BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION
ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:
Marshall E. Miller, Chairman
Gustave A. Linenberger, Jr.
Cadet H. Hand, Jr.

In the Matter of U.S. DEPARTMENT OF ENERGY PROJECT MANAGEMENT CORPORATION TENNESSEE VALLEY AUTHORITY (Clinch River Breeder Reactor)	)	Docket No. 50-537

NATURAL RESOURCES DEFENSE COUNCIL, INC., AND THE SIERRA CLUB
PROPOSED FINDINGS OF FACT ON
<u>CONTENTIONS 1, 2, AND 3 AS THEY RELATE TO SITE SUITABILITY</u>

Pursuant to 10 CFR §2.754(a), and in accordance with the Board's August 27, 1982 rulings, Intervenors, Natural Resources Defense Council, Inc., and the Sierra Club, hereby submit their proposed findings of fact on Intervenors' Contentions 1, 2, and 3 as they relate to site suitability under 10 CFR §50.10(e)(2)(ii) and Part 100.

DS03

A. <u>An LMFBR Requires a Higher Standard of Protection Against CDAs Than an LWR, and Should Include CDAs Within the Design Basis</u>

1. A liquid metal fast breeder reactor (LMFBR) is different from a light water reactor (LWR) in several respects which militate in favor of providing full design basis protection against a core disruptive accident (CDA); that is, providing safety systems meeting the requirements of 10 CFR Part 50 and Appendices, or their equivalent, which would mitigate a CDA and prevent releases of radioactivity in excess of the 10 CFR Part 100 guidelines:

a) An LMFBR can undergo a nuclear explosion (energetic CDA). (Tr. 2765-2781, Cochran).

b) A nuclear explosion in an LMFBR provides a potential mechanism for release, in vapor or particulate form, of substantially larger fractions of fuel (plutonium) and fission products to the containment atmosphere, and consequently to the environment, than would be released following a non-energetic core melt accident.

c) LMFBRs generally contain several times the core inventory of the highly toxic isotopes of plutonium than do LWRs.

d) Release of plutonium into the environment following CDAs in LMFBRs potentially represents a far more serious contamination problem than contamination by fission product release (I-131) following LWR core melt accidents, due to the long half-life and extreme toxicity of plutonium.

e) In contrast with LWRs, over 150 of which have been licensed for construction, there is virtually no experience with reactors of the general size and type as the CRBR.

f) It is not possible to satisfactorily model the behavior of the CRBR core once cladding melting begins.

g) Even if such modeling could be done with sufficient precision, it has not been.

(Tr. 2818-20, Cochran).

2. Respected members of the technical community believe that LMFBRs may require a higher standard of protection against CDAs than LWRs because of the greater potential capacity for harm and the uncertainty and complexity which attend analysis of low probability accidents in fast reactors. (Tr. 2820, Cochran).

B. CDAs Occurred or Were Considered as DBAs in Other U.S. Fast Reactors

3. The Experimental Breeder Reactor-I (EBR-I) was a small (1 MWt), early (initial operation 1951), experimental breeder, the reactor core of which was inadvertently substantially melted in an experiment in 1955. (Tr. 2822-23, Cochran).

a) The Staff did not know whether the accident that occurred at EBR-1 was considered credible at any time. (Tr. 2290, Morris.)

4. The Enrico Fermi-1 plant was a 200 MWt demonstration LMFBR which was licensed by the AEC to operate in 1963. In 1966 Fermi-1 experienced a core melt accident more severe than had been considered "credible" during the plant's licensing. Fuel melted in two subassemblies, but melting in only one subassembly was considered the maximum "credible accident." (Tr. 2823-24, Cochran).

5. SEFOR was an experimental 20 MWt reactor designed to be subjected, in an experimental program, to intentional power excursions in order to test the Doppler coefficient. (Tr. 2824, 2638-39, Cochran).

a) The containment "design basis energy release" for SEFOR was 400 MW-sec., far more than the 100 MW-sec. the AEC staff concluded was the "theoretical upper limit of the energy available as kinetic energy." (Tr. 2825, Cochran).

b) Thus, a CDA was in effect treated as a design basis accident for SEFOR and the containment was designed to withstand the maximum calculated energetic releases with conservative safety margins. (Tr. 2825, 2786-88, Cochran).

c) Applicants admitted that CDAs were within the equivalent of the third level of design safety (the design basis) for SEFOR. (Tr. 1502, Brown).

6. The Fast Flux Test Facility (FFTF) is a 400 MWt fast neutron test reactor which was not licensed but did undergo a safety review by the AEC staff. (Tr. 2825, Cochran).

a) It is a clear inference from the Safety Evaluation Report for the FFTF that CDAs were treated as equivalent to design basis accidents for the plant. (Tr. 2825-27, 2639-40, 2643, 2790-91, Cochran).

b) Applicants testified that a CDA was within the third level of safety for the FFTF. (Tr. 1502, Brown).

c) Although Staff witnesses asserted that CDAs were not considered design basis events for the FFTF, they provided no factual basis for their assertion and failed to address directly the evidence in the FFTF Safety Evaluation offered by Intervenors. (Tr. 2395-96, King, Long).

7. EBR-II is an experimental 67.5 MWt fast neutron test reactor which was not licensed but did undergo an AEC safety review. (Tr. 2823, Cochran).

a) Its primary containment was designed to contain "without breaching" a "reasonable" upper limit on the explosive energy of about 300 lb. TNT. (Tr. 2790, 2823, Cochran).

8. It is clear from Findings 4-7, <u>supra</u>, that of the four U.S. fast reactors of significant size, CDAs were treated, in

effect, as design basis accidents or their equivalent for EBR-II, SEFOR, and FFTF. The fourth fast reactor, FERMI-1, in fact experienced an accident greater than its design basis. (Tr. 2828, Cochran).

C. <u>The NRC Staff Originally Considered CDAs as DBAs for the CRBR and Has Demonstrated No Rational Basis For Its Change in Position</u>

9. In 1975 and 1976, prior to the May 6, 1976 letter from Richard P. Denise to Lochlin W. Caffey (Staff Exhibit 5), the NRC Staff took the position that CDAs should be within the design basis for the CRBR. (Tr.2621-22, 2650-53, 2834, Cochran). The Staff did not contradict Intervenors' evidence on this point. (Tr. 2268, Hulman; 2219, Morris).

a) Applicants in fact included core disruptive accidents within the CRBR design basis in the Parallel Design in order to get the review of the CRBR application underway. (Tr. 1503, Strawbridge; 2831, Cochran).

10. The statement in Staff's Exhibit 5, that the probability of CDAs "can and must" be reduced to a level justifying exclusion from the design basis, is a hypothesis and not a fact. (Tr. 2835-36, Cochran; 2270, Morris).

11. There is absolutely no evidence in the record of any factual or analytical basis for the statement in Staff Exhibit 5

that the probability of CDAs "can and must" be made sufficiently low for the CRBR. (Tr. 2836-37, Cochran). The Staff could offer nothing more than speculation regarding the change in Staff's position in 1976. (Tr. 2270, Morris).

D. The Staff Has Failed to Establish and Justify any Principal Design Criteria Which, If Met, Would Ensure That the Probability of a CDA is Sufficiently Low to Exclude CDAs From the Design Basis

12. The Staff has failed to establish and justify any design criteria which, if met, would ensure that the probability of a CDA for a reactor of the general size and type as the CRBR is "sufficiently low" to exclude CDAs from the design basis. (Tr. 2853-57, Cochran.)

a) There are no general design criteria established for fast reactors. (Tr. 2853-54, Cochran.)

b) The Staff's review of Applicants' proposed general design criteria for the CRBR (1982 SSR, Appendix A) (Staff Exhibit 1, Appendix A) is not complete and will not be set out until the CRBR Safety Evaluation Report (SER) is published. (Tr. 2854, Cochran; 2148-49, 2408, Morris).

c) There is no way of judging whether Applicants' proposed CRBR general design criteria will achieve the goal of comparability between the risks associated with LWRs and the risks associated with the CRBR, since no analysis has been performed to match the existing LWR criteria (10 CFR Part 50) against the proposed CRBR criteria. (Tr. 2854, Cochran).

d) Which LWR criteria should apply to the CRBR, which should be adapted and how that should be accomplished, and what new criteria should be established in areas not covered by LWR criteria, are complex questions which are as yet unresolved. (Tr. 2854, Cochran.)

e) The general design criteria by which the CRBR is supposedly to be judged are being developed at the same time that the design for the plant is being finalized, and are apparently being developed based on the plant's design, rather than vice versa. (Tr. 2855, Cochran).

f) The ACRS letter of July 13, 1982 (Staff Exhibit 4) indicates that the ACRS does not necessarily agree with all the LMFBR Design Criteria specified in Staff Exhibit 1, Appendix A. (Tr. 2856, Cochran.)

g) The Staff claims that these Staff Exhibit 1, Appendix A, criteria illustrate the feasibility of developing criteria suitable for a plant of this general size and type. (Tr. 2409-10, Morris). This claim has no merit. One cannot demonstrate the feasibility of developing suitable criteria without some demonstration that the criteria are in fact suitable. As shown in Finding 12(b)-(f), above, no such demonstration has been made.

E. <u>The Staff's Claim That It Has Established Specific Criteria That Would Render CDAs Sufficiently Improbable Is Without Merit</u>

13. The Staff claims that, despite the admitted lack of principal LMFBR or CRBR design criteria analogous to the LWR General Design Criteria in 10 CFR Part 50, Appendix A, the following requirements constitute a set of specific "criteria" sufficient to render CDAs so improbable that they need not be considered design basis accidents:

(1) shut down the nuclear chain reaction upon initiation of transients;
(2) maintain sufficient coolant inventory;
(3) maintain sufficient coolant flow;
(4) remove sufficient heat from the fuel;
(5) avoid propagation of local fuel faults;
(6) employ generic design measures to achieve reliable safety systems and features.

(Tr. 2458-68; 2406-10, Morris.) This claim is wholly without merit.

a) These "criteria" are so vague as to be meaningless. They provide no indication of what measures are necessary, for example, to shut down the nuclear chain reaction, or what degree of conservatism is appropriate or sufficient.

b) The Staff admits that these criteria do not have specific detail. (Tr. 2206, Morris.)

c) The Staff admits that it can think of no reactor in the world that is not required, for example, to shut down the nuclear chain reaction upon initiation of transients or to maintain sufficient coolant inventory. (Tr. 2207, Morris.)

d) These criteria provide no indication whatsoever that, if met, they will ensure that the probability of CDAs is sufficiently low that they may be excluded from the design basis. In particular, the Staff has failed to demonstrate that meeting these general requirements will ensure that the CRBR meets its safety objective (Findings 14-15 below), or that in fact such criteria can be met at all (Findings 16-19 below).

