the Licensee specifically requested that the Intervenor provide references, the Intervenor failed to do so. Accordingly, the response is incomplete and provides no documentary basis upon which to evaluate the Intervenor's contention.

INTERROGATORY 8b. This interrogatory was not answered. The Intervenor failed to define in its words, "greater severity," and also failed to provide specific magnitudes or levels and its basis for the same. It is apparent that the response to this interrogatory was evasive and does not provide the Licensee with any basis upon which to take additional action.

INTERROGATORY 8c. The Intervenor failed to answer this interrogatory as stated. Licensee requested a quantitative likelihood with justification, for accidents proposed by Intervenor. The Intervenor did not even attempt to characterize and justify the relative likelihood of its proposed accidents that are of a different kind.

INTERROGATORY 9a. Intervenor failed to answer this interrogatory as stated. Licensee requested bases and justification for "assuming that the failure of a storage rack would result in an alleged accident of a 'greater severity' than those in the HSR." Intervenor cites information which is irrelevant to the question of "greater severity," cites information dealing with more than one fuel element storage rack failure, and fails to cite any supporting documentation or justification pertaining to the interrogatory.

INTERROGATORY 10a. Intervenor failed to provide a complete answer to this interrogatory. Intervenor provided an adequate definition of an "experiment failure," but failed to respond to "what specifically occurs during an experiment failure and what results are to be expected?"

INTERROGATORY 10h. Intervenor failed to answer this interrogatory as stated. Licensee requested, "What specific MPC levels (radionuclide and concentration) are the petitioners contending would be exceeded by the alleged failure of an experiment?". Intervenor stated that the answer, "depends on the specific experiment sanctioned by the AFRRI." The

Intervenors' response provides no clarification whatsoever. The Licensee is interested in specific consequences associated with experiment failures that the Intervenor considers possible and important.

INTERROGATORY 12a. The Intervenor failed to answer this interrogatory as stated. The Licensee requested consequence data for the alleged malfunction and presumed accident. The Intervenor stated that it needs more detailed information about the physical layout and operational history in order to answer this interrogatory. Licensee contends that this information is available in the public record or has been directly provided to the Intervenor in documents prepared and submitted in conjunction with this relicensing effort. Moreover, the Intervenor cited the alleged malfunction as part of its contention. Unless the Intervenor has the minimal information required to answer this interrogatory, there would appear to be no basis for inclusion of the alleged malfunction as part of the contention. In any event, the Intervenor's response is clearly evasive and must be treated as a failure to respond.

INTERROGATORY 12b. The Intervenor failed to answer or even attempt to answer this interrogatory. The Intervenor's response in no way addressed the question, "How was this failure cited above detected?". The Licensee contends the answer to this interrogatory is readily available and a matter of public record. The Intervenor should be compelled to answer this interrogatory which the Licensee contends is of significant importance to the associated contention.

INTERROGATORY 12c. The Intervenor failed to answer this interrogatory as stated. The Licensee requested Intervenor's feelings which does not require more detailed information on the physical layout or operational history at AFRRI as Intervenor states. The Intervenor's response therefore is irrelevant, incomplete, and evasive.

INTERROGATORY 12d. The Intervenor failed to answer this interrogatory as stated. The Intervenor stated that it needs more detailed information about the physical layout and operational history in order to answer this interrogatory. Licensee contends that this information is available in the public record or has been directly provided to the Intervenor in documents prepared and submitted in conjunction with this relicensing effort. Moreover, the Intervenor cited the alleged malfunction as part of its contention. Unless the Intervenor has the minimal information required to answer this interrogatory, there would appear to be no basis for inclusion of the alleged malfunction as part of the contention. In any event, the Intervenor's response is clearly evasive and must be treated as a failure to respond.

INTERROGATORIES 13, 14, 15, 16, 17, 18, 19, 20, 21, and 22. The Intervenor has clearly failed to answer or even attempt to answer any of these interrogatories. The Intervenor has incorporated in its Failure of Experiment contention a series of alleged "malfunctions of confinement safeguards at AFRRI." The relationship of the alleged malfunctions to the Intervenor's postulated "release of radiation in excess of occupational and offsite limits" is not clear. Interrogatories 13 through 22 inclusive were designed to elicit clarification of this relationship, if any. The Intervenor totally refused to respond to these interrogatories and, indeed, appears to have attempted to disassociate itself from its own contention. The Licensee and the Board are clearly entitled to know whether the Intervenor contends that any of the alleged malfunctions could give rise to the effect postulated by the Intervenor.

INTERROGATORY 23a. The Intervenor failed to answer this interrogatory as stated. The Licensee requested the Intervenor to cite the mechanism by which a damaged TRIGA fuel element's moderating characteristics would be changed and also specifically requested documentation and references to support such a claimed change in the fuel's moderating effect. Intervenor provided a treatise, with references, on neutron moderation for uranium-zirconium hydride fuel and cited a claimed potential mechanism for altering the moderation characteristics of damaged TRIGA fuel but never cited any substantiation or supporting documentation to verify that its claimed mechanism is valid or is possible. That is, Intervenor cited a mechanism as requested but failed to cite specific references supporting and verifying that its proposed mechanism can indeed alter the moderating effect of damaged uranium-zirconium hydride TRIGA fuel as specifically requested. Therefore, the Intervenor has failed to provide a complete answer to this interrogatory.

INTERROGATORY 23b. The Intervenor has failed to answer this interrogatory due to its failure to provide specific references to support its claims as requested.

INTERROGATORY 23c. Intervenor failed to answer this interrogatory. Licensee requested Intervenor's understanding of the differences between a thermal and fast reactor in order to gain an understanding of the Intervenor's claims and contentions which address criticality and inherent shutdown mechanisms. Intervenor states that this interrogatory is irrelevant. Licensee contends that it is relevant because it has a direct bearing on the Licensee gaining an insight into the Intervenor's contentions and claims concerning the AFRRI reactor, as well as the Intervenor's understanding of the mechanisms controlling thermal reactors in general.

INTERROGATORY 23d. The Intervenor has failed to answer this interrogatory due to its failure to provide specific references to support its claims as requested.

INTERROGATORY 23g. The Intervenor failed to answer this interrogatory as stated due to its failure to answer the question, "Would this be a positive or negative effect?", and its further failure to cite supporting documentation for any claims.

INTERROGATORY 23h. The Intervenor failed to answer this interrogatory due to its failure to cite references or show calculations to support its claims as requested.

INTERROGATORY 23i. The Intervenor totally refused to answer this interrogatory. Rather, it stated that this interrogatory will be addressed by direct testimony at the hearing. The Intervenor's response is clearly evasive and in direct contradiction to the purpose of the discovery phase of these proceedings. Absent a response to this interrogatory, the Licensee can neither proceed with the second round of discovery nor prepare its motion for summary disposition.

INTERROGATORY 24a. Intervenor failed to answer this interrogatory as stated. Licensee specifically asked "What is the basis that the HSR or SAR does not consider multiple cladding failure accidents?" Intervenor's response stated that "They do not believe multiple cladding failure accidents are credible."

Licensee is at a loss as to whom the term "they" refers. More significantly, the Intervenor never cites to any specific portion of the HSR or SAR as clearly required by the interrogatory.

INTERROGATORY 24c. Intervenor failed to answer this interrogatory as stated. Licensee specifically requested Intervenor to identify and cite references to support claimed mechanisms causing "cladding failure." The Intervenor responded by addressing predominantly mechanisms for causing fuel material failure--not necessarily to include the cladding.

INTERROGATORIES 24g, 24h, AND 24i. Intervenor simply and clearly failed to answer or attempt to answer these interrogatories.

INTERROGATORIES 25a, 25b, AND 25c. Intervenor has failed to answer these interrogatories. Although it states that, "Liz Entwistle will respond," to date she has not done so.

INTERROGATORY 26a. Intervenor failed to answer this interrogatory. Licensee requested, "What specific waterborne radioactive emissions are generated by rou'ine operations?". Intervenor responded by quoting the Licensee's statement that "no waterborne radioactive emissions are gererated by routine operations," and then proceeded to ask a question. If the Intervenor does not accept the Licensee's statement quoted in its response, it should be required, in response to this interrogatory, to disclose the basis for its contention.

INTERROGATORY 26g. Intervenor failed to answer this interrogatory. Licensee requested the Intervenor to discuss, "how corrective actions are inadequate to detect and prevent violations of regulatory limits?". Intervenor made a philosophical or political statement but never addressed the interrogatory as stated. If the Intervenor's contention is based solely on the response provided to this interrogatory, it clearly should be seeking rulemaking action rather than intervention in license renewal proceedings.

INTERROGATORY 261. The Intervenor failed to answer this interrogatory. In response to this interrogatory the Intervenor merely alluded to its responses to Interrogatory 26, parts f, g, and h. Review of the cited responses discloses that the Intervenor only cited what it considers inadequacies, but did not provide an answer to Interrogatory 261, specifically; "What safeguards (equipment or procedure) would satisfy the Petitioners that proper monitoring methods are in use at AFRRI for checking emissions from routine operations."

INTERROGATORIES 27, 28, 29, AND 30. Intervenor clearly failed to answer these interrogatories. In its response, the Intervenor directs the Licensee to, "see testimony of Professor Ernest J. Sternglass to be presented at the hearings." The purpose of discovery is to provide the Licensee with specific information and insight into the Intervenor's concerns, contentions, and the bases for its concerns and contentions. Through evading the Licensee's interrogatories, by attempting to defer response until the hearing, the Intervenor clearly violates the purpose and intent of the discovery phase and provides no opportunity for the Licensee to gather information and evidence for summary disposition. Indeed, the Intervenor's refusal to respond leaves the Licensee totally at a loss as to what it must defend against.

INTERROGATORIES 31b AND 31c. Intervenor failed to answer these interrogatories as stated. Intervenor's response to Interrogatory 31 is accepted as an adequate response to Licensee's interrogatory 31a. However, Interrogatories 31b, and 31c, are never addressed or answered. Furthermore, Interrogatory 31 was intended to elicit information which would disclose whether this extremely serious allegation has any foundation. If the allegation of the Intervenor is, as Licensee believes, utterly without foundation, that fact should now be disclosed and the inflammatory contention summarily dismissed. If the Intervenor has a foundation for its contention, it must disclose that information in response to the Licensee's interrogatory.

INTERROGATORY 32b. Intervenor failed to answer this interrogatory as stated. Intervenor postulates, without citing supporting documentation as requested, that "internal gap pressures could rise to 1800 PSI or more depending on the temperature elevation." The Intervenor's response to this very specific interrogatory is vague and evasive. Furthermore, Intervenor failed to "prove," through calculations and references, that its claimed peak internal gap pressures could indeed be achieved for a LOCA at the AFRRI reactor facility, as specifically requested by the Licensee.

INTERROGATORY 32c. Intervenor has failed to completely answer this interrogatory since it refused to provide a response to a significant portion of the interrogatory. Specifically, "Have any resulted in fission product release to the unrestricted area environment? If the answer is yes, please provide documentation or reference it." The Intervenor's limited response does not afford Licensee sufficient information to proceed with the second round of discovery or prepare for summary disposition.

INTERROGATORIES 34a, 34b, 34c, 34d, 34e, AND 34f. The Intervenor totally refused to answer these interrogatories. In its response, the Intervenor states that, "We expect this topic to be discussed by several of our witnesses at the time of the full hearings, including Dr. Irving Stillman, Dr. Ernest Sternglass, and Dr. Irwin Brass." The purpose of discovery is to provide the Licensee with specific information and insight into the Intervenor's concerns, contentions, and the bases for its concerns and contentions. By evading the Licensee's interrogatories, through deferment of information in response to interrogatories until the hearing, the Intervenor clearly violates the purpose and intent of the discovery phase and provides no opportunity for the Licensee to gather information and evidence for summary disposition. Moreover, the Intervenor's refusal to respond leaves the Licensee totally at a loss as to what it must defend against. This matter has now been pending for more than a year. At some point the Intervenor must be required to disclose the bases, if any, for its inflammatory and prejudicial allegations. The Licensee submits that this time has come.

INTERROGATORY 35e. The Intervenor failed to answer this Interrogatory as stated due to its failure to "justify and support any statements" made in its response as specifically requested.

INTERROGATORIES 35f. AND 35h. The Intervenor failed to answer these Interrogatories as stated due to its failure to "cite references to support the elements given" as specifically requested by the Licensee.

INTERROGATORIES 35g. AND 35i. Intervenor clearly failed to answer or even attempt to answer these interrogatories as stated. Licensee requested that Intervenor "show and justify" with documentation that all elements and conditions necessary for an alleged explosive zirconium-air or zirconium-steam interaction are indeed present in the AFRRI TRIGA core. In response to this request, Intervenor provided evasive and incomplete responses replete with conjecture. Absent supporting references, the Licensee cannot be expected to understand the basis for Intervenor's conjecture. If the Intervenor truly has a documented basis for its contention concerning explosive chemical interactions in the AFRRI TRIGA, then it should disclose this information to the Licensee and Board at this time.

INTERROGATORY 36a(2). Intervenor failed to answer this interrogatory as stated due to its failure to "specify which monitor location" and "which year" as specifically requested by the Licensee.

INTERROGATORIES 36a(3) AND 36a(4). The Intervenor failed to answer these interrogatories as stated. The Intervenor states that these interrogatories, "have already been discussed" but does not indicate where in the text the Intervenor feels it has discussed the cited interrogatories. Review of the responses to previous interrogatories fails to disclose answers to the questions asked in Interrogatories 36a(3) and 36a(4). Accordingly, the Intervenor has failed to respond to these interrogatories.

INTERROGATORY 36c. The Intervenor failed to answer this interrogatory as stated. The Intervenor stated that data is contained in Tables 1 and 2 given above. Licensee agrees that data is indeed presented in Tables 1 and 2 but this data has no relevance to Interrogatory 36c. Specifically, Licensee requested Intervenor to specify the residential area in which the population would 'highly probably' have received a dose in excess of 10 C.F.R. 20. This information is not contained within Tables 1 and 2 or elsewhere in the Intervenor's response.

INTERROGATORY 36d. The Intervenor failed to completely answer this interrogatory since it failed to answer or address the following parts: (1) "Please give any information you possess as to why and under what conditions these measurements were made?", (2) "Were they due to environmental releases?", (3) "Cite all references to support any statements made."

INTERROGATORY 36g. Intervenor clearly failed to answer this interrogatory. Licensee here is requesting the Intervenor to explain and show how the AFRRI reactor core radioactive inventory has anything to do with routine emissions as the Intervenor contends. Intervenor does not answer this question and is evasive. Intervenor's response is totally irrelevant to the interrogatory.

REQUEST FOR STAY

As indicated in the attached Declaration, counsel for the Intervenor and the Licensee have discussed the matters raised above and agreed to attempt to resolve the question of the adequacy of the Intervenor's response informally.

Although the motion to compel is being filed at this time to comply with the time requirements set forth in 10 C.F.R. 2.740(f) and protect the rights of the Licensee, we request that the Board stay consideration of the motion for a period of 45 days. The Licensee will submit a supplemental memorandum to advise the Board of the progress made, if any, in the attempt to amicably resolve this matter.

Respectfully submitted

Robert L. Brittigan

ROBERT L. BRITTIGAN
General Counsel
Defense Nuclear Agency
Counsel for Licensee

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)

LICENSEE'S FIRST INTERROGATORIES TO THE

CITIZENS FOR NUCLEAR REACTOR SAFETY, INC.

PLEASE TAKE NOTICE, that pursuant to 10 C.F.R. 2.740b and the Special Prehearing Conference Memorandum and Order filed herein on September 1, 1981 you are hereby directed to answer in writing and under oath or affirmation the following interrogatories within thirty (30) days.

INSTRUCTIONS

A. These interrogatories are directed to the Citizens for Nuclear Reactor Safety, Inc. (CNRS) and the answers hereto are to be completed to the best knowledge of CNRS, its officers, agents, attorneys, investigators, employees, technical advisors, and other representatives.

B. These interrogatories are continuing in nature and CNRS is requested to provide by way of supplementary responses hereto in accordance with 10 C.F.R. 2.740(e) such additional information as may hereafter be obtained

EXHIBIT A

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by CNRS or any person on CNRS's behalf, which will augment or otherwise modify any answers now given to these interrogatories. Such supplementary responses are to be filed and served upon the Licensee within thirty (30) days after receipt of such information, but no later than fourteen (14) days before the final prehearing conference.

C. Where an individual interrogatory calls for an answer which involves more than one part, each part of the answer should be fully set forth and numbered or lettered to correspond with the appropriate sub-part of the interrogatory.

D. For the convenience of the Atomic Safety and Licensing Board, the Nuclear Regulatory Commission Staff, and the parties hereto, the following interrogatories are keyed to specific contentions raised by CNRS. In each case the relevant contention or applicable portion thereof is identified in capital letters immediately preceding the interrogatories.

GENERAL QUESTIONS

INTERROGATORY 1

Please list the name and address of all members of the petitioning group known as Citizens for Nuclear Reactor Safety, Inc. (CNRS).

INTERROGATORY 2

Please list the name and address of all persons who are members of CNRS and participated in preparing the contentions submitted to the Atomic Safety and Licensing Board.

INTERROGATORY 3

Please list the name and address of any person or firm used by the Citizens for Nuclear Reactor Safety, Inc. to prepare any contention before the Atomic Safety and Licensing Board. Indicate which contention they were involved with, the approximate time spent, how much, if any compensation was paid.

INTERROGATORY 4

Please list the qualifications and experience of each and every person involved with formulating the contentions, basis for the contentions and any research required to submit the contentions. If there were no technically oriented individuals preparing the contentions, please give the source of technical information required to formulate the contentions.

INTERROGATORY 5

Please list all persons answering these questions. Indicate which specific question was answered. If these persons are different than those listed in questions 3 or 4, please give name, address, and qualifications to answer questions.

INTERROGATORY 6

Please list the names, addresses, and employers, if any, of any expert or other technical witnesses which CNRS intends to call in support of its contentions. With respect to each such witness state the professional qualifications, the contention with respect to which the witness is expected to testify, and the substance of the expected testimony.

ACCIDENTS I - THE ANALYSIS OF THE "FUEL ELEMENT CLAD FAILURE ACCIDENT," ONE OF THE TWO DESIGN 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.

INTERROGATORY 7

a. Where specifically in the AFRRI Hazard Summary Report or SAR does it assume that a cladding failure would occur at a peak fuel element temperature of less than $100^{\circ}C$?

b. What is the basis for the inference that the HSR erroneously assumes that during an inadvertent transient a peak fuel temperature would be below $100^{\circ}C$? State the references to support your basis.

c. Give references that state the cladding temperature accompanying the "elevated fuel temperature" postulated by the petitioners.

d. Specifically what temperature is the elevated fuel temperature supposed to reach?

e. What is an inadvertent transient?

1. What reactivity insertion is to have occurred?

2. What is the maximum power level this "inadvertent transient" is supposed to reach?

3. How is this "inadvertent transient" initiated?

4. How does this "inadvertent transient" behave differently such that the same temperature shutdown mechanism that controls a planned transient would not control the inadvertent one?

f. How is the elevated fuel temperature (in excess of 400°C) determined, by the petitioners, during a pulse operation that assumes a cladding failure? Please provide a reference to any documents relied upon and a copy of any calculations made.

g. Give the references that show the "much more likely" occurrence of a cladding failure at fuel temperatures in excess of 400°C.

h. Quantitatively, how "much more likely" is a cladding failure at a fuel temperature at 400°C? Above 400°C? Please state references and provide a scenario for achieving your cited temperature above 400°C.

i. How many degrees above 400°C is necessary to cause the "more likely occurrence" of a cladding failure? Please state references.

j. What mechanism results in "the greater gap activity" following a pulse rather than a steady state operation? State conditions necessary for the mechanism and cite references.

k. How does a greater gap activity by itself result in a greater fission product release?

l. The statement "Petitioner contends that such cladding failure would be much more likely to occur at elevated fuel temperatures (in excess of 400°C)," indicates that the petitioners feel that the probability of a cladding failure is the same or essentially the same below 400°C without respect to temperature. Is this statement correct?

m. At what fuel temperature would the petitioners recommend AFRRI calculate fission product releases resulting from a cladding failure and what core history (considering maximum operations) should be used?

n. What conditions (specifically temperature, pressure, etc.) are necessary to cause a cladding failure? State references.

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.

INTERROGATORY 8

a. What specific accidents other than described in the AFRRI Hazard Summary Report or Safety Analysis Report could be "expected to occur" and what is the basis for the statement that these accidents can be expected to occur?

b. What is "greater severity"? Please state a specific magnitude or level, and show the reason for selection of this level.

c. Give and justify the quantitative likelihood (probability) that these accidents of a different kind will occur.

d. What qualifications are necessary for a person to be considered qualified to "properly designate" an accident as a DBA for the AFRRI TRIGA reactor?

e. What specific regulatory limits are AFRRI supposed to ensure would not be exceeded as a result of these alleged accidents?

f. Since one of the purposes of a DBA is to evaluate the hazard of one or two possible "worst case" accidents, what specific accident and what specific magnitude should be considered as the AFRRI DBA. Please site references, data, and any other information to substantiate that any proposed "worst case" accident is in fact valid and more relevant that those considered in the AFRRI Safety Analysis Report or Hazard Summary Report.

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.

INTERROGATORY 9

a. What are the petitioners bases for assuming that the failure of a storage rack would result in an alleged accident of a "greater severity" than those in the HSR? Please cite specific references from which the bases are derived.

b. What is reasonable assurance? Please give a specific statement that defines reasonable assurance.

c. What specific regulation requires that the referenced criticality calculation be published?

d. What is the definition of contact configuration? Explain why this configuration is the most important to analyze. Please cite specific reference or show calculation.

e. What would provide the "reasonable assurance" that a 12 element configuration would remain sub-critical?

f. What documents would petitioners consider an adequate "source" for a statement of experience?

g. Precisely what calculations and data would be required to satisfy the petitioners that the failure of a storage rack, fully loaded with twelve TRIGA fuel elements would present no safety hazard to either operational personnel or the general public?

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.

INTERROGATORY 10

a. What is an "experiment failure"? Please define specifically what occurs during an experiment failure, and what results are to be expected.

b. What type of experiment failure could occur that would release "radiation" to the environment?

c. What type of experiment failure could occur that would release "radiation" in excess of occupational limits?

d. At what points within the facility would these alleged releases in excess of occupational limits occur?

e. What activity levels would be required to exceed occupational limits within the facility?

f. What experiment failures at AFRRI would result in exceeding regulatory limits should there be a malfunction of confinement safeguards?

g. What activity levels are required at the location of an alleged experiment failure to exceed MPC for offsite releases?

h. What specific MPC levels (radionuclide and concentration) are the petitioners contending would be exceeded by the alleged failure of an experiment?

i. What specific safeguards do the petitioners feel could prevent the occurrence of an experiment failure coincident with a malfunction of a confinement safeguard?

SUCH MALFUNCTIONS INCLUDE:

a) A BREACH OF CONTAINMENT CAUSED BY MISSION RUBBER GASKET SEALING MATERIAL ON THE DOUBLE DOORS TO THE CORRIDOR BEHIND THE REACTOR CONTROL ROOM, IN VIOLATION OF APPLICANT'S TECHNICAL SPECIFICATION, I.A.4. (SEE, NOTICE OF VIOLATION, APP. A, NRC INSPECTION REPORT DOCKET NO. 50-170, 10/13/78);

INTERROGATORY 11

a. By what mechanism would radiation be released to the environment even if the rubber gaskets were totally removed from the double doors?

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.);

INTERROGATORY 12

a. In the event of a damper failure during an experiment, what activity levels would be required, and in what areas, to cause a release to the environment above MPC?

b. How was the failure cited above detected?

c. What additional safeguards do the petitioners feel are necessary to attenuate releases in the event of a failure of the automatic dampers to close after a CAM alarm?

d. How does the postulated concurrent failure of an experiment and damper closure lead to an environmental release?

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);

INTERROGATORY 13

a. How does the failure of the lead shield doors to stop at the fully open position result in the release of isotopes above MPC?

b. How does the failure of the lead shield doors to stop at the fully open position result in an experiment malfunction?

c. How does the failure of the lead shield doors to stop at the fully open position have any effect on anything related to personnel radiation exposure, offsite releases or confinement safeguards at AFRRI?

d. Please state how the lead shield door stop is part of a confinement safeguard or experiment failure.

e. However, even in light of Question c, what safeguards would satisfy petitioner that in the event of a lead shield door failure to stop at the full open position there would be no release of isotopes to the environment above MPC?

d) REACTOR CORE POSITION SAFETY INTERLOCK MALFUNCTION ON FEBRUARY 1, 1973 (NOT RECORDED IN DOCKET NO. 50-170).

INTERROGATORY 14

a. How does the failure of the core position interlock result in release of activity above MPC?

b. How does the core position safety interlock relate to a confinement safeguard or experiment failure?

c. What safeguards would satisfy the petitioners that should the reactor core position safety interlock malfunction there would be no release to the environment above MPC?

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:

INTERROGATORY 15

a. Have any of the cited failures resulted in releases to the environment? If so, please document when, where, and by what means these releases occurred. Also list the amounts of specific isotopes released.

b. Specifically, what human errors coupled with what series of safeguard failures could lead to a release of radioactivity in excess of regulatory limits?

c. Which specific regulatory limits could be exceeded by the alleged series of events in b. above?

d. Please give any design (engineered) safeguards not provided at AFRRI that the petitioners feel would further enhance the safety of the AFRRI-TRIGA reactor by decreasing the likelihood of an environmental release above regulatory limits.

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";

INTERROGATORY 16

a. How would the failure of Safety Channel One result in or cause the release of levels above MPC?

b. What would cause the simultaneous failure of redundant safety channel systems such that no safety channel would be operable and a release of activity would be possible?

c. Are such failures, multiple or single, at any TRIGA reactor documented to have caused releases to the environment in excess of 10 C.F.R. 20? If your answer is yes please cite specific references.

d. Cite the source for the statement "An NRC inspection on March 17, 1980, revealed that Safety Channel One would not initiate a scram."

e. Name the NRC inspector that discovered the alleged malfunction.

f. What additional safeguards would satisfy the petitioner that should Safety Channel One completely fail a release to the environment above the limits in 10 C.F.R. 20 could not occur?

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;

INTERROGATORY 17

a. List references for documents that show any releases of any radiation from any TRIGA reactor in excess of 10 C.F.R. 20 caused by an electrical fire in a motor control center.

b. Which faulty transformer caused the alleged power surge?

c. What could AFRRI do to prevent the power surge?

d. What additional safeguards do the petitioners feel could be placed in service such that failure of the reactor exhaust system (EF-1) would not result in the release of isotopes above MPC in the environment?

c) MALFUNCTION OF THE FUEL ELEMENT TEMPERATURE SENSING CIRCUIT CAUSED BY A "FLOATING SIGNAL GROUND," REPORTED BY DNA ON AUGUST 1, 1979;

INTERROGATORY 18

a. How would the malfunction of a fuel element temperature sensing circuit cause a radioactive release to the environment above MPC?

b. How was the malfunction detected?

c. What additional safeguards could AFRRI install to prevent a malfunction of this nature?

d. What additional safeguards could AFRRI implement to satisfy the petitioners that in the event of such failures, radioactivity above MPC would not be released to the environment?

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;

INTERROGATORY 19

a. If the pool level float switch fails during an experiment, how much activity do the petitioners postulate would be released to the environment?

b. How was the malfunction discovered?

c. What can be done to prevent such a malfunction? How would such actions or changes reduce the likelihood of releases in general or above MPC in particular?

d. Were there any reported radioactivity releases associated with this malfunction?

e. Please list and document any releases from any TRIGA reactor resulting from malfunctions of a water level sensing float switches.

f. What additional safeguards could AFRRI place in service to satisfy the petitioner that in the event of this type of malfunction radioactivity above MPC would not be released to the environment.

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);

INTERROGATORY 20

a. Please describe an experiment failure which, when coupled with the stated malfunction, would result in releases above regulatory limits. Show that such experiments are performed at AFRRI.

b. Should the experiment described by the response for Interrogatory 20a and the stated malfunction simultaneously occur, what prevents an operator on duty from manually closing the ventilation dampers?

c. Please list and document any releases from any TRIGA reactor that have resulted from a malfunction of this type.

d. What additional safeguards do the petitioners feel would alleviate any potential releases in the event of a malfunction of this type?

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;

INTERROGATORY 21

a. How does the malfunction of a fuel temperature automatic scram system result in the release of isotopes in excess of MPC to the environment?

b. Were any releases reported in the same malfunction report? In any report in Docket 50-170 associated with this type function, were any releases reported?

c. Please list and document any releases from any TRIGA reactor associated with this type of malfunction.

d. How was this malfunction detected?

e. What additional safety features could be incorporated into the automatic protection circuit to satisfy the petitioners that, should this malfunction occur again, no isotopes would be released to the environment in excess of 10 C.F.R. 20?

g) 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).

INTERROGATORY 22

a. How does the failure of the core position interlock result in the release of radioactivity above MPC?

b. Did this malfunction result in any release to the environment or were any releases reported in the same malfunction report?

c. Has a malfunction of this nature at any TRIGA reactor resulted in or contributed to a release to the environment above MPC. If your answer is yes please cite specific references.

d. What safeguards would satisfy the petitioners that, should the core position safety interlock system fail, there would be no core damage?

e. What safeguards would satisfy the petitioners that should the core position safety interlock system fail there would be no release to the environment above MPC?

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.

INTERROGATORY 23

a. By what mechanism does a "damaged" fuel element change the moderating effect of UZrH fuel? Please cite references.

b. What happens to the reactivity characteristics of a thermal reactor core if the moderating effect in the fuel is significantly reduced? Please cite specific references.

c. What constitutes the difference between a thermal reactor and a fast reactor?

d. By what mechanism is the negative temperature coefficient changed, and by how much is it changed? Cite supporting documentation.

e. What constitutes a damaged fuel element in the context of this contention?

f. How does the petitioner wish AFRRI to show that the reactor's negative temperature coefficient will automatically shut down the reactor during accident situations (i.e. which model, calculation, or experiment)?

g. How much of a reduction in the moderating effect would the petitioners expect from a single damaged fuel element? Would this be a positive or negative effect? Provide supporting documentation.

h. How many fuel elements would have to be damaged to reduce the moderating effect as postulated by the petitioners? Cite references or show calculations.

i. Please give specific references or calculations to show how a reduction in the moderating effect of hydrogen nuclei in a TRIGA fuel matrix will fail to decrease the average energy loss per average logarithmic energy decrement necessary to sustain power in a TRIGA reactor core.

j. What safeguards or data would satisfy the petitioners that the reactor's prompt negative temperature coefficient will always shut down (insert large quantities of negative reactivity) the reactor automatically?