F. The Staff Has Failed to Demonstrate That The CRBR Meets Or Even Approaches the Staff Safety Objective.

14. In order to determine whether CDAs should be within the CRBR design basis, the Staff currently uses the safety objective that there be no greater than one chance in a million (10^{-6}) per reactor year of a CRBR radioactive release with potential consequences greater than the 10 CFR Part 100 dose guidelines. (Tr. 2277-79, Morris; Staff Exhibit 5) The Applicants have also proposed this approach. (Tr. 1483, Clare; Intervenors' Exhibit 1, pp. 7-8.)

a) To apply this safety objective, one must multiply the probability of initiation of a CRBR core disruptive accident times the conditional probability that such an accident will have dose consequences exceeding the 10 CFR Part 100 guidelines. (Tr. 2839-40, Cochran).

b) Neither Applicants nor Staff have considered the conditional probability that a CDA, once initiated, will have dose consequences exceeding the 10 CFR Part 100 guidelines, in determining whether CDAs should be within the CRBR design basis. (Tr. 1488-89, Clare; 2272-73, 2283-84, Morris).

c) Therefore, in applying the safety objective to determine whether CDAs should be within the design basis, the Staff has no basis for assigning a conditional probability that is less than 1. (Tr. 2839-41, Cochran).

d) It follows that the Staff must determine that the probability of initiation of a CDA is less than 10^{-6} per reactor year in order to meet its safety objective and exclude CDAs from the CRBR design basis. (Tr. 2841, Cochran).

e) Applicants used precisely this approach in their Reliability Program, at least as of 1976. (Tr. 2841, Cochran; Intervenors' Exhibit 1).

15. The Staff has failed to demonstrate that either the probability of CDA initiation or the probability of a CDA release beyond 10 CFR Part 100 guidelines is less than or even approaches 10^{-6} per reactor year.

a) The Staff admits that it places no emphasis on probability estimates for its analysis of the CRBR. (Tr. 2191-92, Morris.)

b) The Staff admits that its estimates of the probability of a CRBR release with dose consequences beyond the 10 CFR Part 100 guidelines is unmeasured and unquantified. (Tr. 2280-81, Morris).

c) The Staff admits that it has used no means other than "judgment" for determining whether the probability of a CDA release beyond 10 CFR Part 100 guidelines equals or even <u>approaches</u> 10^{-6} per reactor year. (Tr. 2281-82, Morris).

d) The Staff's use of engineering judgment to determine the probability of CDA initiation, or of a CDA release beyond 10 CFR Part 100 guidelines, is faulty since it fails to take into account all relevant factors which it would be prudent to consider. (Tr. 2866, Cochran.)

i) Staff Witness Morris admitted that in order to make a prudent engineering judgment, one should consider all relevant factors. (Tr. 2176, Morris).

ii) Staff Witness Morris admitted that there are examples of cases in which other techniques than engineering judgment have been used to supplement engineering judgment. (Tr. 2175, Morris).

iii) Staff Witness Rumble stated that it would be prudent to consider the results of specific failure modes/effects analysis in its engineering judgment as to the credibility of a CDA if those results of specific analyses were available. (Tr. 2185-86, Rumble).

iv) The results of specific failure modes/effects analyses are available to the Staff. (Tr. 1647-48, O'Block; 1657, 1680, 1686, Clare.)

v) The Staff has not considered the results of specific failure modes/effects analysis in its engineering judgment regarding the probability of CDA initiation. (Tr. 2178, Morris).

G. The Staff's Reliance on Similarities Between LWR and LMFBR Systems For Assurance That CDAs Will Be Sufficiently Improbable Is Misplaced.

16. The Staff relies on the assertion that the "safety functions which must be achieved for an LMFBR are not fundamentally different from the safety functions successfully implemented for LWRs" as a basis for concluding that these safety functions can be implemented to make CRBR CDAs very improbable. (Tr. 2458; 2205, Morris.) This reliance is misplaced.

a) In the proposed ATWS (Anticipated Transients Without Scram) rule for LWRs, 46 Fed. Reg. 57521, the NRC concluded that "the reliability of current reactor protective systems has not been demonstrated to be adequate and most likely is not adequate." (Tr. 2845, Cochran.)

b) The proposed ATWS Rule concludes that experience to date with LWRs suggests that the frequency of ATWS accidents, though less than once in a thousand reactor years, may not be very much less, and that such frequencies are too high. (Tr. 2846, Cochran.)

c) All proposed alternatives to mitigate the frequency and severity of ATWS events in LWRs require LWR design-specific measures which are not always directly transferable to LMFBRs. (Tr. 2661, 2846, Cochran; 2206, Morris.)

d) Even if LWR shutdown systems could be demonstrated to be adequate for LWRs, that level of adequacy would not be sufficient if applied to an LMFBR because of the differences in potential severity of ATWS events between LWRs and LMFBRs. (Tr. 2846, Cochran; 2431-32, Becker).

e) It is impossible to establish the reliability of CRBR shutdown systems relative to those for LWRs without a comprehensive failure mode and effects analysis or a fault tree/event tree analysis. (Tr. 2662, 2846, Cochran; 2232-33, Morris).

f) The Staff did not and does not intend to analyze the extent to which previously unrecognized interdependencies between various LWR reactor features have been discovered, as a basis for their conclusion that such interdependencies are very improbable for the CRBR. (Tr. 2256-57, Morris.)

g) One of the major causes of uncertainty in WASH-1400 (a comprehensive probabilistic risk assessment for LWRs) cited by the NRC's Risk Assessment Review Group was the variations between reactors, since WASH-1400 examined only one BWR and one PWR. (Tr. 2847, Cochran; 1705-07, Strawbridge).

h) There are substantially larger differences between the major safety systems, e.g., reactor shutdown systems, in a reactor of the general size and type as the CRBR and those in

LWRs, than between systems in reactors of the same LWR type. (Tr. 2847, Cochran; 1705, 1707, Strawbridge.)

j) The fact that the CRBR will have two independent, diverse, and redundant reactor shutdown systems does not afford confidence that their combined reliability will be 3-4 orders of magnitude better than that which has been achieved in existing LWR systems. (Tr. 2847-49, Cochran).

k) Staff Witness Rumble states that the 10^{-3} ATWS estimate for LWRs is not appropriate for the CRBR, but does not explain what probability estimate would be more appropriate for the CRBR (Tr. 2416-17, Rumble).

l) Staff Witness Morris admits that implementation of a particular safety function could be very different for LMFBRs and for LWRs. (Tr. 2206, Morris.)

H. The Staff Has Not Demonstrated Adequately That Its Knowledge Of Other Breeder Reactors Affords Confidence That CDAs Will Be Sufficiently Improbable In The CRBR.

17. The Staff has not demonstrated adequately that its knowledge of other domestic and foreign breeder reactors affords confidence that CDAs will be sufficiently improbable in the CRBR, as asserted at Tr. 2458.

a) Staff Witness Morris states that the Staff relies on a general understanding of fast sodium cooled reactors such as the FFTF and foreign LMFBRs, but stated previously in deposition that the Staff did not intend to use any information about the FFTF at all in the LWA proceeding. (Tr. 2207-08, Morris) (emphasis added).

b) Staff Witness Morris also stated previously in deposition that for purposes of the LWA review, there is no systematic Staff effort to take into account foreign experience with breeder reactors. (Tr. 2209, Morris).

c) Staff Witness Morris was unsure whether the general features required by the Staff for the CRBR (Tr. 2459) are part of the design basis for any other reactor. (Tr. 2210, Morris).

d) The Staff could point to no reactor which contains these general features. Staff Witness King admitted that the FFTF does not contain all these features (Tr. 2215, King).

e) Staff Witness Morris did not know what the design basis is for any reactors around the world. (Tr. 2210, Morris.)

f) Staff Witness Morris admitted that the Staff does not have a good understanding of the specific design features of other domestic or foreign breeder reactors or how such features have been implemented. (Tr. 2212-14, Morris.)

F. <u>The Staff Has Failed Adequately To Consider the Potential For CDA Initiation Resulting From Human Error at the CRBR</u>

18. The Staff has failed adequately to consider the potential for CDA initiation resulting from human error at the CRBR.

a) The Staff admits that human error could cause a undetected interdependence between various elements of the reactor, such as the two shutdown systems. (Tr. 2255, Morris).

b) The Staff admits that operator error, such as inadequate inspection, could result in a failure to maintain sufficient coolant inventory. (Tr. 2226, Morris).

c) The Staff admits that operator error could result in a failure to respond to the loose parts monitoring system. (Tr. 2227, Morris).

d) The Staff admits that the proposed design would be more conservative if the fuel failure propagation detection and prevention system did not require operator action. (Tr. 2237-38, King).

e) The Staff admits that human error could be responsible for CDA initiation conditions in both LMFBRs and LWRs. (Tr. 2263, Morris).

f) The Staff concurs with the NUREG-0572 statement that the contribution of human error to Licensee Event Reports (LERs) in general is high. (Tr. 2255, Morris).

g) The Staff claims that the potential for human error at the CRBR would not differ significantly from the potential for human error at an LWR. (Tr. 22445, Morris).

h) The Staff has not used the estimates of the high contribution of human error to LERs in any way for its conclusion that accidents caused by human error would be very improbable at the CRBR. (Tr. 2246, Morris).

j) The Staff admits that if one could estimate the probability of CDA initiation due to human error for an LWR, this estimate could possibly be extrapolated to an LMFBR.

k) Staff Witness Rumble admits that it would be helpful for the Staff to consider systematic fault tree/event tree analyses in determining the effects of human error in a generic fashion. (Tr. 2420, Rumble).

l) The Staff has not performed any systematic analyses of how human error could initiate or exacerbate an accident at the CRBR. (Tr. 2243, Morris).

m) The Staff has not analyzed the extent to which system interdependencies have been discovered in LWRs for its conclusion that they are highly or very improbable for the CRBR. (Tr. 2256, Morris). The Staff does not intend to perform such an analysis. (Tr. 2257, Morris).

n) One basis for the Staff's conclusion that CRBR accidents resulting from human error will be very improbable is the fact that after the TMI-2 accident, the NRC placed special emphasis on reviews of the adequacy of control room design, operator training, utility management, plant operating and emergency procedures; and that such a review will be carried out for the CRBR. (Tr. 2443, Morris; 2468).

o) The Staff is unaware of any decrease in the occurrence of human errors as a result of the increased NRC attention on human error problems since the TMI-2 accident. (Tr. 2260-61, Morris).

p) Therefore, there is no evidence in the record to indicate that the probability for accident initiation, including CDA initiation, from human error would not be just as high for the CRBR as for an LWR at the time of the TMI-2 accident. (Findings 18(a)-(p)).

J. Staff and Applicants Fail to Justify Their Categorization of Accidents Within Or Outside the CRBR Design Basis

19. Staff and Applicants fail to justify their categorization of accidents within or outside the CRBR design basis.

a) Both Applicants and Staff state that they determine which accidents to include within the CRBR design basis by examining a range of accidents to determine which are "credible." (Tr. 2003, 2450).

b) The Staff denies that it attaches any quantitative or qualitative probability to the word "credible." (Tr. 2173, Rumble; 2191-92, Morris.)

c) Applicants do not use quantitative probabilities or a quantitative threshold criterion for determining whether CDAs are within or outside the DBA envelope. (Tr. 2858, 1480, 1483-84, Clare).

d) When asked to define "credible," Applicants stated, "Credible means that it is of a sufficient likelihood that it should be considered in the design basis envelope." (Tr. 1653, O'Block, Dietrich, Clare, Brown, Strawbridge).

e) When asked to define "credible,"the Staff stated that its only definition of "credible" is that it is synonymous with "accidents within the design basis envelope." (Tr. 2172, Hulman; 2453).

f) Based on these definitions and the failure to use any qualitative or quantitative probabilities, neither Applicants nor Staff have provided an adequate basis for their categorization of accidents within or outside the design basis. (Findings 19(a)-(d)).

20. The Staff has failed to demonstrate that it is feasible to design the CRBR so that CDAs are "incredible."

a) A finding that it is feasible to design the CRBR so that the systems intended to prevent CDAs are "state of the art" is not the same as a finding that it is feasible to design the CRBR so that CDAs are "incredible." (Tr. 2852, Cochran).

b) A finding that it is feasible to design the CRBR so that CDAs are "incredible" requires a demonstration of the reliability of "state of the art" prevention systems individually. (Tr. 2852, Cochran; 2280, Morris).

c) A findingn that it is feasible to design the CRBR so that CDAs are "incredible" requires a demonstration of the combined reliabilities of all CRBR safety systems and their interactions. (Tr. 2852, Cochran; 2280, Morris).

d) The Staff has not quantitatively analyzed the individual or combined reliabilities of the CRBR safety systems. (Tr. 2280, Morris).

21. The Staff cannot logically reach a final determination as to whether CDAs or other accidents should be within the design basis until it has completed a detailed CRBR safety review.

a) The Staff admits that it has not yet determined whether Applicants' list of proposed design basis accidents is sufficient. (Tr. 2192-93, Morris).

b) The Staff admits that it might add to the list of design basis accidents after a detailed safety review. (Tr. 2193, Morris).

c) The Staff stated that it would probably not add CDAs to the list of design basis accidents after a detailed safety review, even if it determined that CDAs are credible (Tr. 2193, Morris), but would instead require that the design be changed. (Tr. 2195, Morris).

d) There is no evidence in the record to indicate that "changing the design" is different from "including the CDA within the design basis." Furthermore, according to the Staff's own testimony, finding CDAs to be credible would automatically place them within the design basis, since "credible" and "design basis" are considered synonymous. (Tr. 2172, Hulman).

K. <u>Applicants Have Not Demonstrated That The Likelihood of a CDA Is So Low That It Can Be Excluded From The Design Basis</u>

22. Applicants' judgment that the likelihood of a CDA is so low that it can be excluded from the design basis is based on Applicants' understanding of:

a) their general approach to the CRBR design (as described in PSAR § 15.1.1);

b) conditions under which a CDA can potentially be initiated; and

c) the CRBR's general design features (as illustrated in CRBRP-3, Vol. 1, Chapter 3) that are provided to "preclude" occurrence of CDAs. (Tr. 2857, Cochran).

23. In making their judgment that the likelihood of a CDA is so low that it can be excluded from the design basis, Applicants do not rely upon:

a) their Reliability Program (documented in PSAR Appendix C) (Tr. 2857, Cochran);

b) the probability of failure of the reactor shutdown systems or any of the general design features (Tr. 2857, Cochran; 1461, Clare);

c) tests of the reactor shutdown or shutdown heat removal or other CDA prevention systems (Tr. 2858, Cochran; 1479, Clare);

d) quantified reliability threshold criteria (Tr. 2858, Cochran; 1480, 1483, Clare);

e) probabilistic risk assessments (Tr. 2858, Cochran; 1484, Clare);

f) analysis or evaluation of designs of plants other than the CRBR (Tr. 2858, Cochran; 1684, 1727-28, Brown; 1487, Clare);

g) sufficiency or completeness of the SSR Appendix A criteria, the Denise-Caffey letter criteria, or any known set of criteria (Tr. 2856-58, Cochran; 1483, 1487-88, Clare);

h) analysis of the CDA once initiated, including Section 5 of Applicants' Exhibit 1 (Tr. 2858, Cochran; 1488-89, Clare);

j) any quantification of the failure rates of the reactor shutdown system, the decay heat removal system, the probability of rupture (larger than the design basis rupture) of the reactor vessel or pipe, or the systems designed to maintain individual subassembly heat generation and removal balance. (Tr. 1461-62, Clare).

L. <u>The Applicants' General Design Approach Does Not Provide a Basis for Excluding the CDA From The DBA Envelope</u>

24. The general design approach Applicants set forth in Chapter 15.1.1 of the PSAR (Applicants' Exhibit 8) is simply the "defense-in-depth" approach characterized by "three levels of design emphasis" (accident prevention, mitigation, and containment). (Tr. 2858, Cochran.) Applicants agreed that the

three levels of safety correspond to a "defense-in-depth" approach. (Tr. 1501, Clare).

25. Neither the PSAR sections admitted as Applicants' Exhibits 2-14, nor any other evidence in the record demonstrates that Applicants' general design approach (Applicants' Exhibit 8) ensures that it is feasible to design a reactor of the general size and type as the CRBR to make CDAs sufficiently improbable that they can be excluded from the design basis. (Tr. 2859, Cochran).

a) Applicants admit that the safety philosophy alone does not dictate which accidents are within the design basis. (Tr. 1509-10, Strawbridge).

26. The three-level design philosophy in Applicants' Exhibit 8 presents no justification for the selection of the design basis events. (Tr. 2859-62, Cochran).

a) The same three-level design philosophy was also applied to the FFTF, which in essence included the CDA within the design basis. (Tr. 2860-61, Cochran.) Applicants conceded that the three levels were applied to the FFTF, and a CDA was included in the third level. (Tr. 1502, Brown).

b) The same three-level design philosophy was also applied to the CRBR parallel design, in which "accidents involving loss of in-place coolable geometry were treated as design basis events." (Tr. 2861, Cochran). Applicants agreed. (Tr. 1503, Strawbridge).

c) A "defense in depth" safety philosophy analogous to the three-level design philosophy was applied to SEFOR. The SEFOR approach included a core disruptive accident in a manner analogous to the third level of safety. (Tr. 1501-02, Brown).

27. The fact that one can establish a general classification scheme for accidents does not insure nor provide confidence that one can properly assign accidents to the respective categories. (Tr. 2861-62, Cochran).

28. The same safety philosophy would apply whether the CDA is deemed within or outside the design basis. (Tr. 2862, Cochran.) Applicants' witness conceded this. (Tr. 1509-10, Strawbridge).

a) Applicants knew of no final list of design basis accidents for a reactor of the general size and type as the CRBR. (Tr. 1476-78, Clare, Brown, Strawbridge, Deitrich).

M. <u>Existing Knowledge of Conditions for CDA Initiation Does Not Provide A Basis for Excluding the CDA from the DBA Envelope</u>

29. In order to reasonably exclude the CDA from the design basis, one must have confidence that:

a) all important classes of CDA initiators have been identified (Tr. 2862, Cochran); and

b) identification and protection against CDA initiators ensures that the probability of a CDA is sufficiently low.

(Tr. 2862, Cochran).

30. One cannot make an affirmative finding on item 29(a) above unless:

a) all important classes of CDA initiators have been identified (Tr. 2862, Cochran); and

b) identification and protection against CDA initiators ensures that the probability of a CDA is sufficiently low. (Tr. 2862, Cochran).

31. The Staff admits it does not have a basis for judging the completeness of Applicants' list of CDA initiators. (Tr. 2863, Cochran).

32. The Staff's position regarding some of the potential CDA initiators identified by Applicants, e.g., double-ended pipe break, is not final (1982 SSR, p. II-9, Staff Exhibit 1). (Tr. 2863, Cochran).

33. Applicants concede that "[i]t is impossible...to confidently list all the important initiators before the event tree and fault tree analyses have been performed." (CRBRP Project, PRA Program Plan, June 18, 1982, p.3). (Tr. 2863, Cochran).

a) Applicants have not considered all CDA initiators, for example, simultaneous pump failure and failure of one pump where one of the piping loops was inoperable. (Tr. 1651, 1653, O'Block).

34. With regard to item 29(b) above, the mere identification of initiators and systems intended to protect against them does not preclude CDAs. (Tr. 2864, Cochran).

N. Applicants Have Not Demonstrated That the General Design Features "Preclude" the Occurrence of CDAS

35. All CDA protective systems have some failure rate. (Tr. 2864, Cochran). Applicants' witness admitted that the major safety feature intended to prevent CDAs can fail. (Tr. 1382-83, 1387, 1391, 1393, Clare).

36. To make an affirmative finding on item 30(a) above, one must consider the effects of human and design errors. (Tr. 2864, Cochran).

37. To make an affirmative finding on item 30(b) above, one must show that multiple and common mode failures cannot significantly affect the probability of a CDA. (Tr. 2864, Cochran).

38. Because multiple failures - whether common mode or otherwise - should be expected, it is essential in order to exclude CDAs from the design basis, to treat event sequences (fault trees) as well as initiating events. (Tr. 2864, Cochran).

a) Determination of the failure rates of the CDA prevention features is crucial to the question of whether a CDA is "credible." (Tr. 2864, Cochran).

b) There are combinations of failures of the shutdown heat removal systems for which one could not assure that a CDA would not be initiated. (Tr. 1596, Clare.)

39. Applicants have not quantified what is an acceptably low likelihood of CDAs. (Tr. 1497, Clare).

40. Applicants have not quantified the probability of failure of the major design features intended to prevent CDAs. (Tr. 1461-62, Clare).

41. Applicants state only that the probability of a failure of the major design features intended to prevent CDAs would be "very low." (Tr. 1462, Clare).

42. The findings and determination identified in Findings 29, 30, 36, and 37 above have not been made as they require detailed, design-specific analyses. (Tr. 2864, Cochran).

O. The Double-Ended Pipe Break Could Cause a CDA In the CRBR, and There Is No Basis for Excluding It From the DBA Envelope

43. A double-ended pipe break is considered a design basis accident for light water reactors, but not for the CRBR. (Tr. 1509, Strawbridge).

44. A double-ended pipe break could lead to a CDA in the CRBR (1982 SSR, p. II-9, Staff Exhibit 1). (Tr. 2029, Applicants' Exhibit 1.)

45. Use of sodium coolant near atmospheric pressure alone is not sufficient basis for concluding that a double-ended pipe break should not be a design basis accident for the CRBR. (Tr. 1534-35, Clare).

a) In some portions of the sodium loops the sodium may be at a pressure of approximately ten atmospheres. (Tr. 1536, Clare).

46. Use of stainless steel piping alone is not enough reason to conclude that a double-ended pipe break should not be a design basis accident for the CRBR. Stainless steel piping has been used in some light water reactors, for which a double-ended pipe break is a DBA. (Tr. 1538, Clare, Deitrich; Tr. 1540, Brown).

47. Placement of piping in nitrogen-inerted cells with low oxygen content alone is not enough reason to conclude that a double-ended pipe break should not be a DBA for the CRBR. There are a number of boiling water reactors now operating with nitrogen-inerted cells, for which a double-ended pipe break is

considered a DBA. (Tr. 1539-40, Brown).