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.

INTERROGATORY 24

a. What is the basis that the HSR or SAR does not consider multiple cladding failure accidents?

b. Have there been any prior multiple fuel element cladding failure accidents in any TRIGA reactor? Cite supporting documents.

c. How is an "uncontrolled power excursion" defined in a TRIGA reactor and what mechanism causes cladding failure for such a hypothetical uncontrolled power excursion in TRIGA fuel? Cite references.

d. How would a loss of coolant accident result in the AFRRI TRIGA reactor result in multiple cladding failures? Cite references or provide any calculations performed.

e. In the event of a LOCA what would be the power history required to breech the integrity of the fuel's cladding? Does the AFRRI TRIGA reactor have this hypothetical power history required to achieve these temperatures?

f. What postulated historical basis (operational) for a 1MW TRIGA reactor would meet the requirement for a cladding failure specified in e. above? Cite references.

g. How many elements should be considered for a multiple cladding failure? Provide the basis for your answer.

h. How many years of operation or how many kilowatt-hours of energy production are necessary to prove the integrity of the fuel elements contained in the TRIGA core?

i. To what specific "natural ("Act of God") accident" does petitioner refer in this contention?

EMERGENCY PLAN - THE EMERGENCY PLAN PREPARED BY APPLICANT IN CONJUNCTION WITH ITS LICENSE RENEWAL APPLICATION DOES NOT COMPLY WITH THE STANDARDS SET FORTH AT 10 C.F.R. PART 50, APPENDIX E.

INTERROGATORY 25

a. Specifically which section of 10 C.F.R. 50 requires that the information set forth in 10 C.F.R. PART 50, APPENDIX E for a research reactor be included in its emergency plan.

b. Please document why, in light of USNRC Regulation Guide 2.6, the AFRRI emergency plan is not adequate.

c. What are the qualifications of the petitioners to assess that the emergency plan fails to provide "reasonable assurance" that appropriate measures will be taken to protect the public health and safety in the event of offsite releases?

d. Please substantiate the statement made in Contention A.3, first paragraph, that the accidents described in ACCIDENTS I and II, SUPRA would cause a major offsite release and necessitate the implementation of the emergency plan. What constitutes a major offsite release in this context?

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 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.

INTERROGATORY 26

a. What specific waterborne radioactive emissions are generated by routine operations?

b. Which specific part of 10 C.F.R. 20 does the AFRRI reactor exceed with respect to the disposal of solid waste generated by routine operations?

c. What types of solid waste are generated by the reactor during routine operations? That is, which specific isotopes are produced and disposed of that would be beyond the limits of 10 C.F.R. 20?

d. What is a "probable violation?"

e. Which specific regulatory limits allegedly have been violated by AFRRI?

f. Which specific radiation monitoring methods are allegedly inadequate? Justify your answer.

g. How are "corrective actions" inadequate to detect and prevent violations of regulatory limits?

h. Which specific corrective action is inadequate to prevent which specific recurrence of which specific regulatory limit?

i. What are the petitioner's basis for stating that "probable violations" have occurred?

j. List all sources of data, information, or reasons they believe that the alleged "probable violations" have occurred, include names of persons or list specific documents and the dates that show these "probable violations" have occurred.

k. The contention above states that actual violations of 10 C.F.R. 20 have taken place at AFRRI. Please list specific violations including the isotope involved, the amount released, the date, the NRC inspector filing the

Notice of Violation, and the NRC report number. Do not include "probable violations" requested above.

l. What safeguards (equipment or procedures) would satisfy the petitioners that proper monitoring methods are in use at AFRRI for checking emissions from routine operations.

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.

INTERROGATORY 27

a. It is contended that AFRRI's equipment, methods and reporting systems do not provide "reasonable assurance" that violations of regulatory limits will be detected. Which specific equipment, methods, and reporting systems do not provide "reasonable assurance" that alleged violations have been or will be detected?

b. What specifically constitutes "reasonable assurance"?

c. From the petitioners' point of view, what type of equipment and methods for any offsite monitoring would be necessary to provide "reasonable assurance" that regulatory limits if exceeded would be detected? Please provide detailed characteristics of the proposed equipment and methods.

d. Which specific regulatory limits are the petitioners contending that AFRRI did or will exceed?

e. What is the basis for the contention that violations of regulatory limits have occurred and have not been detected? Justify any statements made.

f. List specific sources for the contention that regulatory limits have been violated and that such violation was not detected. Include names or specific documents and dates.

g. What equipment, methods, and reporting systems and safeguards would provide reasonable assurance to the petitioners that violations of any regulatory limits would be detected?

h. What isotopes, subsequently released into the Montgomery County Sanitary Sewerage System, are generated by reactor operations?

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 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.)

INTERROGATORY 28

a. What specific isotopes are to be detected by the environmental monitors?

b. What type of dosimetry would be adequate to detect radiation doses due to inhalation or ingestion? Please elaborate.

c. What Federal regulation requires a particulate monitor at AFRRI?

d. What references state that a gross activity detector is required to be isokinetic?

e. How does detector type determine isokinetic characteristics?

f. What evaluations are required of a gross activity detector?

g. What is the petitioners concept of the purpose and use of a particulate monitor for airborne effluents as used at AFRRI.

h. Why is the stack gas monitor not reliable for particulate sampling?

i. What is "meaningful evaluations?"

j. Cite specific sources to support the statement that the environment monitoring is inadequate at AFRRI.

k. What type of environmental monitoring and stack effluent monitoring equipment would be adequate, from the petitoners' point of view, to determine radiation doses to the public due to inhalation or ingestion? Please be specific and provide the minimum characteristics of such equipment.

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.

INTERROGATORY 29

a. What Federal regulation requires AFRRI to publish the methods of sample preparation and analysis?

b. What type of "raw data" should be published by AFRRI?

c. What Federal regulation requires the publication of this raw data?

d. What information would be necessary to be considered adequate?

e. What was the net change in the activity released to the environment as a result of changing the beta self absorption correction factor in the effluent analysis?

f. Specifically how could AFRRI change the currently in use methods and equipment such that the environmental monitoring would be deemed adequate by the petitioners?

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.

INTERROGATORY 30

a. What specifically is unrealistic about the concentric cylinder set model? Cite supportive references for any statements made.

b. What dose assessment method would the petitioners consider realistic? Explain fully and provide characteristics and justification for proposed method.

c. What model was used by the petitioners to determine that the effluents as projected by AFRRI were not within regulatory limits and that therefore the model used by AFRRI was unrealistic? Please cite the reference that insures that the model used by the petitioners is a realistic model?

d. Specifically which regulatory limits are of concern to the petitioners?

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 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 RELEASES IN EXCESS OF THE MAXIMUM PERMISSIBLE CONCENTRATIONS SET FORTH AT 10 C.F.R. PART 20, APPENDIX B.

INTERROGATORY 31

a. What is "highly probable"?

b. Which specific maximum permissible concentrations set forth at 10 C.F.R. 20, Appendix B is AFRRI supposed to have "highly probable" exceeded?

c. What method was used to determine that it was "highly probable" that AFRRI exceeded MPC in 10 C.F.R. 20, Appendix B?

APPENDIX "B" - CONTENTION 1

ACCIDENTS 1

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 PRODUCT RELEASE.

INTERROGATORY 32

a. What is the basis for the conclusion on the part of the petitioners that air convection cooling would not be sufficient to prevent a cladding failure in the event of a Loss of Shielding and Cooling Water (LOCA) accident? Please give references to support any statements made.

b. AFRRI studies indicate that even in the event of such accident conditions the integrity of the cladding will remain. Since the petitioners (CNRS) indicate this is incorrect, please give (considering the power history of the AFRRI core) the maximum internal pressure a single fuel element would experience for a loss of coolant event.

Is this pressure sufficient to rupture the cladding? Please give references or calculations to prove the above.

c. Do the petitioners have any documented evidence that any TRIGA reactor has had a cladding failure, caused by a LOCA? Have any resulted in any fission product release to the unrestricted area environment? If the answer is yes, please provide that document or reference it.

d. How significant a fission product release does CNRS Inc. expect from their postulated LOCA induced cladding failure? Please be specific as to amounts and isotopes released to both the reactor room and the environment. Please justify the answer with calculations and references.

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.

INTERROGATORY 33

a. Please describe in detail the events that could lead to a pulse causing a temperature elevation sufficient to cause multiple cladding failures during a LOCA. Show how these events occur with such timing to allow the alleged accident to occur. Please provide references to support any statements made.

b. Would the temperature reached during this alleged pulse be of such magnitude to cause cladding failure? Please show how this is possible to occur and justify with supporting documentation and references.

c. Which specific limits of 10 C.F.R. 20 are proposed would be exceeded by the alleged fission product release?

d. What is the temperature history during the alleged transient and accompanying LOCA? Provide supporting documentation.

BOTH OF THE DBA ANALYSES IN THE HSR ("FUEL ELEMENT DROP ACCIDENT" AND "FUEL ELEMENT CLAD FAILURE ACCIDENT") ERRONEOUSLY CONSIDER ONLY THOSE RADIATION DOSES TO HUMANS THAT WOULD RESULT FROM SUBMERSION EXPOSURE TO THE NOBLE GASES RELEASED.

PETITIONER CONTENDS THAT IF SUCH ACCIDENTS WERE TO OCCUR, INDIVIDUALS WOULD RECEIVE ADDITIONAL EXPOSURE DUE TO INTERNAL EMISSIONS OF THE NOBLE GASES, SUSTAINING INJURIES FAR GREATER THAN THOSE PREDICTED IN THE HSR.

INTERROGATORY 34

a. Please show how the dose to humans would be increased by considering internal emissions from the noble gases. Substantiate with references.

b. Specifically, how much additional exposure would an individual receive due to internal emissions resulting from the noble gases?

c. Please be specific in the statement that "individuals would receive additional exposure due to internal emission of the noble gases, sustaining injuries far greater than those predicted in the HSR." Exactly how much is "far greater"? Please show and justify.

d. Please explain why in light of the given thyroid dose (SAR Table 5) resulting from a described accident (Fuel Element Drop) the petitioners contend that only radiation doses from the noble gases were considered as possible accident consequences.

e. Please clarify why the petitioners consider the thyroid dose (Iodine) given in the SAR for accident conditions as submersion exposure to the noble gases.

f. What is the petitioners' basis for contradicting the Health Physics standards for exposure to noble gases? (For example, NCRP and ICRP guidelines)

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.

INTERROGATORY 35

a. What constitutes a "significant offsite release"?

b. Please be specific and show how significant the offsite releases would be. For example, which isotopes and how many micro-Curies would be released to an unrestricted area?

c. Please demonstrate how "such accidents" would result in "significant offsite" releases. Justify and cite any supporting documentation.

d. Please show specifically how the petitioners' alleged power excursion accident (PEA) can occur at an elevated temperature with a reduction in the thermalizing effect of hydrogen in the AFRRI TRIGA reactor.

e. What specifically does CNRS propose happens if the thermalizing effect of hydrogen in the AFRRI TRIGA reactor is reduced in a power excursion accident? Please justify and support any statements.

f. What are the necessary elements and conditions for an explosive Zirconium steam reaction? Please cite references to support the elements given.

g. Please show and justify that all elements and conditions necessary for an explosive Zirconium-steam reaction are present in the AFRRI TRIGA core. Please give references that support the elements and conditions stated as being present are in fact present.

h. Please give the necessary elements and conditions for an explosive Zirconium-air interaction. Please be specific and give references to support the elements given.

i. Please show and justify that all elements and conditions necessary for an explosive Zirconium-air interaction are present in the AFRRI TRIGA core. Please give references that support the elements and conditions stated as being present are in fact present. (This may include the loss of cooling water such that air is present.)

j. If the two maximum credible accidents given in 2) are, as stated, beyond the design basis of the reactor (class 9 accidents), how can the petitioners state that they can be expected to occur? Please be specific in justifying this apparent discrepancy.

ROUTINE EMISSIONS I

INTERROGATORY 36

a. Please explain section 3 of Routine Emissions I contention #5. Please be specific as to:

(1) Specifically, where in the Environmental Report or EIA data does it state that the highest average unrestricted area exposure rate from airborne releases extends to residential areas? Historically, what was the highest reported dose in a residential unrestricted area?

(2) Has any environmental monitor since 1970 given a reported dose level that even approaches 0.5 rem/year? If so, please specify which monitor location, which year?

(3) What bases were used to state that "it is highly probable that such exposures have resulted and continue to result in doses to the public in excess of .5 rem"? Please site specifics.

(4) What bases were used to question AFRRI's use and commitment to the ALARA principle concerning emissions from operations? Please site specifics.

b. Please cite specific monitoring locations that show all doses in excess of 10 C.F.R. 20 limits in any residential area resulting from operations under the reactor license.

c. Please specify the location of exposure rates in unrestricted areas discussed in #4 of contention 5. Please be more specific as to the dose rate of concern. Additionally, please specify the residential area eluded to that population of which would "highly probable" have received a dose in excess of 10 C.F.R. 20.

d. Since it is implied in section 4 of contention 5 that measurements were made only "a few times a year" of the dose rates quoted, it is assumed that the dose rates mentioned were arrived at from measurements other than environmental station monitors; is this correct? Please give any information you possess as to why and under what conditions these measurements were made. What is the source or cause of these dose rates? Were they due to environmental releases? Cite all references to support any statements made.

e. It is assumed that the .5 rem/hr is a typographical error; if not, please give the references that support the use of a number of this magnitude.

f. Were the dose rates cited in contention 5.B. section #4 attributed to reactor operations in the reports mentioned? Were the rates from environmental monitor locations?

g. Please explain how the question and answer (#11 Autumn 1979) referred to in 5.B.3 which contains a statement of by-product material (fission products and decay products) contained in the AFRRI core has any bearing on section 3 of Contention 5.

h. Please list names and addresses of CNRS members who live within 600 feet of the AFRRI stack. Please give the address of any residence within 600 feet of the AFRRI stack.

CONTENTION 7 - SECURITY

NEITHER THE PHYSICAL SECURITY PLAN FOR THE FACILITY NOR APPLICANT'S HISTORY OF SECURITY VIOLATIONS AND SUBSTANDARD MANAGEMENT AND OPERATION PROCEDURES DEMONSTRATE THAT THE CONTROLLED ACCESS AREAS CAN BE PROTECTED FROM SABOTAGE OR DIVERSION OF SPECIAL NUCLEAR MATERIAL ACCORDING TO THE STANDARDS SET FORTH AT 10 C.F.R. PART 73.

THE DRAFT AUDIT REPORT OF THE AFRRI FACILITY PREPARED BY THE DEFENSE AUDIT SERVICE IN 1979 CITES FREQUENT INSTANCES OF SECURITY AND MANAGEMENT VIOLATIONS, INCLUDING:

1) EIGHTEEN ACTIVATIONS OF THE FACILITY ALARM SYSTEM DURING A 34-DAY PERIOD, CAUSED BY PERSONNEL LEAVING WORK 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;

2) UNAUTHORIZED PEOPLE ENTERING THE FACILITY BY FOLLOWING EMPLOYEES IN WHO USED THEIR MAGNETIC CARDS TO UNLOCK THE DOOR;

3) FAILURE TO ESCORT VISITORS ATTENDING WEEKLY SEMINARS AND PROVIDE THEM WITH DOSIMETERS;

4) FAILURE OF EMPLOYEES ENTERING AND EXITING THE BUILDING AFTER HOURS TO SIGN A LOG SHOWING THEIR TIME OF ARRIVAL AND DEPARTURE.

5) VIOLATIONS OF APPLICANT'S ACCOUNTING AND DISPENSING PROCEDURES FOR CONTROLLED SUBSTANCES SUCH AS NARCOTICS.

INTERROGATORY 37

a. Please document the statement that standard operating procedures fail to protect the CAA from sabotage or diversion of special nuclear materials.

b. Of the eighteen cited activations of the facility alarm system, how many involved the reactor controlled access areas?

c. Please provide documentation that any unauthorized people entering AFRRI also entered the reactor controlled access areas.

d. Please explain how any of the cited instances of security and management violations reflected on the integrity of the controlled access areas.

Submitted by

Robert L. Brittigan

ROBERT L. BRITTIGAN
Counsel for Licensee

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 September 1981 FILING

I hereby certify that true and correct copies of the foregoing "LICENSEE'S FIRST INTERROGATORIES TO THE CITIZENS FOR NUCLEAR REACTOR SAFETY, INC." were mailed this 30th day of September, 1981, by United States Mail, First Class, to the following:

Louis J. Carter, Esq., Chairman
Administrative Judge
Atomic Safety and Licensing Board
23 Wiltshire Road
Philadelphia, PA 19151

Mr. Ernest E. Hill
Administrative Judge
Lawrence Livermore Laboratory
University of California
P.O. Box 808, L-123
Livermore, CA 94550

Dr. David R. Schink
Administrative Judge
Department of Oceanography
Texas A&M University
College Station, TX 77840

Mr. Richard G. Bachmann, Esq.
Counsel for NRC Staff
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Elizabeth B. Entwisle, Esq.
8118 Hartford Avenue
Silver Spring, MD 20910

Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Atomic Safety and Licensing Appeal Panel (5)
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Secretary (21)
U.S. Nuclear Regulatory Commission
ATTN: Chief, Docketing and Service Section
Washington, D.C. 20555

Robert L. Brittigan

ROBERT L. BRITTIGAN
Counsel for Licensee

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 Reactor)	:	

INTERVENOR CNRS's RESPONSE TO

LICENSEE's FIRST SET OF INTERROGATORIES

NOW COMES the Intervenor in the above-captioned case and pursuant to 10 C.F.R. §2.740b, responds to the Licensee's first set of Interrogatories as follows:

INTERROGATORY 1

Answered by Entwisle.

The Intervenor objects to this question. The only relevance it has to this proceeding is whether the Intervenor has legal standing. This issue has already been resolved, and the names and addresses of members were given in the affidavits that were submitted to establish the Intervenor's standing.

INTERROGATORY 2

Answered by Entwisle.

Elizabeth B. Entwisle, Esq.
8118 Hartford Avenue
Silver Spring, MD 20910

Irving M. Stillman, M.D., Ph.D.
5480 Wisconsin Avenue
Chevy Chase, MD 20815

INTERROGATORY 3

Answered by Entwisle.

None.

INTERROGATORY 4

Answered by Entwisle.

Entwisle is an attorney and co-author of a study prepared for the President's Council on Environmental Quality, "NRC's Analysis of Nuclear Accidents: Is it Adequate?" March 1980.

EXHIBIT B

8201260017

Stillman is a physician and physicist who has participated in the Three Mile Island proceedings before the NRC and whose advanced interdisciplinary training in medicine and physics qualifies him to speak about the biological impact of radiation associated with the operation of the Licensee's reactor.

The contentions are based on examination of the Licensee's documents, such as the Draft and Final Audit Reports, and of documents in the Licensee's docket in the Public Documents Room of the NRC, 1717 H Street, N.W., Washington, D.C.

INTERROGATORY 5

Answered by Entwisle.

Same as Interrogatory 4.

INTERROGATORY 6

Answered by Entwisle.

As of the present, the Intervenor has not determined which, if any, expert witnesses will be called to testify on any contention.

INTERROGATORY 7

Answered by Stillman. Unless designated otherwise, every Interrogatory hereinafter is answered by Stillman.

a. The Licensee describes two DBAs involving clad failures in their Safety Analysis Report (SAR), namely: a "Fuel Element Drop Accident" and "Fuel Element Cladding Failure Accident." In the Drop Accident the fuel element "has been allowed to decay after being taken out of the operating core and placed in storage. The fission products released from the gap will depend upon the temperature of the fuel following two weeks delay. This temperature is expected to be less than 50°C." (See quote in SAR, pp. 6-16.)

For the Cladding Failure Accident they postulate a gap activity of only 1.4 percent (of the total radioactive inventory in the fuel element) and a maximal release of only 0.2 percent of the iodines. Since the temperature needed to volatilize iodine is 183°C (see SAR, pp. 6-16), it follows that this DBA is presumed to occur at a temperature far below 180°C in order to meet their own criteria for maximal iodine release and total gap activity.

b. Throughout the Hazard Summary Report (HSR), peak fuel temperatures above 600°C are never acknowledged. Furthermore, the selected (fuel element) gap activities and potential radioactive gaseous releases are only realistically compatible with much lower temperatures. For example, the very low values for radioiodines that would be released in a cladding failure accident. For the record, the Intervenor believes that during an inadvertent transient the peak fuel temperature could rise several hundred degrees.

c. The Intervenor makes the conservative assumption that the cladding temperature will essentially mirror the fuel temperature, i.e., an adiabatic transfer of heat between them. Since the Adiabatic Model for heat transfer is given in any elementary thermodynamics text it is unnecessary to cite specific references.

d. During an inadvertent transient, the fuel temperature can rise several hundred degrees depending on the exact core conditions (e.g., specific fuel element configuration, position of the control rods, mechanical malfunctions, operator errors, preceding power history, experiments in progress, loss of coolant, placement of the core within the reactor pool).

e. An "inadvertent transient" or power excursion occurs when there is a sudden core insertion of excess reactivity above cold critical to produce a <u>large rapid increase in neutron flux</u>.

1. According to the Licensee's "technical specifications" the maximum step insertion above critical can be as much as 2.8 percent $\Delta k/k$ reactivity in the pulse mode without any potential danger.

2. If the AFRRI-TRIGA reactor is functioning within the permitted specifications, the maximum reactivity transient that could possibly occur (according to the Licensee) would be that produced by the rapid insertion of the entire available amount of reactivity, namely, 3.5 percent $\Delta k/k$ ($5.00) excess reactivity above cold critical (with or without all experiments in place). The maximum power level associated with such a transient is $\leq$ 10,000 MW. The Licensee maintains that based on the operating experience of the Advanced TRIGA Prototype Reactor (ATPR) in the General Atomic laboratories and calculations using the Fuehs-Nordheim mathematical model, "it can be concluded that the rapid insertion of the total excess reactivity of 3.5 percent $\Delta k/k$ would not represent an undue risk."

3. There are several ways in which an "inadvertent transient" could be initiated and trigger a Power Excursion Accident (PEA), such as:

(1) Improper fuel loading - a reactor operator inadvertently inserts a fuel element into the reactor core when it is already critical.

(2) Failure of an experiment - resulting in an instantaneous insertion of excess reactivity (i.e., the radioactivity associated with the experiment itself) to produce a dangerous transient.

(3) A stuck transient rod - if the most reactive control rod (i.e., the transient rod) is stuck out

of the reactor when the core is already loaded to its total excess reactivity.

(4) Pulsing with the transient rod greater than \$3.00 (2.1% $\Delta k/k$) reactivity - after withdrawal of the three standard control rods (previously withdrawn to achieve a steady state power greater than 1 MW).

4. An "inadvertent transient" requires, by definition, an unplanned error or malfunction in the operation of the TRIGA reactor. Such human errors or equipment failures are very often multiplied during the course of any reactor accident. The history of nuclear reactor accidents, in general, is literally replete with examples of a single malfunction or human error compounded by a series of errors and additional malfunctions. Such a set of circumstances could prevent the safeguards that normally control a "planned transient" from functioning properly, i.e., "within the permitted specifications."

f. During a pulse operation that results in a PEA with cladding failures, the Intervenor postulates that both the fuel-moderator matrix and the claddings will have reached temperatures of 900°C or more. In spite of repeated assurances by the Licensee that the built-in and natural safeguards of the AFRRI-TRIGA would prevent fuel temperatures from rising to and above the safety limit, 1000°C, we contend that such safeguards are not foolproof (see our Interrogatory Answers to question 8a, part 3) and further that there must be circumstances under which such temperature elevations are possible. To document this contention the Licensing Board is referred to

1. The AFRRI Hazards Analysis reviewed by the Test and Power Reactor Safety Branch Division of Licensing and Regulation, Docket No. 50-163, p. 3, 1963, which states that if the three standard control rods are withdrawn to obtain a steady state power of 2 MW, then pulsing with the transient rod of \$3.00 (2.1% $\Delta k/k$) reactivity could raise the peak fuel temperature "to about 900°C due to the temperature at the steady state compounded with the temperature increase from pulsing." Clearly, then, pulsing with a transient rod of more than \$3.00 reactivity could raise the peak fuel temperature to well above 1000°C.

2. Calculations have been made to determine the temperature rise in a central TRIGA fuel element if the cooling water is lost instantaneously (see the 1963 GA-2025 Hazards Summary Report for the 250 KW Mark II TRIGA Reactor [Pulsing] located at the Columbia University in New York City). These calculations clearly demonstrate that a LOCA (in this tank-type TRIGA reactor) can result in fuel element temperatures up to 1200°C.

3. The many experiments routinely performed during the last twenty years at the General Atomic Laboratories

in California with TRIGA fuel elements in which temperatures of 1000°C or more were rather easily attained (regardless of negative temperature coefficients).

g. Experiments have been performed on hydrided 10 wt% U-Zr fuel elements that were rapidly heated by induction. "Results indicated that within about 75 sec the surface temperature reached 930° to 970°C with only minor hydrogen evolution. Abruptly thereafter, the surface was observed to crack parallel with the cylindrical axis, with strong outgassing rates" (see "The $U\text{-}ZrH_x$ Alloy: Its Properties and Use in TRIGA Fuel," M. R. Simnad, pp. 2-18). Another reference is H. H. Hausner and J. T. Schumar ("Nuclear Fuel Elements," p. 84) where surface cracks appeared in fuel element claddings when they were overheated to 900°C or more. In addition to these specific references, a clear general mechanism is present for concluding that cladding failures are "much more likely" at fuel temperatures greater than 400°C, namely, at elevated temperatures there would be a corresponding increase in the total gap pressure (produced by the rise in fission gas pressure, residual air pressure, and the peak equilibrium hydrogen pressure) that would put the cladding under much greater stress.

h. It is not possible to quantitatively assess the risk of a cladding failure at any fuel temperature. If the Licensee knows some exact way of determining such risks (without knowing the actual probability for each component event) then they should share that knowledge with the rest of the world. Temperatures of operating fuel elements well above 400°C may easily be achieved through pulse heating (see R. E. Taylor's "Pulse Heating of Modified Zr-H," U.S. AEC Report NAA-SR-7736, North American Aviation, 1962). Another scenario for fuel temperature elevation is described under the LOCA-induced multiple cladding accident scenario (see Answer to Interrogatory 24, parts d, e, f).

i. Cladding failures are more likely the higher the fuel element temperature. They are less likely at temperatures below 800°C and become much more possible at temperatures of 900°C or more. For references, see Interrogatory answers to both 7.g. (given above) and 24.c. (given below).

j. Repeated activity in the pulse mode may result in pulse beating. If the peak fuel temperature stays below 550°C the emission of radioactive gases is largely controlled by recoil effects which are not very sensitive to the fuel temperature. However, should the pulse beating result in temperatures above 600°C, the process of gas emission into the gap becomes mostly diffusion controlled and results in "greater gap activity." By contrast with the recoil mechanism, diffusional gas emission is extremely temperature sensitive so that gap activity rises rapidly as a function of increasing fuel temperature.

k. "Greater gap activity" implies larger partial pressures of the radioactive gases contained within the gap. If there is no cladding failure and the cladding maintains perfect structural integrity, then "greater gap activity" will not, of itself, result in a greater fission product release. However, any structural disruption resulting in cladding degradation would result in "greater fission product release" if there were "greater gap activity." Thus cracks or penetrations in the cladding would permit the radioactive gases in the gap to stream out under pressure and if the total gap pressures become great enough to approach 1,800 psi, then these excessive pressures would cause additional breaks in the cladding permitting more rapid release of the gap activity. Total gap pressures in excess of 1,800 psi could even cause complete rupture of the cladding without any prior deterioration.

l. Below 400°C the possibility of a cladding failure is relatively independent of the fuel temperature (e.g., the temperature-independent recoil mechanism is operating at temperatures of 400°C or less).

m. 1,200°C. A core history of at least 100 hours of 1 MW steady state operation.

n. A cladding failure is more apt to occur at fuel temperatures above 1,000°C and total gap pressures in excess of 1,800 psi. The greater the temperature and gap pressure, the more likely the cladding failure.

INTERROGATORY 8

a. Accidents that might occur other than those described in the AFRRI-HSR include

1. Fuel element storage rack failures. Because of their very high radiation levels, spent fuel elements are stored under water for shielding purposes. They are therefore stored in aluminum racks under the pool of water in the reactor tank at the AFRRI facility. The Licensee states that "experience shows it requires approximately 67 fuel elements, of the design used at AFRRI, in a close packed array to achieve criticality." The Intervenor would like to know the exact nature of the "experience" that demonstrates the requirement for "approximately 67 fuel elements" to achieve criticality, given the fact that unlike ordinary fuel elements (in most power reactors) that contain only 3% enriched Uranium - 235, the TRIGA fuel elements contain nearly 20% enriched Uranium-235. Furthermore, even by this overly conservative estimate, the AFRRI could conceivably accumulate this number of fuel elements in their reactor pool over the next 20 years if on-site storage (versus Away From Reactor) remains the guiding principle for the handling of high level radioactive waste. There is also the possibility that unforeseen conditions may require the rapid discharge of the full core load of fuel (i.e., 85 fuel elements). What plans has the AFRRI made to handle such an emergency? Would it really be safe to store this number of fuel elements in their aluminum racks inside the reactor tank?

2. Failure of an Experiment resulting in a significant release of radioactive material. This can result from a runaway experiment undergoing activation either within the reactor core (such as the CET) or in the exposure rooms. There have been at least two documented malfunctions that could effect the release of significant amounts of radiation (from a runaway experiment) into the reactor room, namely:

(i) a safety interlock malfunction that occurred on February 1, 1973;

(ii) malfunction of the lead door rotation on July 27, 1976.