48. The existence of a material surveillance program alone is not enough reason to conclude that a double-ended pipe break should not be a DBA for the CRBR. None of Applicants' witnesses were familiar with the material surveillance program requirements for LWRs. (Tr. 1540-41, Clare, Brown, Strawbridge, Deitrich, O'Block).

49. The assertion that CRBR piping would retain its integrity even if one or two snubbers were to fail during plant operational loadings (Tr.2030) is not based on familiarity with requirements for snubbers in light water reactors (Tr. 1542-49, Clare, Brown, Strawbridge, Deitrich, O'Block), and should therefore be given little weight.

50. Although Applicants assert that they apply more restrictive specifications for the quality of piping material and welds to CRBR than are required by the ASME Code for LWRs (Tr. 1552, Clare), there is no evidence that NRC will require Applicants to meet the more restrictive specifications. (Tr. 1555, Clare).

51. The existence of a "comprehensive quality assurance program" to assure that piping quality specifications are met

does not distinguish CRBR from LWRs. (Tr. 1552-53, Clare).

52. The existence of a "comprehensive in-service inspection program" to assure that there is little potential for initiating pipe flaws during the plant life does not distinguish the CRBR from LWRs. (Tr. 1553-54, Clare, Strawbridge).

53. Applicants have no statistical basis for their statement that worldwide operating experience with sodium systems strongly supports the overall conclusion that the likelihood of double-ended pipe ruptures is low (Tr. 1567-68, Clare), other than the fact that Applicants are unaware of any double-ended pipe ruptures that had occurred in LWRs. (Tr. 1568, Clare).

54. The leakage detection system is designed to alert the operator if there is leakage from the primary coolant piping. (Tr. 1547, Clare).

55. It is possible for the operator to ignore the signal from the leakage detection system. (Tr. 1547-48, Clare).

a) None of Applicants' witnesses were familiar with the fact that during the accident at TMI-2, the operator ignored a signal that a valve was stuck open. (Tr. 1548,49, Clare, Brown, O'Block, Strawbridge).

b) One of Applicants' witnesses testified that he was not aware that Applicants analyzed the TMI-2 accident in determining what the DBAs for CRBR should be. (Tr. 1549, Strawbridge).

P. Applicants Have No Analytical Test For Selection of DBAs and No Basis for Excluding CDAs From the DBA Envelope

56. Applicants' 1976 Reliability Program (excerpted in Intervenors' Exhibit 1) clearly indicated that reliability methodology was employed to select the limiting design basis accidents. (Tr. 2865, Cochran). Applicants admitted this was an objective of the Reliability Program submitted to the NRC staff in 1976. (Tr. 1475, Clare).

a) The "Clinch River Breeder Reactor Project Reliability Program," the Applicants' January 1976 reliability program plan, described the CRBRP reliability program activities and presented the relationship of the reliability program to the overall safety and licensing approach, as of January 1976. (Exhibit 1, p. 1.)

b) Due to the lack of precedents for LMFBR plants, the CRBRP design approach utilized reliability techniques extensively to provide a systematic determination of events to be included in the plant design basis. (Exhibit 1, p. 6.)

c) In cases where the accomodation of certain severe events are not specified in appropriate Regulatory Guides or Federal Regulations, and where licensing of LWR plants does not establish precedents, a systematic approach using reliability methodology was employed to select the limiting design basis. The remaining accidents with potential to exceed 10 CFR 100 guidelines were either in the design basis envelope of the plant or excluded from it depending on the probability of the event which initiates the accident. (Exhibit 1, pp. 6-7.)

d) The reliability program is an integral part of the overall safety and licensing approach and was used to assure and confirm the low probability of specific initiators not covered by precedent or regulations and thereby allow exclusion of these initiators from the design base. (Exhibit 1, p. 7.)

e) The initial selection of DBAs was very early in the design stage, beginning in 1974, and only minor adjustments have been made since then. (Tr. 1634, Clare).

57. Presently Applicants contend that they established the CRBR design basis accidents without the use of the Reliability Program and the adequacy of that program. This is inconsistent with Applicants' earlier assertions (Finding 56). (Tr. 2865, Cochran; 1463, Clare).

58. The concept of "engineering judgment" should not be abused in the reactor licensing process or used to hide the absence of empirical evidence or confirmatory analysis. (Tr.

2866, Cochran)

59. The Safety Analysis Group at Los Alamos National Laboratory, in its Final Report on "Reactor and Structural Systems Analysis for CRBR Licensing," prepared under contract to the NRC Staff (Tr. 2688, Cochran), made findings similar to that in item Finding 58:

> Any cavalier approach justified by the hypothetical (often equated with impossible) status of these [CDA] accidents can degenerate quickly to judgements (perhaps hunches or guesses) instead of facts or quantified certainties.

(Tr 2866, Cochran). The Los Alamos report was not addressing the specific question whether the CDA should be in the design basis, but rather made recommendations on necessary CDA analysis. (Tr. 2687-88, Cochran.)

60. Since the withdrawal of Applicants' Reliability Program as the basis for selection of the design basis events for CRBR (See Findings 56 and 57), no alternative analytical test of Applicants' hypothesis that the CDA can be excluded from the design basis has been provided. (Tr. 2867, Cochran.)

61. Findings of the "feasibility" of making CDAs "incredible for a reactor of the general size and type as CRBR" must be anchored in past experience supplemented by analytically rigorous prediction. (Tr. 2867, Cochran).

a) In sharp contrast, Applicants' conclusion that "any one of the three overall Heat Transport System (HTS) paths has the capability to independently reject the reactor decay heat" (Tr. 2025, Applicants' Exhibit 1), is based only on an "understanding" of basic physical heat generation and transfer properties and nothing more. (Tr. 1586, Clare.)

62. Applicants and Staff lack the precedent of even one substantially similar fast reactor during the licensing of which it was demonstrated that the probability of a CDA is "sufficiently low." (Tr. 2868, Cochran).

a) Applicants' testimony demonstrates that their use of terms such as "low," "very low level," "extremely unlikely," "prevent," and "high likelihood" are not clearly defined. (Tr. 1385-86, 1495-96, 1616, 1637, 1639, Clare.)

63. Applicants and Staff make a circular argument concerning CDAs: "We will require CDAs to be of low probability, hence they will be of low probability." (Tr. 2868, Cochran; 2225, Morris).

a) NRC "required" the TMI 2 core not to be severely damaged, yet it was severely damaged. (Tr. 2868, Cochran).

b) The AEC "required" that melting should occur in no more than one subassembly in the Fermi-1 core, yet there was melting in two subassemblies. (Tr. 2868, Cochran).

64. CDAs cannot be considered incredible for the CRBR, or for a reactor of the general size and type as the CRBR. (Tr. 2868, Cochran).

Q. <u>The Assumed Fission Product Release in the Site Suitability Source Term Chosen By the Staff Is Not Sufficiently Conservative.</u>

65. The history of 10 CFR Part 100 demonstrates that a very high degree of conservatism should be used.

a) The 10 CFR Part 100 Reactor Site Criteria were promulgated in 1962 with the intent to provide a substantial additional layer of conservatism above and beyond that provided by safety features designed to mitigate against design basis accidents. (Tr. 3057, Cochran).

b) The AEC determined that, even if the plant were designed to prevent and mitigate against all credible accidents, the possibility for a much more serious, though highly improbable, accident could never be completely discounted, and therefore its consequences must be considered when siting the plant. (Tr. 3057, Cochran).

c) The site suitability source term (SSST) for LWRs was developed after many years of licensing and operating experience with LWRs. (Tr. 3058, Cochran).

d) The SSST for LWRs was based upon a step-by-step analysis of a major postulated accident, one with consequences far exceeding those of any LWR design basis accident. (Tr. 3058, Cochran).

e) The SSST for LWRs was derived using the highly conservative assumptions that:

i) the coolant piping ruptures completely from high internal pressures due to uncontrolled internal heat generation - this could occur only if:

(1) reactivity control mechanisms fail to function,
(2) high pressure relief systems fail to perform, and
(3) pressures exceed rupture limits of the piping material;

ii) decay heat is sufficient to increase fuel temperature to the melting point; and

iii) safeguards systems provided to flood or spray the core with water are either inoperative or insufficient to keep fuel temperatures from rising.

(Tr. 3058-59, Cochran).

f) This postulated SSST accident (in Finding 65(e)) is orders of magnitude larger than the limiting design basis accident for LWRs. (Tr. 3059, Cochran).

g). Additional conservatisms were built into the site suitability analysis for purposes of determining the extent of the fission product release from the SSST accident (Finding 65(e)) and the amount released to the environment. (Tr. 3059-60, fn. 3, Cochran).

h) Very conservative procedural or analytical methods are employed to calculate doses to individuals at the exclusion area and low population zone boundaries. (Tr. 2558-79, Attachment A to Staff's Exhibit 3).

j) The AEC concluded that the net effect of the assumptions and approximations (described in Finding 65(e) is believed to give more conservative results (greater distances) than would be the case if more accurate calculations could be made. (Tr. 3060, Cochran).

k) The Commission explicitly recognized that even more conservatism is required in siting reactor types with no previous licensing experience:

> [F]or reactors that are novel in design and unproven as prototypes or pilot plants, it is expected that these basic criteria will be applied in a manner that takes into account the lack of experience.

10 CFR §100.2(b). (Tr. 3061, Cochran).

l) The Commission envisioned that an applicant could trade off the use of engineered safeguards for site isolation only when the safeguards were "extensive and well proven," based on operating experience from plants already licensed. (Tr. 3061-62, Cochran).

m) The Advisory Committee on Reactor Safeguards believed that novel or unproven reactor types, which necessarily lacked previous licensing experience, "belong at isolated sites -- the degree of isolation required depending on the amount of experience which exists." (Tr. 3061-62, Cochran).