Technical Specifications, section I.A.4 states "The reactor room shall be designed to restrict air leakage when the positive sealing dampers are closed." To accomplish such containment "the door to the corridor behind the reactor control room . . . is a double door that is sealed with compressible rubber gaskets and latched. The double doors at the opposite end of the corridor . . . is also sealed with a compressible gasket." Contrary to this specification, as of October 13, 1978, the above doors were not maintained as designed, in that gasket material was missing on both doors preventing fulfillment of the design function (see Notice of Violation, Appendix A, NRC Inspection Report Docket No. 50-170, 10/13/78). Hence any significant radiation release into the reactor room resulting from an experiment failure could have leaked out of the reactor room into the rest of the AFRRI facility. Another example of this breach of reactor room containment occurred on August 26, 1975, due to failure of the Reactor Room ventilation dampers to close when the Continuous Air Monitor (CAM) was alarmed. Both of these examples plainly demonstrate that any radiation released into the reactor room has a distinct possibility of leaking out into the entire AFRRI building and even outside the building into the environment (given a large enough radiation release) thereby endangering the public health and safety (see the AFRRI Abnormal Occurrence Report to the Directorate of Reactor Licensing, dated 9/3/75).

3. Failure of one or more of the "built-in safeguards," such as the:

(i) safety system channels
(ii) safety system settings
(iii) radiation monitoring systems
(iv) "negative temperature coefficient of reactivity" mechanism for automatic shutdown.

The Licensee describes several built-in safeguards (listed above) that would either alert the reactor operator or some automatic mechanism to effect the necessary corrective measures (e.g., initiate a reactor SCRAM, engage the appropriate interlock system, add water coolant, close the ventilation system, etc.) should an accident even threaten. However, the Intervenor contends that human errors coupled with equipment failures can render these safeguards ineffective, as has occurred repeatedly in nuclear reactor accidents nationwide, such as:

(i) the jumpered safety interlocks of Vermont Yankee;

(ii) the Dresden 2 blowdown in 1971;

(iii) the Millstone seawater intrusion in 1972;

(iv) the Brown's Ferry fire in 1975;

(v) the inversion of control rods and the Rancho Seco control rod drive failure in 1975;

(vi) the relief valve that malfunctioned and stuck open during the accident sequence at Three-Mile-Island in 1979.

To demonstrate that such failures (both human and mechanical) can also occur with the TRIGA reactor at the AFRRI, we shall cite several instances of relevant malfunctions involving these safeguards (reported by the AFRRI to the appropriate federal regulatory agency), including:

(a) On February 1, 1973 the Reactor Core Position Safety Interlock System that coordinates the lead door/core movement (to bring the door into near contact with the core shroud) malfunctioned due to a faulty de-energizing relay.

(b) On January 29, 1974 the Fuel Temperature-Automatic Scram System malfunctioned "due to the build-up of high resistance material on the mechanical contacts of the T2 output meter that initiated the automatic scram through a relay."

(c) On August 26, 1975 the Radiation Monitoring System malfunctioned, i.e., the reactor room ventilation dampers failed to close when the Continuous Air Monitor sensing device was manually triggered. "Inspection revealed that two wires in the control box were loose" and that this was the apparent cause of the malfunction.

(d) On July 10, 1979 there was a malfunction of the Pool Water Level Sensing Float Switch that monitors the reactor pool water level in case of an impending LOCA. "The malfunction was caused by wear on the jacketing around the wires leading to the switch which provided a path to ground, thereby circumventing the switch function."

(e) On July 30, 1979 there was a malfunction of the fuel temperature indicators (i.e., fuel element temperature sensing circuit) ostensibly caused by a "floating signal ground with respect to the system ground." The Licensee admits that "since this system monitors the principal safety parameter of the reactor, it was felt that a more secure ground was required."

(f) On August 9, 1979 the reactor exhaust system malfunctioned due to an electrical fire (in the EF-1 cubicle of the motor control center) caused by a power surge due to a faulty transformer.

(g) On March 15, 1980 there was a malfunction of Safety Channel One such that most of the scram indicators on the reactor control console were illuminated. Further, an inspection on March 17, 1980 "revealed that Safety Channel One would not initiate a scram in accordance with the Technical Specifications of Reactor License R-84. "The cause of the malfunction" was attributed to a damaged operational amplifier on a Safety Channel One circuit board "when electrical power had been reapplied to the console after a power outage."

The Licensee alleges that even if there is a power excursion in the AFRRI-TRIGA and the built-in safeguards malfunction, the reactor will automatically shut down due to the prompt negative temperature coefficient (i.e., -0.126% $\Delta k/k$ decrease per 1°C rise in fuel temperature). This automatic shutdown is entirely dependent on the relative amount and energetic state of the hydrogen nuclei within the U-ZrH_x alloy. Thus, any significant deviation of the hydrogen parameters from their expected values or curves will cause a drastic change in either the prompt and steady-state negative temperature coefficients. Such deviation of the hydrogen parameters are likely under accident conditions where large internal pressures and elevated temperatures may produce phase changes within the U-ZrH_x alloy (see a discussion of the "hydride phases" in "The U-ZrH_x Alloy" by M. T. Simnad, February 1980). Such phase changes will affect the vibration frequency, , of the hydrogen nuclei and thereby seriously alter the negative temperature coefficients of reactivity which depend on the transfer of energy quanta (of magnitude h) from warm or excited hydrogen nuclei to slow or thermal neutrons (via elastic collisions).

4. Multiple cladding failure accidents.

Such accidents may result from any one or more of the following:

(i) Defects in the material integrity of the fuel elements themselves.
(ii) Uncontrolled power excursion (or inadvertent transient) in the operating reactor core (PEA).
(iii) Sabotage or a natural accident (e.g., "act of God") involving the AFRRI-TRIGA Reactor.

The Intervenor contends that cladding failures may occur during operation of the TRIGA reactor secondary to inherent defects or weaknesses in the material integrity of the fuel elements themselves. These may go unnoticed in the required annual fuel element inspections by the Licensee. Note, the frequency of these inspections (for the AFRRI) was decreased from six to every twelve months (in May 1972). Whereas three cladding failures have already been reported by General Atomic in their Torrey Pines TRIGA reactor (see the AFRRI Safety Analysis Report), we contend that there is little reason to believe that they cannot happen in Bethesda, Maryland. In fact, there apparently have been at least two reported cladding failures at the AFRRI itself. The first was on August 17, 1964 whereupon a telegram was dispatched to the AEC in which "the institute has determined that at least one TRIGA type pulsing fuel element exhibits cladding failure." The second occurred on October 19, 1967 in which "small amounts of gas bubbles were observed to be released from a C-ring fuel element." The report (to the AEC on October 24, 1967) goes on to say that "the leaking fuel element was taken out . . . and transferred to the fuel element storage rack in the pool." Clearly there have been and will continue to be cladding failures within the core of the operating AFRRI-TRIGA reactor.

The several ways in which PEAs can be initiated was described above (see answer to Interrogatory 7.e., part 3). The common factor in all of these initiating incidents is that there is a sudden insertion of excess reactivity (within an already critical reactor core) to produce a rapid, large increase in neutron flux (i.e., a prompt power excursion or transient) capable of causing cladding failures at elevated fuel element temperatures. As noted above, if the AFRRI-TRIGA reactor is functioning "within the permitted specifications," the maximum reactivity transient that could occur would be that produced by the rapid insertion of the entire available amount of reactivity, namely 3.5% $\Delta k/k$. Based on the operating experience of General Atomic's ATPR and calculations rooted in the Fuchs-Nordheim mathematical model, the Licensee concluded that "the rapid insertion of the total excess reactivity of 3.5% $\Delta k/k$ would not represent on undue risk." This confidence, however, may be misplaced since the ATPR is certainly not identical with the AFRRI-TRIGA and is under considerably more expert scrutiny and experimental control by the General Atomic scientists. Consider the past history of mechanical malfunctions within the core of the AFRRI-TRIGA reactor itself, including:

(i) cladding-damaged fuel elements on August 17, 1964 and October 19, 1967;

(ii) separation of the transient rod from its connecting rod discovered on July 17, 1973;

(iii) crack detected in the top weld of the transient control rod on May 1, 1974;

(iv) detection of a tilted fuel element within the reactor core, reported January 31, 1975;

(v) misalignment of two fuel assemblies occurred on August 22, 1978.

Consequently, one cannot presume that the AFRRI-TRIGA will always function "within the permitted specifications" and therefore is subject to a serious Power Excursion Accident (PEA) involving one or more cladding failures.

The potential for sabotage or terrorist activity was dramatically pointed out in an April 3, 1979 Draft Audit Report by the Defense Audit Service (DAS), charging that frequent security and safety violations were being committed at the AFRRI. Specifically, the draft audit states that although "NRC's inspections have generally shown that AFRRI's security and safety operations have been satisfactory . . . our review showed that frequent safety and security violations were being committed." Even Admiral Robert Monroe, former Director of the Defense Nuclear Agency (DNA), admitted (see The Washington Star, August 14, 1979) that the possibility for sabotage is real when he said: "If a group of heavily armed, desperate men stormed into the building, there'd be nothing out there to stop them." Clearly, any serious explosion within the reactor room that permitted release of the radioactive inventory into the Facility and beyond, would seriously threaten public health and safety.

As for the possibility of an accident, consider an airplane crash into the AFRRI Facility. There are two major airports (National and Dulles) within 15 miles of the National Naval Medical Center producing extremely heavy air traffic above Bethesda, which is more than five times what is considered safe (from plane crashes) for any nuclear site according to the American National Standards Institute. In addition, a helicopter pad, located on-site at the medical center, is less than one-third of a mile from the nuclear reactor. Other types of accidents are also quite possible. For example, less than a thousand yards from the AFRRI Facility a new Metro subway station and tunnel are being constructed. The Intervenor warns that not only the drilling and dynamite explosions during construction, but future train accidents, might result in conditions that predispose the AFRRI-TRIGA to a LOCA resulting from either rupture of the reactor tank itself, damage to the AFRRI cooling tower, or damage to any part of the pumping system. Such a construction or train accident could also affect one or more of the safeguard systems (listed above) thereby potentiating a dangerous PEA.

5. Maximum Credible Accident (class 9 accident) resulting from an

(i) explosive zirconium-steam (water) interaction (at fuel temperature $\geq$1,000°C) following a PEA-induced multiple cladding failure (without a LOCA), or

(ii) explosive zirconium-oxygen (air) interaction (at fuel temperature $\geq$1,000°C) following a LOCA-induced multiple cladding failure.

The several ways in which a PEA could be initiated and lead to multiple cladding failures were outlined above (see answers to Interrogatory 7.e.3. and 24.c.). As indicated, in those multiple cladding failure accidents due to uncontrolled, prompt, power excursions there is likely to be an associated elevation of the fuel-moderator temperatures (within the damaged fuel elements) to 900°C or more. In particular, those fuel elements reaching temperatures above 1,000°C might produce total gap pressures (≥1,800 psi) capable of rupturing their already damaged stainless steel claddings. Rupture of a fuel element cladding would expose hot $Zr\ H_x$ (the major component of the fuel element) to the tank water. An NRC report indicates that the rate of a violent zirconium-water (or steam) reaction becomes significant at about 900°C (see NRC memorandum to Roger Mattson from R. O. Meyer, dated April 14, 1979, "Core Damage Assessment for the TMI-2," p. 25). For the strongly exothermic reaction of zirconium with steam approximately 140 k cal per g-mole of zirconium is released at 1,000°C. Each fuel element contains nearly 2 kg of zirconium-hydride, hence a pressure explosion within the ruptured fuel elements would essentially strip these elements and release their entire radioactive inventory into the reactor room. This would also lead to a series of chain-like explosions from additional zirconium-steam interactions as well as other chemical explosions (e.g., from ignition of the hot hydrogen chemically reacting with the oxygen in the reactor room) that would ultimately release hundreds of thousands of curies of mixed fission and activation products into the unprotected atmosphere. Unprotected because the AFRRI-TRIGA reactor (unlike a power nuclear reactor) is not enclosed in any reinforced containment dome. Since there are a few hundred pounds of zirconium-hydride within the core of the TRIGA reactor, which is explosively equivalent to almost half a ton of gunpowder when it reacts with water or steam (at temperatures ≥1,000°C), an explosion within the reactor room would disperse radioactive material over a very densely populated area of many square miles.

Perhaps the most serious credible accident that might befall the AFRRI-TRIGA Reactor would begin with a LOCA. The water coolant could swiftly leave through an open water line, rupture the reactor tank and aluminum tank liner, or be pumped out of the reactor pool. Any of these could be initiated by sabotage, inadvertent accident, mechanical malfunction, or human error either individually or in combination (as was noted in authoritative reports of the infamous Three Mile Island Accident). If the water leaves rapidly (approximately 250 gallons per minute) then the fuel element temperature would rise suddenly (see the Hazards Summary Report, GA-2025, 1963, for the 250 kW Mark III TRIGA Reactor at Columbia University). As noted above (see H. H. Hausner and J. F. Schumar in "Nuclear Fuel Elements," p. 84) a sharp temperature fluctuation of this nature is apt to induce multiple cladding failures. Calculations appearing in the Hazards Summary Report prepared by General Atomic scientists for the Columbia-TRIGA Reactor (a smaller but otherwise similar tank-type of nuclear reactor), show the maximum fuel element temperature resulting from a LOCA might reach 1,200°C. Such peak fuel temperatures (i.e., ≥1,000°C) could produce excessive total gap pressures (i.e., ≥1,800 psi) sufficient to rupture several of the already damaged stainless steel claddings. Under these conditions, rupture of the claddings would expose the $Zr\text{-}H_x$ to air (or oxygen) at temperatures of or above the 1,000°C safety limit. According to Professor Earl A. Gulbransen the chemical reaction between $ZrU_{0.034}H_x$ with air is even more violently exothemic than the zirconium-water reaction, releasing more than 260 k cal per g-mole of zirconium at 1,000°C. Furthermore, once started, he claims there is no easy way to stop the explosive reaction. In a certain sense, the explosive mechanism becomes auto-catalytic in that a single explosion would rupture more fuel elements releasing additional zirconium and hydrogen which is then available for further explosive chemical interactions with the oxygen in the air. A series of such core explosions would result in dispersing the radiation inventory of the entire AFRRI nuclear facility over a very large area in and around Montgomery County. The public living within the 5-mile ingestion zone (more than 100,000 people) would, in effect, be showered with such radionuclides as Uranium-235, Strontium-89, Iodine-131, Cerium-144, Cesium-134, Yttrium-91, Krypton-85, Strontium-90, and Cesium-137; all with considerable activities and prolonged half-lives.

b. The Intervenor takes great exception to the Licensee's broad allegation that "accidents ranging from failure of experiments to the largest core damage and fission product release considered possible, result in doses of only a small fraction of 10 CFR part 100 guidelines and are considered negligible with respect to the environment." In fact, all of the accidents described above (in Section 8.a. of these Interrogatories) could violate those guidelines and especially the last two scenarios (involving zirconium explosions) would absolutely result in radioactive releases far in excess of the 10 CFR 100 guidelines. Since the HSR and SAR admit to only minimal population exposure (i.e., "doses of only a small fraction of 10 CFR part 100 guidelines") all of the acciden s described by the Intervenor should be considered of "greater severity."

c. In order to quantitatively evaluate the risk of any reactor accident you must know the specific probability for each item in the postulated event-tree as well as the reliability of the subsystems involved. Without an adequate data base, calculation of the probability for each component event is virtually impossible. Similarly, there is a lack of reliability data on many of the essential subsystems. Unfortunately, this type of data is not yet available and even if it were, there are associated theoretical controversies that still plague interested scientists and mathematicians, the infamous Rasmussen Report being a case in point. If only we could quantitatively evaluate the risk of accidents this intervention would probably be unnecessary, for then no one would dare put a nuclear reactor in Bethesda.

d. No one person is really qualified to "properly designate" an accident as a DBA for the AFRRI-TRIGA reactor. It would take a team of qualified experts representing several scientific disciplines including nuclear physics, nuclear engineering, materials science, chemical physics, radiation medicine and health physics. These experts should all be Ph.D.s or M.D.s or both. When dealing with a public danger of this magnitude, it behooves us to use the very best talent we can muster to evaluate the true DBAs - whether it be for the AFRRI-TRIGA or any other nuclear reactor.

e. The Federal Guidelines as they presently exist. It would be advisable to include both the EPA as well as the NRC guidelines.

f. The two maximum credible accidents described in Interrogatory Answer 8.a.(5) above should be designated as DBAs for the AFRRI-TRIGA, because they are the two possible "worst case" accidents. The accidents described in the HSR and SAR are also possible but they are almost trivial compared with the magnitude of the two potential explosive zirconium accidents. For documentation, refer to the testimony presented by Professor Daniel M. Pisello at the Environmental Protection Committee of the New York City Council Hearing on the Hazards of Nuclear Power Plants, June 15, 1979.

INTERROGATORY 9

a. It is common knowledge that spent fuel elements from power reactors are stored in racks under water, because they are highly radioactive. If the racks used to store the elements should fail then enough fuel elements may come together at the bottom of the pool to reach criticality (i.e., produce a critical power excursion). It is a problem which currently concerns many nuclear scientists. At the AFRRI the spent fuel elements are stored 12 to a rack within the reactor pool itself and if the elements must be kept at the AFRRI Facility (because of no federally designated or available AFR disposal site) then the total number of spent fuel elements stored in the reactor pool may become a serious hazard. Unlike the fuel elements in power reactors which are only 3% U-235 enriched, the TRIGA fuel elements are nominally 20% U-235 enriched and are consequently a greater threat. It therefore becomes necessary to determine the maximum number of spent fuel elements that are safe to store assuming the optimum reactive geometrical array if they should come together at the bottom of the reactor pool. The Licensee assures us that if fewer than 67 fuel elements come together, nothing can happen. We simply want to know the "experience" and calculations on which this assurance is based. It also becomes very important should the need suddenly arise to dismantle and temporarily close the entire core of fuel elements, about 85. What provisions have been made for such an emergency?

b. Reasonable assurance could come from two sources, namely: (i) experiments in which TRIGA fuel elements are placed in the contact configuration under water and the actual reactivity measured with local power range monitors (to measure the local power distribution in the fuel element array);

(ii) criticality excursion calculations for the worst possible geometrical array - that can be evaluated by non-government and non-industry scientists.

c. No specific regulation. However, this is a shortcoming that must be corrected immediately since inadequate fuel element storage now looms large as a terribly significant problem for the entire nuclear industry.

d. The contact configuration represents the optimum reactive geometry, that is, the geometric array most likely to achieve a critical power excursion (i.e., criticality).

e. The values obtained for $k_{eff} \leq 0.746$ and $\dot{m}/m_{crit.} \leq 0.415$, are a "reasonable assurance" that a 12 element configuration would remain subcritical.

f. The Intervenor accepts the data represented in Figure 2 as adequately representing the experience of the Los Alamos Scientific Laboratory and the Oak Ridge National Laboratory in their experiments U-235 enriched fuel elements Memorandum for Record (January 19, 1981) submitted by the Licensee.

g. The Intervenor is satisfied by the data presented (in their Memorandum for Record) that failure of a storage rack, fully loaded with twelve TRIGA fuel elements would present no safety hazard to either operational personnel or the general public. However, if additional storage racks are contained in the reactor pool, they should be limited to two or three.

INTERROGATORY 10

a. An experiment fails when it either results in an instantaneous insertion of reactivity into the reactor core (type I), or there is a release of radioactive material from an experiment undergoing activation in the reactor (type II).

b. Either type I or II, but by an entirely different mechanism for each. If the type I failure resulted in a PEA (depending on the level of reactivity already operative within the core) and cladding failures (from overheating), then these sets of circumstances could lead to an escape of the radioactive gases from the damaged fuel elements into the reactor room and pool water. A type II failure results directly in the release of radioactive material that could also escape into the reactor room.

c. the same as described in Part 10.b. above.

d. Initially in the reactor room. However, if there is a breach of containment (as described in answers to Interrogatory 8.a.) then the radioactive gases could reach other areas of the AFRRI Facility depending on the nature of the containment breach.

e. Please refer to the Federal Regulations for the occupational limits on each radionuclide.

f. The same as described in part 10.b. above.

g. See Federal Regulations.

h. Depends on the specific experiment sanctioned by the AFRRI.

i. Make certain that the confinem nt safeguards are intact and functioning properly by more frequent, competent, and independent third-party inspections.

INTERROGATORY 11

a. If the rubber gaskets are totally removed then gaseous radionuclides can enter the ventilation system through adjacent rooms or even penetrate through these rooms to the entire facility.

INTERROGATORY 12

a.
b.
c.
d.

To answer these questions accurately, we would have to have more detailed information concerning the physical layout and operational history of the AFRRI Facility.

INTERROGATORIES 13, 14, 15, 16, 17, 18, 19, 20, 21, 22

These questions all refer to specific malfunctions and violations incurred by the AFRRI during their operation of the TRIGA reactor. It makes no sense for us, as outsiders, to second-guess information which is more readily available to them through their own documents or by their direct observation and measurement. The charges we have made are a matter of public record and are in full agreement with the designated regulations and technical specifications necessary to operate the TRIGA reactor safely. If the Licensee is serious about trying to remedy these situations by including our technical input, we recommend that they consult with us and the Union of Concerned Scientists on some formal basis.

INTERROGATORY 23

a. The moderating effect of the Zr H_x is largely mediated by the hydrogen nuclei. Experiments performed at the Brookhaven National Laboratory (on neutron thermalization by chemically bound hydrogen) gave results for Zr H_x compatible with a solid lattice of regular tetra hedra of zirconium atoms with the hydrogen atoms occupying sites at the center of each tetrahedron. The hydrogen lattice vibrations could be described by an Einstein model with a characteristic energy $h\nu \approx 0.130$ electron volts, where he is Planck's constant and ν is the hydrogen lattice vibration frequency (see A. W. McReynolds, M. Nelkin, M. N. Rosenbluth, and W. Whittemore, "Neutron Thermalization by Chemically Bound Hydrogen and Carbon," Proceedings of the Second U.N. International Conference on the Peaceful Uses of Atomic Energy, Geneva, September 1-13, 1958, Paper UN/P/1540). The moderating effect of the hydrogen nuclei may be achieved by elastic collisions with fast or slow neutrons;

that is, prompt or fast neutrons can be slowed down or thermalized by giving up a quantum of their energy, $h\nu$, to the sluggish (or cool) hydrogen nuclei, or slow neutrons may be speeded up by receiving the quantum of energy, $h\nu$, from the energetic (or warm) hydrogen nuclei. For the most part Zr H_x is not effective in thermalizing neutrons (because $h\nu \gg kT$), but it can speed up neutrons already thermalized (by the hydrogen nuclei in the tank water). Clearly, anything that changes the hydrogen lattice vibration frequency, ν, will alter the "moderating effect of the UZr H_x fuel." In turn, the vibration frequency depends on the fuel temperature, the equilibrium hydrogen pressure (between the hydrogen in the fuel-moderator and the gap hydrogen pressure), and the zirconium-hydrogen phase relationships. Damage to a fuel element is likely to affect one or more of these parameters and thereby change ν, which controls the moderating effect of the hydrogen nuclei in the U-Zr H_x.

b. Under normal operating conditions a reduction in the moderating effect of the Zr H_x would not appreciably affect the reactivity characteristics of a thermal reactor. However, if the fuel temperature goes above 600°C the hydrogen nuclei in the Zr H_x ordinarily reduce the reactivity (i.e., reduce the number of uranium fissions) by warming up the neutrons so they are no longer easily captured by the U-235 nuclei. If the moderating effect of the Zr H_x is reduced (e.g., by loss of the hydrogen through cracks in the fuel element claddings) then the reactivity characteristics will show a positive increase (i.e., increase the number of uranium fissions). In other words, the protective "warm neutron effect" which would ordinarily decrease the positive reactivity (or equivalently, increase the negative reactivity) is no longer available because of the reduction of the moderating effects usually mediated by the hydrogen nuclei in the zirconium-hydride.

c. The question is irrelevant since the AFRRI-TRIGA is strictly a thermal reactor.

d. Mathematically, the negative temperature coefficient is primarily a function of $\exp -h\nu/kT$, so that any change in the hydrogen lattice vibration frequency, ν, will necessarily modify this coefficient of reactivity. As discussed above, in part a, ν is a function of the zirconium-hydrogen phase relationships which, in turn, depends on the fuel moderator temperature and the equilibrium hydrogen pressure. Thus, any core condition that significantly changes these parameters within the fuel elements will affect the negative temperature coefficient of reactivity. Fuel element cladding failures that permit the escape of hydrogen, will undoubtedly affect the equilibrium hydrogen pressure which ultimately reduces the availability of hydrogen nuclei directly and may induce a phase transition indirectly (due to the reduced concentration or density of hydrogen). Thus, cladding damaged fuel elements can profoundly change the effectiveness of the ordinarily protective negative temperature coefficient by removing a substantial number of hydrogen nuclei (the direct effect) and by modifying ν through a phase transition (the indirect effect). This is why the Intervenor contends that whereas the mechanism for the negative temperature coefficient may operate well under ideal conditions, it may not work very well in a real accident situation (e.g., when the fuel elements may be bent, scratched, corroded, and inadequately cooled) in which case the moderating effect of the hydrogen nuclei (within the U-Zr H_x) could be seriously impaired. Therefore, we argue that this automatic shutdown mechanism is, like other so-called "failsafe" mechanisms, <u>not absolutely foolproof</u>.

e. In the context of this contention, a damaged fuel element may be functionally defined as a fuel element that either leaks hydrogen or undergoes an unusual change in the magnitude of its hydrogen lattice vibration frequency, ν, as the peak fuel temperature goes beyond 600°C.

f. Design an experiment in which cladding damaged fuel elements (that leak hydrogen) are rapidly warmed up (to temperatures of nearly 1,200°) by pulse heating. Keep all the other variables in their usual condition. Be sure to do this experiment in a safe place, not Bethesda.

g. About 1% reduction in the total core moderating effect.

h. This depends on one's criteria for significance. If 10% reduction is significant, then about ten damaged fuel elements would be required.

i. These calculations will be presented by direct testimony at the hearing itself.

j. Data from experiments such as those described above in part f of this interrogatory.

<u>INTERROGATORY 24</u>

a. They do not believe multiple cladding failure accidents are credible.

b. Not that we know of.

c. An "uncontrolled power excursion" may be defined as a large, rapid increase in neutron flux resulting from an unscheduled insertion of excess reactivity above cold critical. Associated with any power excursion is the abrupt rise in fuel temperature. The combined effects of the sudden temperature elevation and the large rapid increase in neutron flux, stress the fuel elements (involved in the power excursion) in several important ways, including:

(i) thermal migration stresses - which arise when hydrogen migrates from the higher to the lower temperature regions of the fuel element, causing the colder regions to expand and the hotter regions to contract. This results in a "migration stress" which is in the opposite sign (or direction) of the thermal stress. The brittle nature of zirconium hydride makes it susceptible to "thermal stress cracking." [See Meyer, R. D., and J. G. LeBlanc, "Negative Thermal Expansion in UZr H Reactor Fuel," <u>Trans. Am. Nucl. Soc</u>. 13, p. 2, 1970.] Furthermore, if the radial temperature gradient on the fuel element is asymmetric, bowing of the rod will occur.

(ii) anomoulous oscillations (secondary to sustained cycling) of the fuel elements if during this redistribution (of hydrogen) and bowing change the thermal gradients are altered, as would be the case in an operating reactor, the conditions for sustained cycling are obtained (i.e., anomalous oscillations). The oscillatory behavior was probably due to the clustering of fuel elements under a thermal gradient, followed by an abrupt declustering (caused by the rehydriding of the fuel elements) which applies a force in the opposite direction. [See Simnad, M. T., "The U-Zr H_x Alloy: Its Properties and use in TRIGA Fuel," GA-4314, 1980, pp. 2-17.]

(iii) surface cracks in the fuel elements effected by the escape of hydrogen - measurements and calculations have been reported of hydrogen loss from hydrided U-Zr fuel elements which were rapidly heated by induction to temperatures ranging from 900° - 1,000°C. After 75 sec at those temperatures, there was an abrupt crack in the surface of the fuel elements associated with major hydrogen evolution (i.e., strong outgassing rates). [See Leadon, B. M. et al., "Aerospace Nuclear Safety-Measurements and Calculations of Hydrogen Loss from Hydrided U-ZrH Fuel Elements During Transient Heating to Temperatures Near the Melting Point," Trans. Am. Nucl. Soc., 8, 1965, p. 8.]

(iv) excessive gap pressures - at elevated temperatures there is an increase in all three component gap pressures (i.e., residual air pressure, fission gas pressure, and the peak equilibrium hydrogen pressure) to produce a total gap pressure that may approach or even exceed 2,000 psi. Note, at temperatures above 800°C the equilibrium hydrogen pressure is, by far, the major contributor. Gap pressures of such magnitude, must put considerable stress on the fuel elements.

(v) irradiation stress resulting from the high neutron fluence caused by the power excursion (e.g., a peak power level of 8,000 MW will create a neutron fluence of about 1.0×10^{15} nvt) - large neutron fluxes induce structural flaws in the substance and claddings of the fuel elements. That is, bombardment by energetic neutrons will produce solid state defects which can migrate and coalesce to establish significant weaknesses and degradations in the fuel elements.

d. As described above in Interrogatory Answer 8.a., part 5), a LOCA would result in a sudden, large elevation of fuel temperature which is apt to produce multiple cladding failures by all of the stress mechanisms outlined in part c of this interrogatory (except that the neutron flux is generated by pulsing rather than an uncontrolled transient).

e. The conditions necessary "to breech the integrity of the fuel's cladding" are entirely divorced from the past power history of the AFRRI-TRIGA reactor. However, the immediate power history of the reactor is important in that "a recurrent pulsing mode" of operation is a vital contributing factor.

f. If the 1MW TRIGA reactor was not capable of the pulsing operation, it is unlikely that cladding failures would result from a LOCA involving that reactor. It is noteworthy, however, that the long operational history (about 20 years) of the AFRRI-pulsing TRIGA reactor is more likely to have a multiple cladding failure accident simply because of its longevity (i.e., greater accumulation of radiation-induced solid state defects within the exposed fuel elements themselves).

INTERROGATORY 25

a, b, c Liz Entwistle will respond.