66. The assumed CRBR site suitability fission product release is insufficiently conservative whether or not CDAs are considered credible.

a) The Staff asserts that the SSST set forth in the 1982 Site Suitability Report (Staff Exhibit 1) at III-11 is nonmechanistic, and is directly analogous to the LWR source term, modified only to include the release of 1% of the plutonium from the core. (Tr. 3062, Cochran).

b) If CDAs are credible accidents, the Staff's source term clearly does not bound the consequences of a major CDA. (Tr. 3063, Cochran). The Staff has admitted it would have to redo its analysis of the source term if CDAs were considered credible. (Tr. 3065-66, Cochran 2274, Morris).

c) When the Staff derived an SSST for Applicants' Parallel Design, in which a CDA was considered a credible accident within the design basis, the assumed fission product release included 10% of the plutonium. (Tr. 3063, Cochran).

d) Even larger SSSTs (than that referenced in Finding 66(c) have been used in the past to bound CDAs in other reactors, e.g., EBR-II and SEFOR. (Tr. 3063-64, Cochran).

e) The Staff has done no analysis of the potential consequences of CDAs. (Tr. 3065, Cochran).

f) Applicants have not performed the necessary analysis of whether the Staff's source term is sufficiently bounding if CDAs were considered credible. (Tr. 3066-67, Cochran).

g) If it is shown that CDAs are credible accidents for a reactor of the general size and type as the CRBR, both Staff and Applicants will have to redo their source term analyses to determine whether and how the source term should be revised. (Tr. 3067-68, Cochran; Tr. 2274, Morris).

h) Evidence from the treatment of other reactors (Finding 66(d), and from the Staff's own preliminary analysis of the CRBR Parallel Design (Finding 66(c)), indicates that the assumed plutonium release from the core would have to be increased by at least a factor of 10 if the CDA were deemed credible. (Tr. 3058, Cochran).

j) The Staff may not treat this first-of-a-kind reactor as it would a tested, proven LWR design. 10 CFR §100.2 (b). (Tr. 3068, Cochran).

k) The Staff must apply additional conservatisms to a novel reactor of the general size and type as CRBR to take into account the lack of breeder reactor licensing experience. (Tr. 3068, Cochran).

l) The Staff must factor in these additional conservatisms (Finding 66(k)) either by selecting a more isolated site than it would for a tested design or by requiring extensive and well-proven engineered safeguards. (Tr. 3068, Cochran).

m) The Staff should not extrapolate directly from the LWR source term without substantial additional margins of safety to account for the uncertainties inherent in the novel design of the CRBR. (Tr. 3068-69, Cochran).

n) The Staff and Applicants should not rely on engineered safeguards which have not been proven or previously licensed and which will not even be fully scrutinized until a later licensing stage. (Tr. 3069, Cochran).

o) The source term chosen for a reactor with unusually severe potential accident consequences, on the basis of incomplete analysis at the LWA stage, should be large enough to bound any accidents which the Staff may later determine to be credible after a full safety review. (Tr. 3069, Cochran).

p) The design approach for the CRBR must be of an enveloping nature and sufficiently conservative to account for further design modifications and uncertainties. (Tr. 3069, Cochran).

q) The Staff used to take such a conservative approach (Finding 66(p)) elsewhere in the siting analysis by lowering the organ dose guideline values by a factor of 10 (now only 2) during the construction permit and LWA review stages. In applying the same principle here, the Staff should increase its plutonium release fraction by a factor of at least 10 to account for the substantial possibility that CDAs will be found credible after a full safety review. (Tr. 3070-71 Cochran).

r) A highly conservative approach similar to that used in LWRs is necessary to achieve Part 100's objective of protecting against excessive exposure doses from conceivable though highly improbable accidents. (Tr. 3071-72, Cochran).

s) The maximum capacity for harm from an LMFBR accident has been estimated to be an order of magnitude greater than that from an LWR. (Tr. 3072, Cochran).

t) The higher capacity for harm from LMFBR accidents is not reflected in Staff's choice of the source term, namely the LWR source term plus 1% of the plutonium. (Tr. 3072, Cochran).

u) Applicants' own analyses of CDAs have postulated the release of up to 10% of the plutonium from the core. (Tr. 3072, Cochran).

v) A fuel release fraction of 10% of plutonium would be appropriate for the source term even if CDAs are not within the design basis. (Tr. 3072, Cochran).

R. The Staff's Proposed Source Term Does Not Include the Pressure and Thermal Effects Associated with Core Meltthrough, and Is Therefore Non-Conservative.

67. The Staff's proposed source term is premised on the occurrence of a CDA. (Tr. 3073, Cochran).

a) In a site suitability analysis one should conservatively assume, as Applicants have, that all accident sequences leading to a CDA would lead to whole core involvement. (TR. 3073, Cochran).

b) In a site suitability analysis one should conservatively assume that for a CDA the molten fuel will penetrate through the bottom of the reactor vessel and guard vessel. (Tr. 3073-74, Cochran).

c) A core meltthrough (as described in Finding 67(b)) was the basis for the NRC Staff's radiological site suitability source term analysis for the FFTF. (Tr. 3074, Cochran).

68. Once meltthrough of the reactor vessel and guard vessel occurs, all of the available sodium in the reactor vessel and primary loops (approximately 1.1 million pounds) would very likely be dumped into the reactor cavity. (Tr. 3074, Cochran).

a) The sodium released from the reactor vessel would be expected to result in sodium fires and interaction with the concrete in the reactor cavity, resulting in overpressurization and high thermal loadings of the secondary containment. (Tr. 3074, Cochran).

b) Finding 68(a) is consistent with Applicants' predicted progression of a core melt scenario, described in CRBRP-3, Vol. 2, at pp. 3-18 - 3-26. (Tr. 3074, Cochran).

69. The Staff's SSST analysis with regard to the containment evaluation ignores the effects of overpressurization and thermal loading in the containment. (Tr. 3075, Cochran; Tr. 2309, Eltawila).

a) The Staff's SSST analysis should not have ignored the pressure and thermal loading implications of a CDA with core meltthrough, since it is based on a CDA. (Tr. 3075, Cochran).

b) The Staff's containment evaluation incorrectly models the actual containment that is being proposed for the CRBR. (Tr. 3075, Cochran).

i) The Staff's SSST analysis assumes that radiological releases to the environment, even from the most severe accident, will only occur via annulus filtration and bypass leakage of 0.001% per day. (Tr. 3075, Cochran).

ii) Applicants have proposed a system whereby, in a CDA, radioactivity would be released directly from the secondary containment to the environment through filtered vents. (Tr. 3075, Cochran).

iii) The Staff has previously suggested that, following an accident, containment integrity need be maintained for only 24 hours without venting. (Tr. 3075, Cochran). The Staff has admitted that the period during which venting should be prohibited might be reduced as a result of current Staff review of that proposed criterion. (Tr. 2282, Morris).

iv) Applicants have indicated they consider 10 to 12 hours a sufficient delay prior to venting. (Tr. 1880, Clare).

v. The Staff now assesses the suitability of the CRBR site based upon a containment design with no vents, but includes venting to accomodate a CDA, the very accident from which the SSST is derived. (Tr. 3076, Cochran).

70. Rather than provide a second level of defense (Tr. 3059, Cochran), the approach described in Finding 69(b) means that in this regard the site suitability analysis is in fact less conservative than the supposedly more realistic accident analysis for the CRBR plant. (Tr. 3076, Cochran).

71. Based on the foregoing (Findings 65-70), the Staff's proposed SSST for a reactor of the general size and type as the CRBR is inadequate because:

a) the assumed fuel release fraction is insufficiently conservative; and

b) it does not properly consider the pressure and thermal effects associated with core meltthrough. (Tr. 3076, Cochran).

S. Staff Has Not Correctly Performed the Dose Calculations in the SSST Analysis.

72. In the 1982 SSR the Staff has calculated the whole body, thyroid, lung, and bone doses at the exclusion area and low population zone boundaries (1982 SSR (Staff Exhibit 1), Table IV, p. III-11) for purposes of comparing these against dose guidelines as required under 10 CFR Part 100. As shown below, these calculations are in error in at least the following respects:

a) failure to consider the dose "from the entire passage of the radioactive cloud;"

b) failure to use conservative values for the plutonium isotopic concentrations that may be utilized in a reactor of the general size and type as the CRBR;

c) failure to consider all isotopes of interest;

d) failure to use current dosimetric and metabolic models and properly to calculate internal organ exposures;

e) failure to consider all pathways (Tr. 3126-3127, Morgan); and

f) failure to consider the entire life of the exposed individual. (Tr. 3173-74, Morgan).

73. The site suitability source term (SSST) analysis fails to consider the dose from the entire passage of the radioactive cloud.

a) 10 CFR §100.11(a)(2) requires that the low population zone (LPZ) outer boundary dose be calculated for the radioactive cloud "during the entire period of its passage." (Tr. 3127, Morgan; Tr. 2351, Bell).

b) The Staff's LPZ dose calculations, presented in the 1982 SSR (Staff Exhibit 1), Table IV, p. III-11, were truncated at the end of 720 hours (30 days). (Tr. 3127, Morgan).

c) The Staff admits that emissions from the postulated accident, for purposes of conducting the site suitability analysis under 10 CFR Part 100, continue after the 30 day period. (Tr. 2353, Bell).

d) The Staff performed calculations to determine the effect of including emissions beyond 30 days; i.e., the entire passage of the cloud. (Tr. 2353, Bell).

i) The Staff found that in the case of LWRs, the dose contribution beyond 30 days is negligible.

ii) In the case of the CRBR, the doses were found to be significantly larger for a puff release at the end of 30 days (considered by the Staff to be the worst case condition), than doses calculated for the first 30 days. (Tr. 2399, Bell).

e) While emissions from LWRs (and therefore dose consequences) after 30 days are negligible, the Staff admits that in the case of LMFBRs (e.g. CRBR) you cannot consider them negligible. (Tr. 2399, Bell).

f) The LPZ doses calculated by the Staff, which include the emissions beyond 30 days, are as follows:

Table 1

Organ	LPZ dose (rem) (0-30 days)	LPZ dose (rem) (0-30 days plus Puff Release)
Whole body	0.34	0.47
Thyroid	6.8	12
Lung	0.37	1.6
Bone	9	38
Bone Marrow	2.1	9.1
Bone Surface	27	115
Liver	0.98	4.1
Skin	1.3	1.5

(Tr. 3128, Morgan).

For purposes of comparisons the values on the left in Table 1 above represent the calculated values where the calculation was truncated at 30 days. (Tr. 3128, Morgan). The values in this column, for whole body, thyroid, lung and bone, after rounding off, were reported in the 1982 SSR (Staff Exhibit 1), Table IV, p. III-11).

g) In the treatment of emissions beyond 30 days (right hand column in Table 1 (Finding 73(f) above), the Staff assumed that at the end of 30 days the emissions remaining in the containment were essentially instantaneously released (actually released over a 1 hour period), or "puffed," to the environment through the annulus filtration system. (Tr. 2356-2357, Hulman and Bell).

h) The Staff in response to Interrogatory 33 in NRC Staff's Supplemental Answers to NRDC's Twenty-Sixth Set, August 5, 1982, p. 14, used the "puff release" calculational method to treat emissions beyond 30 days. (Tr. 3128, Morgan). No other methodology was offered or suggested.

j) The Staff treatment of the "puff release" (Findings 73(g) and 73(h) above) is appropriately conservative for purposes of a 10 CFR §100.11 site suitability analysis; it is more appropriate and more realistic than calculations which do not consider emissions after a 30-day period.

i) 10 CFR §100.11(a)(2) requires consideration of exposure to the radioactive cloud resulting from a fission product release during the entire period of its passage. The Staff acknowledged this. (Tr. 2350-51, Bell).

ii) Staff witness Bell testified during cross-examination by Intervenors that the calculation including the puff release, while "very conservative," incorporates the "appropriate degree of conservatism" with respect to treatment of post-30-day releases. (Tr. 2354, Bell).

iii) Mr. Bell also stated that the calculation which included the puff release was "more appropriate" and "more realistic" than calculations that do not consider any emissions after a 30-day period. (Tr. 2355, Bell).

iv) Mr. Bell's later, contradictory testimony during cross-examination by counsel for Applicants (Tr. 2403-04, Bell) should not be credited, as it was elicited by leading questions from counsel for a non-adverse party.*/

k) Since the Staff treatment of the puff release was appropriately conservative (Finding 73(j), the values in the right-hand column of Table 1 (Finding 73(f) above are more appropriate and should be tentatively substituted for the dose calculations reported in the 1982 SSR; i.e., those identified in the left-hand column of Table 1 above (0-30 day truncated). [Errors in calculating the values in the right-hand column of Table 1 still must be corrected further as indicated below.]