D. The Licensee must concede that there are at least three essential conditions to prevent serious fuel element cladding failures, namely:

(i) fuel element temperature should never exceed the safety limit (1,000°C);

(ii) there must be no sudden core insertion of excess reactivity (i.e., >3.5% $\Delta k/k$ excess reactivity above cold critical) to produce a very large, rapid increase in neutron flux;

(iii) the fuel elements must contain the proper ratio of hydrogen to zirconium and the appropriate mixture of phases of the U-Zr H_x, so that the hydrogen can remain an effective moderator and gap pressures do not become excessive (≥1,800 psi).

Violation of one or more of these conditions can result in (multiple) cladding failure accidents (see answers to Interrogatory 8). At fuel temperatures of 1,000°C or more, almost 10% of the radioactive gases contained in the cladding-damaged fuel elements (that have redistributed into the gaps) will then diffuse out through the degradations or breaks in the claddings (see

In fact, the equivalent of one pound of radium per defective fuel element (in gaseous form) could leak out of the cladding-damaged TRIGA reactor. Another danger to public health and safety is contamination of the tank or cooling water by the (20% enriched) uranium and stored fission products in the cladding-damaged fuel elements. For example, in the event of even a single cladding failure the water activity could easily reach a level of 1μCi/cc and that level would remain elevated (because of the small decay constant of iodine-131 and iodine-133, among other reasons), which greatly exceeds the MPCs outlined in 10 CFR Appendix B. This would not only be a hazard to persons in the reactor room (occupational exposure) but if, by human error, the contaminated water leaked out into the sewage system of Montgomery County and the District of Columbia, it could cause considerable damage to the people living in these communities. Now contemplate a multiple cladding failure and the reason for our concern becomes quite understandable. The two worst accidents (i.e., maximum credible accidents) involving zirconium explosions would, of course, release the entire radioactive inventory of the reactor into the environment. Given the population density, clustering of hospitals, schools, churches, etc., in the Bethesda area, such an accident would be sheer catastrophe.

INTERROGATORY 26

a. The Licensee claims that "no waterborne radioactive emissions are generated by routine operations." Where then did the radionuclides found by the Washington Suburban Sanitary Commission reports (see WSSC reports for 1980-81) come from?

b. & c. The Licensee is legally responsible for disposing of its generated solid radioactive waste which included:

(1) contaminated animal carcasses, tissues and wastes,
(2) mixed laboratory wastes,
(3) scintillation vials with scintillant,
(4) filters which collect reactor by-products,
(5) spent radioisotopic targets,
(6) worn-out or spent fuel elements.

Solid radioactive waste is disposed of in two different ways, namely:

(i) transferred to waste disposal contractor in steel barrels for shipment to radioactive waste burial grounds,
(ii) transferred to the NNMC Radiological Safety Department in boxes for incineration.

The Intervenor contends that incineration of these solid wastes on an unrestricted area of the NNMC grounds results in the airborne release of radioactive gases and particulates that endanger the public health. For example, the AFRRI on one occasion reported the transfer of 160 boxes of contaminated waste to the NNMC for incineration, noting that the principal isotope was Sodium -24. The estimated total activity of this isotope (i.e., Na-24) was approximately 1.6 mCi. Now the annual licensed quantity equivalent (per 10 CFR Appendix C) is no more than 10 μCi, so that this represents (more than a one-hundredfold excess) a clear violation of that safeguard. Another example of dangerous accumulated quantities of radioactive solid waste occurred in 1973 when shipment records indicate that 80 Ci of tritium (targets) were transferred offsite. Since the annual licensed quantity equivalent (per 10 CFR 20 Appendix C) is only one m Ci, this again represents an infraction of Federal regulation and a potential public hazard (see NRC Inspection Reports covering 1975-76, specifically Report No. 50-170/March 1977).

d. We define a "probable violation" as one in which a violation is apt to have been committed except that one or more of the following applies:

(i) the appropriate data were not recorded which would have established the violation,
(ii) the appropriate data were recorded but the reports have been doctored or kept secret,
(iii) the instrumentation or method of measurement was inadequate to actually obtain the pertinent data demonstrating the violation.

e. Specific regulatory limits that have been violated by the AFRRI include:

(1) The concentration of the gaseous radionuclide Argon-41 (Ar-41) released from the AFRRI stack in

1962, 1963, and 1964 exceeded the MPCs listed in 10 CFR 20 Appendix B.

(2) The yearly environmental monitoring data (obtained from the AFRRI Perimeter Monitoring System) demonstrate that the AFRRI has exceeded the well-known Federal regulation that the average yearly ambient radiation levels be less than the 0.5 Rem per year for unrestricted areas. This was the case for both 1962 and 1963.

(3) The data (see Table 2 below) clearly demonstrate that the AFRRI has consistently exceeded the annual exposure EPA limit of 25 mRem for unrestricted areas surrounding a nuclear reactor (see EPA Resolution No. 40 CFR 190).

(4) Technical specifications require the AFRRI reactor building ventilation to exhaust to a stack having a minimum elevation of 18 feet above the roof level of the highest building in the AFRRI complex. Contrary to the above, a leak through a stack drain line discharged part of the exhaust at ground level outside the building <u>for a period of several months</u>.

(5) During the period January 1, 1970 to July 1, 1971 the "normal exposure rate" was 0.5 mRad/hr, however, there were several unrestricted areas where the exposure rate rose to 1 mRad/hr or more and at least one specific unrestricted area where the dose rate approximated 5 mRad/hr. The maximal permissible annual exposure, by NRC regulations, is 500 mRem per year. Hence, any person who lived or worked in these unrestricted areas where the dose rate was 1-5 mRad/hr, would have received excessive radiation if they had been exposed for merely 500 hours during the entire year, or about 10 hours per week. Since the Licensee has failed to convincingly demonstrate that this could not have occurred, it represents a clear violation of the ALARA principle (the goal embodied in 10 CFR part 50) as well as the Federal regulation (requiring that no one receive more than 500 mRem per year). The locations of maximum ionization chamber readings were partly in residential areas. Note approximately 50-60% of the area within a one-mile radius of the AFRRI stack is, in fact, residential.

(6) According to 10 CFR 20.201(b) the "Licensee shall make or cause to be made such surveys as may be necessary for him to comply with the regulations in this part." Contrary to this directive, gross beta measurements of liquid effluents to assure compliance with 10 CFR 20.303, "Disposal by release into sanitary sewage systems," made for the period January 1976 to January 1977 were inadequate in that the gross beta measurements were made without the use of beta self-absorption correction in the presence of significant amounts of suspended solid material.

(7) The AFRRI on one occasion reported the transfer of 160 boxes of contaminated waste to the NNMC for incineration, noting that the principle isotope was Sodium-24. The estimated total activity of this isotope (i.e., Na-24) was approximately 1.6 mCi. Now the annual licensed quantity equivalent (per 10 CFR 20 Appendix C) is no more than

$10\mu Ci$, so that this represents (more than a one hundred-fold excess) a clear violation of that safeguard.

(8) Another example of dangerous accumulated quantities of radioactive solid waste occurred in 1973 when shipment records indicate that 80 Ci of tritium (targets) were transferred offsite. Since the annual licensed quantity equivalent (per 10 CFR 20 Appendix C) is only one mCi, this again represents an infraction of Federal regulation and a potential public hazard.

f. Specific radiation monitoring methods that the Intervenor considers inadequate, include:

(i) The statistical uncertainty in the annual perimeter dose per monitoring station is ± 20 mRad at the 95% confidence level. This is totally inadequate and could easily be remedied.

(ii) The environmental film dosimetry method employed at the monitoring stations detects only external gamma radiation. Thus, the population radiation exposure dose due to the inhalation or ingestion of radionuclides is entirely neglected.

(iii) The particulate radioactivity monitor for airborne radioactive effluents (i.e., a pancake-probe G-M counter) is not isokinetic and therefore cannot be used for any quantitative evaluations.

(iv) The dose rates (using ionization chamber type instrumentation or an alternative) are not determined with sufficient frequency either for restricted or unrestricted areas (both on and offsite).

(v) The "Environmental Sampling and Analysis" program has been criticized for calculational omissions, the manner in which the samples were prepared for analysis, and the type of instrumentation used to perform the analyses. For these and other reasons, this program should be administered by private and public scientific agencies outside of the Department of Defense (e.g., Washington Sanitary Sewage Commission, the EPA, the Sierra Club, etc.).

g. The principle of self-regulation when it comes to radiation monitoring is highly suspect. The public would be better served if such monitoring were left to private scientific laboratories under the authority and inspection of local government agencies. This puts responsibility for public safety and protection precisely in the hands of the local people who need that protection. Federal agencies could advise, fund, and help implement the appropriate radiation monitoring methods when "corrective actions" are truly indicated to prevent violations of regulatory limits. As long as the Licensee itself has the primary monitoring responsibility they are likely to be inadequate.

h. We are not presently aware of detailed "specific corrective actions" being undertaken by the AFRRI. If the Licensee wishes to share such information, we would be ready to comment on its "adequacy or inadequacy" to prevent a recurrence.

i. We believe "probable violations" have occurred in the following instances:

(1) Since the statistical uncertainty in the annual perimeter dose per monitoring station (at the 95% confidence level) is $\pm$ 20 m Rad, it is likely that the annual population exposure at several unrestricted area stations has exceeded the EPA limit (i.e., 25 m Rem) for just about every year during the past 20 years of the AFRRI Facility operation.

(2) The absence of data due to omission of internal radiation makes it virtually impossible to evaluate the true population exposure to radiation, let alone determine whether the Federal regulatory limits have actually been exceeded.

(3) The only two AFRRI particulate radioactivity monitoring systems (i.e., the pancake-probe G-M counter and the radioactive gas monitor) are not reliable for quantitative particulate radioactive sampling. Hence, one can only obtain crude estimates of the airborne radioactive particulates that have been dispersed into the environment. The true values may, in fact, have exceeded public safety limits.

(4) The maximum permissible annual exposure, by NRC regulations, is 500 mRem per year. It is very likely that any person who lived or worked in an unrestricted area where the dose rate was 1-5 m Rad/hr, would have received radiation in excess of this "maximum permissible annual exposure."

j. The following is a list of sources used to document the contention raised above, that the past and present operation of the AFRRI reactor has resulted in probable violations of 10 CFR part 20:

(1) See the letter from the AEC to the AFRRI dated October 6, 1961 which predicts that, according to their calculations for Argon-41 concentrations in unrestricted areas, AFRRI will probably not be able to meet the MPC release standards for unrestricted areas as stated in 10 CFR 20, Appendix B.

(2) See Environmental Release Report (AFRRI-TRIGA Reactor) covering the period 1 Jan. 1970 to 30 Sep. 1971, issued on December 14, 1971.

(3) See Inspection Report No. 50-170/77-01-03 that discusses (gaseous effluent) airborne particulate evaluation.

(4) See the AFRRI-TRIGA Reactor Environmental Release Report issued on December 14, 1971 by AFRRI and the DNA.

(5) In its Environmental Impact Appraisal the Licensee notes that the highest average unrestricted area exposure rates corresponding to given airborne releases are 4.1 m Rem/hr for Argon-41, 4.3 m Rem/hr for the combination of both Nitrogen-13 and Oxygen-1-, and 0.5 m Rem/hr for Xenon-133. These are high dose rates and since they admittedly extend to residential areas, it is quite possible that people may have received in excess of 500 m Rem (the regulatory limit) in any given year. It is also another illustration of violation of the ALARA principle originally designed to protect the public from excessive exposure.

(6) See the AFRRI Environmental Release Data and Perimeter Monitoring Reports Docket No. 50-170 (e.g., May 27,

1966 report, September 20, 1966 report, and December 14, 1977 report).

(7) See the AFRRI's written response to Mr. Joe Miller's (from Citizens for Nuclear Reactor Safety) question #11.

(8)

Table 2

Year	Average Annual Perimeter Dose (per monitoring station)
1962	242 mRad
1963	231 mRad
1964	89 mRad
1965	55 mRad

k. The following is a list of sources used to document the contention raised above, that the past operation of the AFRRI Reactor has resulted in actual violations of 10 CFR part 20 (also see answer to part e. where the actual violations are listed).

(1) With respect to the excessive release of Argon-41 please see the AFRRI Airborne Release Reports for 1962, 1963, and 1964; and the AEC Inspection Reports for 1962, 1963, and 1964, in Docket No. 50-170.

(2) The yearly environmental monitoring data (obtained from the AFRRI Perimeter Monitoring System reports) demonstrate that the AFRRI has exceeded the annual federal limit, 0.5 Rem, for the average yearly ambient radiation.

Table 1

Year	Maximum Annual Exposure	Specific Perimeter Station
1962	> 500 mRad	2c
1963	> 500 mRad	16A
1964	116 mRad	2A
1965	112 mRad	16A
1970	76 mRad	16A
1978	30 mRad	11A

(3) The data (see Table 2 above) also clearly demonstrate that the AFRRI has consistently exceeded the annual exposure EPA limit of 25 mRem for unrestricted areas surrounding a nuclear reactor (see EPA Resolution No. 40 CRF 190).

(4) Regarding the ground level leak of gaseous effluent, see NRC Inspection Report conducted on January 11, 1979 (i.e., Inspection Report No. 50-170/79-01).

(5) See Violation Notice of Gross Beta Effluent Analysis based on the NRC inspection of January 12-14, 1977. Also see Inspection Report No. 50-170/77-01-02.

(6) With regard to violations concerning solid radioactive waste see NRC Inspection Reports covering 1975-76, specifically Report No. 50-170/March 1977.

l. The answer to this question is essentially contained in the answer provided to Interrogatory 26, parts f, g, and h, given above.

INTERROGATORIES 27, 28, 29, 30

See Testimony of Professor Ernest J. Sternglass to be presented at the Hearings.

INTERROGATORY 31

All of the component questions center on the meaning given to the phrase "highly probable." In order to assess the probability that the MPCs (set forth in 10 CFR 20, Appendix B) have been exceeded, we would need considerable information (not made available) regarding the individual radionuclide concentrations, air flow parameters and meteorological data occurring during those several months.