74. The SSST analysis fails to utilize conservative values for the plutonium isotopic concentrations that may be utilized in a reactor of the general size and type as the CRBR.

a) In calculating the SSST dose at the exclusion and LPZ boundaries, the Staff assumed that the plutonium had the following isotopic concentrations (weight %):

1% Pu-238

74% Pu-239

20% Pu-240

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*/ See comments of Chairman Miller, Tr. 2425:

> Judge Miller: [W]e always invoke the rule that nobody can cross-examine his own witness, and we look through the form to the substance as they do in federal court.
> We know very well that in this matter, that the Applicants and Staff positions are very similar. That's why we wouldn't let him lead.

5% Pu-241

0% Pu-242

(Tr. 3128, Morgan).

b) The Staff has no documented basis for this choice of plutonium isotopic concentrations. (Tr. 2346, Bell).

c) Without any apparent logical basis, the Staff claims to have "[come] up with those concentrations by taking the total number of curies for 1121 megawatts and worked backwards to calculate the mass and...ratio percentages." (Tr. 2346, Bell). "These source tables were ginned up a while back" (Tr. 2346, Bell)..."by people who had done previous calculations on Clinch River." (Tr. 2347, Bell). The Staff worked backwards from a calculation of total curie release made some 5 or more years ago, for the CRBR homogeneous core fueled with recycled LWR fuel.

d) The isotopic concentration of Pu-238 and Pu-241 are controlling in terms of bone dose as can be seen from the Hazard Index calculated in Table 2 below.

Table 2

Isotope (Pu-i)	Weight%	Weight % Normalized to Pu-239	Curies/ gram	(A) Ci Pu-i/ Ci Pu-239	(B) Bone Surf. Dose Norm. to Dose Due to Pu-239	(A)x(B) Hazard Index
Pu-238	1	0.0135	16	3.5	0.81	2.8
Pu-239	74	1	0.062	1	1	1
Pu-240	20	0.27	0.22	0.96	1	0.96
Pu-241	5	0.068	120	130	0.019	2.35

(Tr. 3129-30, Morgan).

e) While Staff's choice of Pu isotopic concentrations is more conservative than Applicants', neither is conservative compared to high burnup LWR fuel, e.g., burnup on the order of 33,000 Mw-d/MT (or higher). This can be seen from the columns labeled 1-4 in Table 3 below:*

TABLE 3

CALCULATED PLUTONIUM COMPOSITION - PERCENT

	1 Pu Recovered From Spent U Fuel	2 Pu After One 4-year Recycle	3 Pu After Two 4-year Recycles	4 Pu Recycle Model BWR
^{238}Pu	1.9	3.46	4.87	3.4
^{239}Pu	57.9	38.2	29.4	41.7
^{240}Pu	24.7	29.4	33.5	29.2
^{241}Pu	11.0	17.2	17.4	15.2
^{242}Pu	4.4	11.7	14.9	10.4
Pu_f*	68.9	55.4	46.8	57.0

*$Pu_f = {}^{239}Pu + {}^{241}Pu$

f) Referring back to Table 2 (Finding 74(d) above), the hazard index after two four-year recycles (Column 3 in Table 3

* This table of Pu isotopic concentrations is taken from USNRC, "Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors," NUREG-0002, Vol. 3, p. IV C-70. Similar values are reported by Cullingford, Hatice S., "Alternatives to Proposed Replacement Production Reactors," LANL, LA-8867, June 1981, p. 6.

(Finding 74(e) above) is no longer 2.8 for plutonium-238, but is now 34; and the index for plutonium-241 has risen from 2.35 to 20.6. That is, if an accident in the future releases breeder fuel, the cancer risk from plutonium is 55 times greater from Plutonium-238 plus plutonium-241, than from plutonium-239. (Tr. 3130, Morgan).

g) Accounting for the reduced fissile content of mixed oxide (MOX) fuel as it is recycled further increases the assumed plutonium release and bone surface dose in the SSST analysis by a factor of 1.7.** (Tr. 3131, Morgan).

h) Making the corrections identified in Findings 74(f) and 74(g) the total bone surface dose assuming recycled MOX (Column 3 in Table 3 (Finding 74(e) above) is 5.6 times greater than the bone surface dose calculated by the Staff for the plutonium isotopic concentrations assumed by the Staff (See Finding 74(a) above).***

j) DOE plans to construct a Developmental Reprocessing Plant (DRP) for the purpose of reprocessing and recycling CRBRP fuel (USNRC, Draft Supplement to FES CRBR, NUREG-0139,

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** The problem is further compounded because the hazard of plutonium-238 relative to plutonium-239 under certain circumstances is several orders of magnitude greater than unity (see K.Z. Morgan, W.S. Snyder, and M.R. Ford, Health Physics 10, 151-169 (1964) (Tr. 3132, Morgan).

*** Morgan's estimate that the overall hazard would be 50 times greater than that assumed by the Staff (Tr 3130 and 3172, Morgan) is in error due to misplaced decimal point (56 instead of 5.6) and conservative roundoff (50 instead of 56). See attached Affidavit of Karl Z. Morgan.

Supplement No. 1, p. D-11). (Tr. 3132, Morgan).

k) It is appropriate to assume that CRBR will be fueled with recycled (LWR or LMFBR) MOX with the higher concentrations of Pu-238 and Pu-241 comparable to those in column 3 in Table 3 (Finding 74(e) above) and that the curie levels for these isotopes should be further increased because of the lower fissile content. (Tr. 3132, Morgan).

l) The current CRBR application is based upon an initial fuel loading of low Pu-240 grade fuel. (Tr. 1833, Strawbridge).

m) Both the Applicants and the Staff admit, however, that the project at some later time may decide to change to some other type of fuel (Tr. 1833, Strawbridge; Tr. 2348-2349, Hulman); e.g., recycled MOX. Furthermore, Applicants do not know what type of fuel should be considered regarding a reactor of the general size and type as the CRBR (Tr. 1833, Strawbridge).

n) For purposes of determining the suitability of the site under 10 CFR §100.11 for a reactor of the general size and type as the CRBR, the analysis should be based on the types of fuel (and their isotopic concentrations) that are likely to be utilized during the lifetime of such a reactor, and should not be limited to the fuel composition currently proposed for the first core of the CRBRP.

o) The Staff's and the Applicants' treatment of the plutonium isotopic concentrations is inappropriate for the purpose of assessing the suitability of the site under 10 CFR § 100.11. When appropriate plutonium isotopic concentrations are used the plutonium bone surface (and bone) doses are increased by a factor of approximately 5.6 over the doses calculated by the Staff. (Tr. 2880, 3130, 3172, Morgan; See also attached Affidavit of Karl Z. Morgan).

p) As can be seen from Table 1 (Finding 74(f) above), correcting this factor would lead to a bone surface dose at the LPZ (115 x 5.6 = 644 rem) exceeding the guideline value for bone surface dose recommended by the Staff for use at the CP (or LWA-1) stage (150 rem) by a wide margin.

q) As indicated in Table 4 below, the Staff's guideline values are slightly exceeded even if the LPZ dose is truncated at 30 days (151 rem).

TABLE 4

Organ	LPZ dose (rem)* (0-30 days)	LPZ dose (rem)* (0-30 days plus puff release
Bone Surface	151	644

* These bone surface dose values are found by multiplying the bone surface dose values in Table 1 by 5.6.

75. The SSST analysis fails to consider all isotopes of interest.

a) In the Staff's SSST analysis reported in the 1982 SSR (Staff Exhibit 1) consideration is given to the dose contributions of only the following isotopes:

I-131	Kr-83m	Xe-131m	Pu-238
I-132	Kr-85m	Xe-133m	Pu-239
I-133	Kr-85	Xe-133	Pu-239
I-134	Kr-87	Xe-135m	Pu-240
I-135	Kr-88	Xe-135	Pu-241
I-136	Kr-89	Xe-137	
		Xe-138	

b) The Staff performed an analysis that included all transuranic elements (Tr. 2358, Bell) and covered the 30 day period plus a puff release. (Tr. 2359, Bell). These calculations were not reported in the 1982 SSR. (Staff Exhibit 1).

c) Including these additional isotopes (Finding 75(b)) increases the LPZ doses indicated in Table 1 (Finding 73(f)) above as follows:

TABLE 5

Organ	LPZ dose (rem) Without all transuranics	 With all transuranics
Bone Surface	115	119
Lung	1.6	3.37
Liver	4.1	4.52

d) While these corrections (Finding 75(c) taken alone are not significant, they are significant when taken in combination with other corrections. For example, applying the 3.4% correction (119 ÷ 115=1.034) to the LPZ dose without the puff release in Table 4 (Finding 74(q)) above, the bone surface dose is increased from 151 rem to 156 rem.

76. The SSST analysis was not performed using current dosimetric and metabolic models and the analysis failed to properly calculate internal organ exposures.

a) The Staff used the same bone and lung dose commitment factors (DCF) for plutonium isotopes in the SSST analysis (1982 SSR, Staff Exhibit 1) that Staff was using in 1976. (Tr. 3134, Morgan).

b) These bone and lung dose commitment factors, computed in NUREG-0172, were based on the dosimetric and metabolic models of ICRP Publications 2, 6 and 10. (Tr. 3134-3135, Morgan).

c) There are several discrepancies in the old ICRP methodology that have been corrected in the newer models, including increasing the quality factor for alpha irradiation from 10 to 20 (Tr. 3163, Morgan); defining the bone marrow and bone surface as the critical tissues (organs of interest) rather than treating the entire skeleton as the critical organ; and including the dose to the organs of interest from radionuclides in surrounding organs. (Tr. 2957-2958, Morgan; Tr. 1915, McClellan).

d) Using the newer dosimetric and metabolic models employed in ICRP-23 and -30, the lung, bone, and liver doses from plutonium (and other transuranic elements) can be expected to differ (in some cases significantly) from the doses calculated using ICRP-2 methodology. (Tr. 3135, Morgan).

e) Applicants' witness Thompson testified that the dosimetric and metabolic models employed by ICRP-30 are appropriate for purposes of radiation protection including calculating organ doses to the bone surface, thyroid and lung. (Tr. 1902-03, Thompson).

f) Applicants' witness Thompson testified that the dosimetric and metabolic models included in ICRP-30 reflect a more up-to-date view of our knowledge than those in ICRP-2. (Tr. 1903, 1907, Thompson).

g) While the Staff calculated the bone and lung doses using the older ICRP dosimetric and metabolic models (Findings 76(a)-(b)) and reported the bone dose, rather than the bone surface and bone marrow doses, in the 1982 SSR (Staff Exhibit 1)at Table IV, p. II-11, the Staff calculated the bone marrow, bone surface and liver doses based on dose conversion factors reported in NUREG/CR-0150 which are based on the newer ICRP models (e.g., ICRP-30). (Tr. 2360-2361, Bell; 2389-2390, Branagan).

h) Based on the Findings above (76(a)-(g)), the bone and lung doses reported in the 1982 SSR at Table IV, p. III-11 (Staff Exhibit 1), and reproduced in column 1 of Table 1 (Finding 2(f) above), are inappropriate for site suitabililty analysis. The bone surface and bone marrow doses reported in Tables 1, 4, and 5 above, which were derived using the DCF's from NUREG/CR-0150, based on the newer ICRP models, represent the more appropriate dosimetric and metabolic methodology.

j) The Staff testified that the lung dose at the LPZ would be about 8 times higher if the newer ICRP models were utilized. (Tr. 2320, Branagan).

k) Correcting the Staff's lung dose values, reproduced in Table 5 (Finding 75(c) above), by this factor of 8 gives a value of 27 rem (8x3.37=27) to the lung at the LPZ boundary for the 30-day plus puff release.

l) The 27 rem value is the more appropriate value for the lung dose at the LPZ because this calculation is based on the entire passage of the cloud and the newer dosimetric and metabolic models. (Findings 73(j) and 76(a)-(h) above).

m) This value of 27 rem to the lung exceeds the 4.5 rem to the lung guideline value (at the CP/LWA stage) proposed by the Staff in the 1977 SSR.

n) The LWA/CP guideline value of 7.5 rem, a reduction by a factor of 10 from the OL value, is more appropriate than the 35 rem value now recommended for the reasons stated in Findings 76(a)-(m).