INTERROGATORY 32

a. Our contention that air convection cooling alone would not be sufficient to cool an operating TRIGA reactor core during and immediately following a LOCA, is based on calculations contained in the 1963 GA-2025 Hazards Summary Report for the 250 kW Mark II TRIGA Reactor (Pulsing) located at Columbia University in New York City.

b. Internal gap pressures could rise to 1,800 psi or more depending on the temperature elevation. This pressure is capable of producing breaks or penetrations of the fuel element cladding (see Figures 2-9, "Equilibrium Hydrogen Pressure over Zr $H_{1.65}$ versus Temperature" in the 1980 GA-4314 Report by M. T. Simnad).

c. TRIGA reactors have definitely had cladding failures. However, whether any such failure has ever been the result of a LOCA is unclear.

d. The Licensee asserts that the maximum amount of fission products that could be released in the event of a cladding failure of a single average fuel element in the AFRRI-TRIGA core is less than 7 curies during steady state operation and also 7 curies during pulse operation (following steady state operation). These calculations, in turn, are based on the assumption that the fraction of gaseous fission products (i.e., radioisotopes of Iodine, Krypton, and Xenon) released from U-Zr H_x fuel into the gap between the fuel material and the fuel element cladding is only 0.1%. That assumption is valid, however, only if the fuel temperature is below 550°C where emission of radioactive gases is largely controlled by recoil effects. However, in a LOCA-induced cladding failure the temperature will rise way above 550°C, so that the process of gas emission into the gap becomes mostly diffusion controlled and radioactive gases begin to stream out of the cladding-damaged fuel element. In fact, a LOCA is apt to produce temperatures in excess of 1,000°C, which means that nearly <u>10% of the radioactive gases in the fuel element escape into the gap region</u> between the fuel and the cladding. Practically all of these gaseous radioisotopes will find their way out of the fuel element gap into the reactor room atmosphere through the penetrations in the damaged cladding. Thus, if 0.1% fraction of gaseous fission products results in a release of above 7 curies, then a fraction of nearly 10% would result in the release of <u>at least 500 curies</u> from a LOCA-induced cladding failure of a single average fuel element. The presence or absence of any breach of containment within the reactor room, will determine the amount of gaseous radioisotopes (e.g., Iodine, Krypton, and Xenon) that ultimately leak into the outside environment (see Figure 5-1, "Fractional Release of Gaseous Fission Products from TRIGA Fuel," p. 5-3 (in the 1980 GA-4314 Report by M. T. Simnad).

INTERROGATORY 33

a. Pulse heating leading to sudden elevations of temperature sufficient to cause multiple cladding failures, has been described several times throughout the body of these Interrogatory Answers (e.g., see Answers to Interrogatories
In the specific case of a LOCA-induced multiple cladding failure, the timing aspect is crucial. It is crucial because the loss of the water is not only a loss of coolant, for a TRIGA reactor it is also the loss of the primary moderator since it is the hydrogen nuclei within the water which actually thermalize the prompt neutrons (i.e., the fast neutrons leave the fuel elements and enter the tank water where they give up their excess energy to become slow neutrons capable of initiating more U-235 nuclear fission). Thus, if the loss of tank water occurs too rapidly there can be no pulsing or inadvertent power excursions (transients). Realistically, we calculate that water could leave the tank at a maximum rate of about 250 gallons per minute (in the several possible ways described above). At this rate of moderator (water) loss, pulsing would not be prohibited and yet cooling would be seriously impaired, producing exactly the necessary conditions for cladding failures to occur.

b. Temperatures above 900°C are rarely, if ever, reached by a single pulse. Indeed, it would take many pulses to establish fuel temperatures of this magnitude rapidly. Temperature fluctuations required to produce cladding failures in a reactor core could result by repeated activation of the pulsing mode following a period of steady state operation.

c. In the answer to Interrogatory 32, part d, we established a fission product release of about 500 curies from a single cladding failure accident if the fuel temperature goes above 1,000°C. Thus, even a single cladding failure would release gaseous radionuclides (e.g., Argon, Krypton, Xenon, Iodine) beyond the limits imposed by 10 CFR 20. One should also allow for radionuclide leaks into the water coolant, given a serious cladding failure (see Interrogatory Answer   for details). A multiple cladding failure accident would result in a horrendous release of fission and activation products.

d. The temperature history during a LOCA associated with one or more transients depends entirely on a number of conditions, including the rate of loss of coolant (water), the number and magnitude of the pulses or transients, the previous events immediately preceding the accident (e.g., duration of steady state operation), condition of the fuel elements, etc. Given these and other boundary conditions, we might be able to calculate the temperature history very approximately!

INTERROGATORY 34

The medical concern over radiation exposure secondary to inhalation of gaseous radioisotopes (such as the noble gases) continues to engender controversy among physicians and health physicists. Internal emissions have been a topic of medical symposia such as the NIH Conference about one year ago. In order to do this subject technical justice will require considerable scientific explanation and justification. We expect this topic to be discussed by several of our witnesses at the time of the full hearings, including Dr. Irving Stillman, Dr. Ernest Sternglass, and Dr. Irwin Brass.

INTERROGATORY 35

a. The release of any radionuclide (or combination thereof) into the environment that would expose the public to a dose rate of 100 mRem/hr or more for at least one hour (i.e., a total exposure of at least 100 mRem), constitutes, in our opinion, a "significant offsite release."

b. Since the AFRRI nuclear reactor has been operational for the past 18 years, one can estimate that most of the 87 fuel elements have a radioactive inventory equivalent to about ten pounds of radium per element. This inventory specifically includes, among others, such radionuclides as Uranium -235, Strontium-89, Iodine-131, Cerium-144, Cesium-134, Yttrium-91, Krypton-85, Strontium-90, and Cesium-137; all with considerable activities and prolonged half-lives. Thus, if the radioactive contents of even one fuel element were dispersed into the environment by a chemical explosion (described above) the fission and activation products released to an unrestricted area would violate the 10 CFR (part 100) guidelines many times over.

c. Any of the chemical explosions described above in the two "worst-case" scenarios (e.g., hydrogen-oxygen, zirconium-steam and zirconium-oxygen) could trigger a series of such explosions. Since the AFRRI-TRIGA has no containment dome, these types of accidents would widely disperse the radioactive inventory originally contained within the reactor core. These offsite releases would not merely be "significant," they would be catastrophic.

d. This has been detailed in the answer to Interrogatory given above. However, the sequence may again be briefly outlined as follows: the power excursions produce fuel-moderator heating in an abrupt or rapid manner. The acute temperature fluctuations and massive, sudden increases in neutron flux, effects one or more cladding failures, causing a loss of the hydrogen (that migrates to and accumulates in the gaps) through breaks in the element claddings. Thus, the thermalizing effect of the hydrogen within the fuel elements is severely compromised.

e. The loss of hydrogen nuclei from the U-Zr H_x necessarily reduces its moderating effects and via changes in hydrogen density can also induce phase changes, both of which modify the prompt negative temperature coefficient. A reduction in the effectiveness of the negative temperature coefficient to protect the TRIGA reactor during an accident, would have serious consequences.

f. The explosive zirconium-steam reaction requires zirconium, water or water vapor, and high temperatures (see

g. To the best of our knowledge, all of these necessary conditions can exist within an overheated AFRRI-TRIGA nuclear core (see

h. The explosive zirconium-air interaction requires zirconium, air or oxygen, and high temperatures (see

i. To the best of our knowledge, all of these necessary conditions can exist within an overheated TRIGA core following a LOCA (see

j. This is merely a matter of semantics. Obviously we believe that the two maximum credible accidents (described above) should be designated as the design basis accidents for the AFRRI-TRIGA, rather than the two relatively trivial accidents presently designated as such.

INTERROGATORY 36

a. (2) During the period January 1, 1970 to July 1, 1971 the "normal exposure rate" was 0.5 mRad/hr, however, there were several unrestricted areas where the exposure rate rose to 1 mRad/hr or more and at one specific unrestricted area where the dose rate approximated 5 mRad/hr. The maximum permissible annual exposure, by NRC regulations, is 500 mRem per year. Hence, any person who lived or worked in these unrestricted areas where the dose rate was 1-5 mRad/hr would have received excessive radiation if they had been exposed for, at most, 500 hours during the entire year, or about 10 hours per week (see the AFRRI-TRIGA Reactor Environmental Release Report issued on December 14, 1971 by AFRRI and the DNA).

(1) In its EIA, the Licensee notes that the highest average unrestricted area exposure rates corresponding to given airborne releases are 4.1 mRem/hr for Argon-41, 4.3 mRem/hr for the combination of both Nitrogen-13 and Oxygen-15, and 0.5 mRem/hr for Xenon-133. These are high dose rates and since they admittedly extend to residential areas, it is quite possible that people may have received in excess of 500 mRem (the regulatory limit) in any given year.

(3) and (4) have already been discussed.

b. and c. Data contained in Table 1 and Table 2 given above.

d. The dose rates alluded to were determined by ionization chamber instruments, not the film badges at the environmental station monitors. The large dose rates were attributed to excessive X-Radiation coming from a large X-Ray machine (called the Maxitron). However, this interpretation was never verified.

e. If it is a typographical error then it was made by the AFRRI or DNA in the Environmental Release Report issued on December 14, 1971.

f. We believe so. No.

g. This information describes the radioactive inventory contained within the AFRRI-Reactor Core. It was used whenever such information was needed to make a specific point.

h. Answer by Liz Entwisle. No CNRS members live within 600 feet of the AFRRI stack. The Intervenor is without knowledge of the address of any residence within 600 feet of said stack.

INTERROGATORY 37

a. Admiral Robert Monroe, The Washington Star, Tuesday, August 14, 1979; "Colonel MacIndoe," Montgomery Journal, by Sandy Golden, Wednesday, June 27, 1979.

b. The Intervenor is unable to answer this question without access to information not in the public record.

c. Same as b.

d. Each of the cited instances demonstrates a break in the first and last layer of security between the controlled access areas and the public.

Respectfully submitted,

Elizabeth B. Entwisle
Counsel for Intervenor

AFFIDAVIT OF ELIZABETH B. ENTWISLE

I, Elizabeth B. Entwisle, being duly sworn, do state:

1. That the Response of Intervenor Citizens for Nuclear Reactor Safety, Inc. to the Licensee AFRRI's First Set of Interrogatories were prepared under my direction and supervision.

2. That the responses therein designated "Answered by Entwisle" were answered by me.

3. That the responses designated "Answered by Stillman" were answered by Dr. Irving Stillman.

4. That the responses are true to the best of my knowledge, information, and belief.

X
Elizabeth B. Entwisle

SUBSCRIBED AND SWORN TO before me this 24th day of December, 1981.

Donna J. Zebe
Notary Public 7-1-82

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

CERTIFICATE OF SERVICE

I hereby certify that the foregoing Response of CNRS to Licensee's First Set of Interrogatories and Affidavit was served on the following by depositing in the United States Mail, first class, this 29th day of December 1981.

Louis J. Carter, Esq., Chairman
Administrative Judge
Atomic Safety and Licensing Board
23 Wiltshire Road
Philadelphia, PA 19151

Mr. Ernest E. Hill
Administrative Judge
Lawrence Livermore Laboratory
University of California
P.O. Box 808, L-123
Livermore, CA 94550

Mr. David R. Schink
Administrative Judge
Department of Oceanography
Texas A & M University
College Station, TX 77840

Docketing and Service Section
Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. Robert L. Brittigan, Esq.
General Counsel
Defense Nuclear Agency
Washington, D.C. 20305

Mr. Richard G. Bachmann, Esq.
Counsel for NRC Staff
Office of the Executive Legal Director
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. Stuart A. Treby, Esq.
Assistant Chief Hearing Counsel for NRC Staff
Office of the Executive Legal Director
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Atomic Safety and Licensing Appeal Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Elizabeth B. Entwisle
Counsel for Intervenor

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)

DECLARATION OF ROBERT L. BRITTIGAN

The Intervenor's response to Licensee's First Interrogatories to the Citizens for Nuclear Reactor Safety, Inc. (Exhibit B), was received in my office on January 4, 1982. The attached motion to compel, under the provisions of 10 C.F.R. 2.740(f), should have been filed on January 14, 1982. On January 14, 1982, however, a severe snowstorm had rendered travel in the Washington, D.C. area hazardous and non-essential Federal employees were asked to remain at home. Accordingly, typing and reproduction could not be completed on that date. It is the position of the Licensee that January 14, 1982 should be treated as a legal holiday for the purposes of computing time under 10 C.F.R. 2.710 and 10 C.F.R. 2.740(f).

I have discussed the matters raised in the attached motion to compel with Ms. Elizabeth B. Entwisle, Esq., Counsel for the Intervenor, and we have agreed to attempt to resolve the question of the adequacy of the Intervenor's response informally during the next 45 days. I have been authorized to inform the Board that Ms. Entwisle joins in the Licensee's request that the Board defer ruling on the attached motion to compel for a period of 45 days to permit informal resolution of the matter.

I hereby declare under penalty of perjury that the foregoing is true and correct.

Robert L. Brittigan

ROBERT L. BRITTIGAN
Counsel for Licensee

Signed at Headquarters, Defense Nuclear Agency, 6801 Telegraph Road, Alexandria, Virginia this 15th day of January 1982.

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)

ORDER

Upon consideration of the Licensee's Motion to Compel Answers to Licensee's First Interrogatories to the Citizens for Nuclear Reactor Safety, Inc., the opposition thereto, and the entire record herein, the Board finds, pursuant to 10 C.F.R. § 2.740(f), that Intervenor CNRS's Responses to Licensee's First Set of Interrogatories are incomplete, evasive, or both, with respect to interrogatories numbers 6, 7b, 7c, 7d, 7e1, 7e2, 7e4, 7j, 7n, 8b, 8c, 9a, 10a, 10h, 12a, 12b, 12c, 12d, 13, 14, 15, 16, 17, 18, 19, 20, 21, 22, 23a, 23b, 23c, 23d, 23g, 23h, 23i, 24a, 24c, 24g, 24h, 24i, 25a, 25b, 25c, 26a, 26g, 261, 27, 28, 29, 30, 31b, 31c, 32b, 32c, 34a, 34b, 34c, 34d, 34e, 34f, 35e, 35f, 35g, 35h, 35i, 36a(2), 36a(3), 36a(4), 36c, 36d, and 36g and that the Citizens for Nuclear Reactor Safety, Inc. has therefore failed to respond to the cited interrogatories.

Accordingly, it is hereby ORDERED that Intervenor Citizens for Nuclear Reactor Safety, Inc. provide to the Licensee full and complete answers to the interrogatories listed above not later than 20 days from the date of this ORDER.

FOR THE ATOMIC SAFETY AND
LICENSING BOARD

Louis J. Carter, Chairman
ADMINISTRATIVE JUDGE

Dated at Bethesda, Maryland

this ____ day of ________ 1982.

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 15 JANUARY 1982 FILING

I hereby certify that true and correct copies of the foregoing "MOTION TO COMPEL ANSWERS TO LICENSEE'S FIRST INTERROGATORIES TO THE CITIZENS FOR NUCLEAR REACTOR SAFETY, INC.," "MEMORANDUM IN SUPPORT OF MOTION TO COMPEL ANSWERS TO LICENSEE'S FIRST INTERROGATORIES TO THE CITIZENS FOR NUCLEAR REACTOR SAFETY, INC," "DECLARATION OF ROBERT L. BRITTIGAN," and proposed form of "ORDER" were mailed this 15th day of January, 1982, by United States Mail, First Class, to the following:

Louis J. Carter, Esq., Chairman
Administrative Judge
Atomic Safety and Licensing Board
23 Wiltshire Road
Philadelphia, PA 19151

Mr. Ernest E. Hill
Administrative Judge
Lawrence Livermore Laboratory
University of California
P.O. Box 808, L-123
Livermore, CA 94550

Dr. David R. Schink
Administrative Judge
Department of Oceanography
Texas A&M University
College Station, TX 77840

Mr. Richard G. Bachmann, Esq.
Counsel for NRC Staff
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Elizabeth B. Entwisle, Esq.
8118 Hartford Avenue
Silver Spring, MD 20910

Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Atomic Safety and Licensing Appeal Panel (5)
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Secretary (21)
U.S. Nuclear Regulatory Commission
ATTN: Chief, Docketing and Service Section
Washington, D.C. 20555

Robert L. Brittigan

ROBERT L. BRITTIGAN
Counsel for Licensee