77. The Staff fails to consider all pathways of interest.

a) In the site suitability analysis, the Staff's estimates of dose rates and dose commitments are based on an assumption that no fallout occurs during radionuclide transport outside the containment. (Tr. 2428, Bell).

b) This assumption increases the dose contributions due to direct inhalation and direct irradiation from the passing cloud (immersion) (Tr. 2561, Attachment A to Staff Testimony,

Exhibit 3) and eliminates the dose contributions from ground shine, resuspension and dietary pathways.

c) The Staff assumed the assumption in Finding 6(a) would lead to a measure of conservatism in the computation of the dose from sodium-24 (Tr. 2995, Morgan), but this is not necessarily correct. (Tr. 3162, Morgan).

d) Based on Findings 6(a)-(c), the Staff conclusions that Na^{22} and Na^{24} isotopes would not appreciably alter the calculated doses and were of little significance insofar as radiological effects were concerned (Tr. 2498, Bell), and the calculation of the relative toxicity of sodium-24 (Tr. 2501, Bell) and conclusions derived therefrom (Tr. 2501-02, Bell) are not necessarily correct.

78. The Staff failed to consider the entire life of the exposed individual by integrating the dose beyond 50 years.

a) The Staff's estimates of whole body and internal organ doses to the maximally exposed individual at the exclusion area and LPZ boundaries were calculated based on the assumption that a person exposed in an accident will die 50 years later. The Staff, in effect, assumes that when the various isotopes of plutonium are fixed in the skeleton and/or in the endosteal and periosteal surface tissues of the trabecular bone that in such case this person is going to die at age 50, if he was exposed at a very early age. (Tr. 3173, Morgan).

b) Fifty years is an appropriate period of integration for doses involving occupational exposure. However, for purposes of assessing the suitability of the CRBR site an 80 year period should be utilized to reflect the fact that members of the public can be exposed at a much earlier age. (Tr. 3174, Morgan).

c) The Staff's estimates of the bone, bone marrow and bone surface doses should be increased by a factor of 1.5 to correct for the Staff's underestimate of the longer age (80 years rather than 50 years) of the maximally exposed individual. (Tr. 3170-3171, Morgan).

d) Applying this correction to the Staff's estimate of the bone surface dose at the LPZ gives:

	LPZ dose (rem) (0-30 day plus puff release)
Bone Surface Dose (80 yrs)	119 x 1.5 = 178

where 119 in the equation above is the 50 year dose estimated by the Staff considering all transuranic elements (See discussion at Finding 76(c) above).

e) The Staff's treatments of the age of the maximally exposed individual and the isotopic concentrations of plutonium are in error. Applying the factor of 1.5 to correct the assumed age of the maximally exposed individual, and correcting for the isotopic concentration of recycled MOX fuel, (See Table 4 (Finding 74(q) above) gives:

	LPZ dose (rem) (0-30 day plus puff release)
Bone Surface Dose (80 yrs)	644 x 1.5 = 966

f) Correcting also for contribution from transuranics other than plutonium (3.4%) (Finding 75(d)), the 80-year bone surface dose would be 966 x 1.034 = 1000 rem, well above the guideline value recommended by the Staff.

T. <u>The Factor of Two Reduction Used by Staff to Lower the Lung and Bone Dose Guidelines at the CP and LWA Stages Does Not Account for Uncertainties in Dose Models and Radiological Risks</u>.

79. In the 1977 SSR, the Staff used a factor of 10 to reduce the dose guidelines for the lung and bone dose at the CP and LWA stages. This factor of 10 was the product of two factors:

a) a factor of about 2 to take into account uncertainties in final design detail, meteorology, new data and calculational techniques that might influence the final design of engineered safety features or the dose reduction factors allowed for those features; and

b) a conservative factor of 5 to take into account uncertainties in dose and health effects models. (Tr. 3081, Cochran).

80. In the 1982 SSR (Staff Exhibit 1) (p. III-9), the Staff reduced this uncertainty factor from 10 to 2, claiming that the factor of 5 to take into account uncertainties in dose and health effects models is no longer needed. (Tr. 2513-14, Branagan).

81. The claim in Finding 80 is not supported by the evidence, since:

a) there is a factor of 2 uncertainty in the meteorology alone (see Finding 82), and

b) the uncertainty in the estimates of lung and bone surface doses due to plutonium (which is controlling) exceeds a factor of 10 (see Finding 83 below).

82. Staff witness Spickler testified that the meteorological chi over Q values differ by a factor of 2 between the 1977 SSR and the 1982 SSR. (Tr. 2394, Spickler).

83. The adequacy of the current Federal radiation protection standards for plutonium and other transuranic elements has been a matter of considerable debate for a number of years. (Tr. 3081, Cochran). In this regard, there are several examples which evidence the uncertainty cited in Finding 81(b) above:

a) The first example is based on the arguments set forth by Dr. Karl Z. Morgan in the American Journal of Industrial Hygiene (August 1975).

i) The current plutonium-239 standard (based on ICRP-2) was established using 0.1 microcuries of radium-226 as the reference standard. (Tr. 3142, Morgan; Tr. 2084, McClellan, Healy and Thompson).

ii) Deriving the bone surface dose directly from the radium-226 standard based on the approach of Morgan, K.Z., American Journal of Industrial Hygiene, August 1975, (Tr. 3141, Morgan) is a preferred methodology for estimating the bone surface dose due to plutonium exposure and for establishing the maximum permissible bone (and bone surface) exposure levels. (Tr. 2960-2961, 3139-3142, Morgan).

iii) Applying Morgan's approach would increase the Staff's estimate of the bone dose by a factor of 240. (Tr. 3141, Morgan). By the same token, current NRC standards for plutonium exposure are too high by a factor of 240. (Tr. 3141, Morgan; Tr. 3082, Cochran).

iv) In order to provide adequate protection to the public (and radiation workers), one should reduce the current plutonium standard by a factor of 240, or alternatively increase the quality factors used in calculating the bone dose (in rems) by the same factor of 240. (Tr.3141, Morgan; Tr. 3082, Cochran).

v) Applicant's witness Thompson testified that ICRP-30 considered the factors of concern to Morgan, e.g., problems in the dosimetry of plutonium, but did (not) employ the numbers which Dr. Morgan suggested (Tr. 1912-1915, Thompson, McClellan).

vi) Using the dosimetric and metabolic models employed in ICRP-30 as a reference, and accepting Morgan's thesis (Finding 83(a)) the quality factors used in the ICRP-30 methodology would have to increase by a factor of

80 (240÷3) in order to be fully consistent with the numerical result under the Morgan thesis.

vii) Applicants' witnesses claimed that the difference between the ICRP-2 and ICRP-30 methodologies is a result of many counterbalancing changes, but the total net numerical effect can be ascribed to an increase in the quality factor from 10 to 20, which applies to all alpha-emitters and is based on no considerations of radionuclide distribution within the bone. (Tr. 2085, McClellan, Healy, Thompson).

viii) The claim by Applicants' witnesses in Finding 83(a)(vii) is incorrect as evidenced by the factor of 3 difference between the bone dose (calculated assuming quality factor of 10, as in ICRP-2) and the bone surface dose (calculated assuming a quality factor of 20, as in ICRP-30) estimates made by the Staff and reproduced in Table 1 (Finding 73(f) above), where 97% (115÷119) of the exposure is due to plutonium isotopes (see Table 5, Finding 75(c)).

b) A second example of possible nonconservatism is the hypothesis of E.A. Martell that the principal causal factor in tobacco-related carcinoma is a result of inhalation of Po-210 (an alpha emitter) in cigarette smoke, often referred to as the "warm particle hypothesis." (Tr. 3082-3083, Cochran).

i) With regard to Martell's hypothesis, it is noted in a series of Letters to the Editor appearing in the <u>New</u>

England Journal of Medicine Vol. 307, 29 July 1982, at pp.309-313, that the localized distribution of Po-210 in the bronchial region of the lung "now appears to be 1000 times more carcinogenic than gamma radiation -- as compared to the factor of 10-20 currently assumed." (Tr. 3083, Cochran). Applicants offered no evidence concerning this hypothesis. Staff's witnesses had virtually no familiarity with it. (Tr. 2336, Branagan).

ii) Witness Cobb also cites the Po-210 work as part of the basis for his view that "present and proposed standards or guidelines for plutonium and other alpha-emitting radionuclides like americium and uranium may be seriously inadequate to protect the public." (Tr. 3101-3102, Cobb).

c) A third example of possible nonconservatism is the evidence presented by Dr. John C. Cobb (Tr. 3101-3109, Cobb) to the effect that present and proposed standards or guidelines for plutonium and other alpha-emitting radionuclides like americium and uranium may be seriously inadequate to protect the public. (Tr. 3101, Cobb).

i) Cobb's concern was based on the Findings of recent research in four related areas:

(1) The findings of our EPA-contracted study of plutonium burdens in the post-mortem tissues of people who had lived near the Rocky Flats plutonium weapons facility.

(2) The findings of several epidemiological studies showing an excess of cancer mortality and incidence in the areas near to and downwind from Rocky Flats.

(3) The findings of animal experiments suggesting that at very low dose rates, alpha-emitters like plutonium-239 and polonium-210 are very much more carcinogenic than had previously been suspected, perhaps by as much as a hundred times.

(4) The findings of animal experiments showing that plutonium and other alpha-emitters cause mutations and genetic defects as well as cancers. (Tr. 3102, Cobb).

ii) Cobb concluded, based on his findings (Tr. 3103-3105, Cobb) that "we may have <u>under</u>estimated the toxicity of plutonium by a large factor and we have probably <u>over</u>estimated our ability to control it, as shown by our experience with the Rocky Flats plutonium weapons facility." (Tr. 3109, Cobb).

iii) The plutonium burden in humans near Rocky Flats, a plutonium facility (Tr. 2884-2885, Cobb) and the cancer incidence in that area (Tr. 2898, Cobb), suggest that the quality factor for plutonium alpha radiation may have to be as high as 1000, if, indeed, the cancers which have been observed in the area near Rocky Flats are caused by the plutonium which is found in humans in that area. (Tr.

2888, 2919, Cobb).

d) A fourth example of possible nonconservatism, although one not generally accepted, is the "hot particle hypothesis," proposed by Arthur R. Tamplin and Thomas B. Cochran in a series of NRDC reports. (Tr. 3083-3084, Cochran).

84. There remain substantial uncertainties concerning the dose and health effects associated with alpha radiation.

a) While none of the hypotheses cited in Finding 83 are proof that the risks of alpha-emitters are as high as the respective hypotheses suggest, they demonstrate that there is a wide range of interpretation of the data and that different experts have widely divergent views regarding the calculated dose and health effects associated with alpha radiation. (Tr. 3084, Cochran).

b) The authors of the BEIR-III Report concluded, with regard to the possible influence of "hot spots" of insoluble radioactive particles deposited in pulmonary tissues on cancer risk, that:

> The evidence is still insufficient to determine whether aggregates of radioactivity that remain localized in specific regions of the lungs give a greater or smaller risk of lung cancer per average lung dose than uniformly deposited radiation. Preliminary experimental data indicate that a small fraction of inhaled insoluble particles

may remain in the bronchial epithelial layer for long periods, but the significance of this local exposure on lung-cancer risk is still uncertain. (Tr. 3084-85, Cochran).

85. Based on the foregoing facts (Findings 82-83):

a) There is an uncertainty on the order of a factor of 80 in estimates of the bone surface dose due to plutonium (see Finding 83(a)(vi)) and

b) There is an uncertainty on the order of a factor of 50 in estimates of the lung dose due to plutonium which represents the difference between an assumed quality factor of 1000 (Finding 82(b)(i) and 83(c)(iii)) and a quality factor of 20 assumed for alpha radiation in ICRP-30. (Tr. 3163, Morgan).

U. The Proposed Dose Guideline Values for Lung and Bone Are Too High

86. There are now no dose guideline values for bone and lung in 10 CFR Part 100. (Tr. 3013, Cochran).

87. There are alternative ways to select guideline values for bone and lung. (Tr. 3013, Cochran; Tr. 2511 at A53, Branagan).

88. The Staff's proposed dose guideline values (OL stage) of 75 rem to the lung and 300 rem to the bone surface were derived

from the 300 rem thyroid dose guideline value (10 CFR §100.11) and the stochastic weighting factors in ICRP-26. (Tr. 3078, Cochran; Tr. 2511 at A53, Branagan).

89. The Staff ignored the additional limits on organ doses of 50 rem/year to the lung and bone surface recommended by ICRP-26 in order to prevent non-stochastic effects. (Tr. 3078, Cochran).

90. The EPA proposed an even lower dose commitment limit of 30 rem/year to these same organs to prevent non-stochastic effects. (Tr. 3078-79, Cochran).

91. The original intent behind the 10 CFR Part 100 dose guideline values was to ensure that siting of a plant would not result in "serious injury to individuals offsite if the unlikely, but still credible accident should occur." (26 Fed. Reg. 1224 (Feb. 11, 1961)). (Tr. 3079, Cochran).

92. There is strong evidence that the dose guideline levels proposed by the Staff for limiting exposure to plutonium are nonconservative. (Tr. 3139, Morgan).

a) The Staff's proposed dose guideline values (at the CP or LWA stage) of 150 rem to the bone surface and 35 rem to the lung would result in serious consequences and are far beyond acceptable levels. (Tr. 3142, Morgan).

b) The ACRS first suggested as criteria which should be useful in the selection of sites for nuclear reactors, values of 25 rem to the whole body, 300 rem to the thyroid, and 25 rem to the bones and lung. (Tr. 3079, 2987-2988, Cochran).

c) Annual dose-equivalent limits can be used to give some indication of where one should properly establish dose guideline values for lung and bone (or bone surface) dose from plutonium exposure to protect the public health under 10 CFR Part 100. (Tr. 3004, Cochran).

d) Using the annual dose equivalent limits set forth in EPA's uranium fuel cycle regulations (40 CFR §190.10(a)) rather than ICRP-26 non-stochastic risk factors, the lung and bone surface doses equivalent to 25 rem to the whole body would be 25 rem to the lung and bone surface. (Tr. 3080, Cochran).

e) These annual dose limits (Finding 92(d)) are based in part on consideration of the "as low as reasonably achievable" (ALARA) principle. (Tr. 2991-92, Cochran).

f) The Environmental Protection Agency's "Proposed Guidance on Dose Limits for Persons Exposed to Transuranium Elements in the General Environment", EPA 520/4-77-106, Sept. 1977 (Tr. 3139, Morgan; Tr. 2884, 2890-2893, Cobb),which is based "on possible remedial actions for the protection of public health in instances of presently existing contamination of possible future unplanned release of transuranic elements" states that the alpha dose to the critical segment of the exposed population as a result of exposure to transuranic elements should not exceed either one millirad per year to the pulmonary lung or three millirad per year to the bone. (Tr. 2913, Cobb; 3139, Morgan).

g) While there is no proof that EPA's proposed dose limit guidelines are inadequate, there are indications that they may be seriously inadequate to protect the public health. (Tr. 3101, 2907, Cobb).

h) The Colorado State guidelines for permissible levels of plutonium in the environment (2 disintegrations per minute per gram of soil) are even stricter than the EPA guidelines, by a factor of about 25. (Tr. 2098, 3103, Cobb).

V. <u>Neither Applicants nor Staff Have Established That the Models, Computer Codes, Input Data and Assumptions Used to Analyze CDAs and Their Consequences Are Valid</u>

93. The Applicants used the SAS3D, PLUTO, VENUS, REXCO-HEP, COMRADEX III, CACECO, and HAA-3B computer codes in their analyses of CDA energetics, CDA consequences and site suitability. (Tr. 3088, Cochran).

a) The Applicants used SAS3D, PLUTO, VENUS and REXCO-HEP computer codes to analyze CDAs and their consequences within the reactor vessel. (Tr. 3088, Cochran).

b) The assumptions and results of CDA analyses using these computer codes are often design specific. (Tr. 3090, Cochran).

c) Many parameters in the codes are left to the user to determine, which actually regulates the sequence, timing and ultimate energy release of the accident. (Tr. 3089, Cochran).

d) The predicted CDA energy release (and therefore the source term) associated with a CDA is regulated by the users' (Applicants') input assumptions. (Tr. 3090, Cochran).

e) The parameters used in the computer codes by Applicants to analyze CDAs and CDA consequences have not been reviewed by the Staff. (Tr. 3090, Cochran).

f) An internal ORNL memorandum suggests that Applicants do not have the capability, using the computer codes, to predict the mechanistic disassembly of a reactor during an accident to the accuracy required if such an accident is declared a design basis accident (DBA). (Tr. 3090, Cochran).

94. The SAS3D computer code, which is used to calculate the occurrence potential, accident progressions and nuclear explosive potential of the CRBR core (Tr. 3093, Cochran), has numerous problems which seriously impair confidence in its results:

a) Several errors and seeming inconsistencies have been detected in the SAS3D input manual and computer code. (Tr. 3092, Cochran).

b) The SAS3D code lacks complete documentation and has not been adequately checked out. (Tr. 3092, Cochran).

c) A 1977 memorandum from the chief engineering officer of the CRBR project to the Chief of the division responsible for planning, development, coordinating and executing policies and plans in the areas of public safety, environmental affairs and nuclear licensing called unambiguously for the systematic deletion from an Argonne National Laboratory report that represents the principal technical documentation for the validity of the SAS3D code, of "negative" information that would interfere with the licensing of the CRBR. (Tr. 3093, Cochran).

95. The Staff used three computer codes to calculate the site suitability analysis, HAA-3, PAVAN, and TACT 5. (Tr. 3086, Cochran; Tr. 2518, Bell).

a) While the Staff claims that TACT 5 has been validated/verified by hand calculation, benchmark data execution and comparison against Applicants' data (Tr. 2520, Bell), there is no evidence that a formal code review process has been conducted by the Staff. (Tr. 3087, Cochran).

b) The lack of a formal code review is evidenced by the fact that:

i) unspecified modifications were made to the TACT code subsequent to the time the programmers manual was written (Tr. 3088, Cochran); and

ii) the TACT 5 computer code is not documented (Tr. 2519, Bell).

c) Given the inadequacies in the documentation of the TACT 5 code (Findings 95(a)-(b)), the Staff's calculations cannot be accepted as reliable. (Tr. 3088, Cochran).

96. Based upon Findings 93-94, the Applicants' codes should not not be relied upon at the LWA-1 stage. (Tr. 3095, Cochran).

Respectfully submitted,

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Dated: October 4, 1982

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

Marshall E. Miller, Chairman
Gustave A. Linenberger, Jr.
Cadet H. Hand, Jr.

In the Matter of:	
UNITED STATES DEPARTMENT OF ENERGY PROJECT MANAGEMENT CORPORATION TENNESSEE VALLEY AUTHORITY	Docket No. 50-537
(Clinch River Breeder Reactor Plant)	October 1, 1982

AFFIDAVIT OF KARL Z. MORGAN

I, Karl Z. Morgan, being duly sworn, do hereby affirm and say:

1. My name is Karl Z. Morgan. I reside at 1984 Castleway Drive, Atlanta, Georgia 30345.

2. I testified as an expert witness on behalf of Intervenors Natural Resources Defense Council, Inc. and the Sierra Club at the Clinch River Breeder Reactor site suitability hearing before the Atomic Safety and Licensing Board on August 27, 1982 in Oak Ridge, Tennessee.

3. This affidavit is prepared for use in the above-captioned proceeding.

4. At the hearing on August 27, 1982, I testified as follows:

> That is, if an accident in the future releases breeder fuel, the cancer risk from Plutonium is 55 times greater from Plutonium-238, plus Plutonium-241, than from Plutonium-239, and 50 times greater than the NRC staff assumed.

Hearing Transcript at 3130. (This statement was added to my prefiled testimony as an amendment at the beginning of my oral testimony. Hearing Transcript at 2880.)

5. The number "50" in the last clause of the above-quoted passage of my testimony is in error. My calculation had yielded the result "56", which I conservatively rounded off to "50". Due to a misplaced decimal point, the correct number should have been "5.6".

6. The correct statement is that the cancer risk from Plutonium is 5.6 times greater than the NRC Staff assumed. Accordingly, my testimony at Hearing Transcript pages 2880, 3130, and 3172 should be amended to so read.

Executed on October ____, 1982 in Atlanta, Georgia.

Karl Z. Morgan

Date: Karl Z. Morgan

Sworn and subscribed before me this 1st day of October, 1982.

Julie K Parker

Notary Public

My commission expires ________.

Notary Public, Georgia, State At Large
Commission Expires Nov. 29, 1983