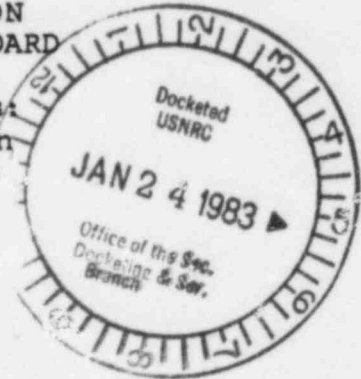


January 24, 1983

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
ATOMIC SAFETY AND LICENSING BOARD

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



In the Matter of)
)
)

UNITED STATES DEPARTMENT OF ENERGY)
PROJECT MANAGEMENT CORPORATION)
TENNESSEE VALLEY AUTHORITY)

(Clinch River Breeder Reactor Plant))
)
)

Docket No. 50-537

INTERVENORS, NATURAL RESOURCES DEFENSE
COUNCIL, INC. AND THE SIERRA CLUB,
PROPOSED FINDINGS OF FACT FOR THE
LIMITED WORK AUTHORIZATION (LWA-1)
PROCEEDING

Pursuant to 10 CFR §2.754(a), and in accordance with the Board's rulings of December 17, 1982 and January 4-5, 1983, Intervenor, Natural Resources Defense Council, Inc. and the Sierra Club, hereby submit their proposed findings of fact for the limited work authorization (LWA-1) proceeding in the above-captioned case.

DS03

Contentions 1(a), 3(b) and 3(d)

1. The envelope of DBAs should include the CDA.
 - a) Neither Applicants nor Staff have demonstrated through reliable data that the probability of anticipated transients without scram or other CDA initiators is sufficiently low to enable CDAs to be excluded from the envelope of DBAs.
3. Neither Applicants nor Staff have given sufficient attention to CRBR accidents other than the DBAs for the following reasons:
 - b) Neither Applicants' nor Staff's analyses of potential accident initiators, sequences, and events are sufficiently comprehensive to assure that analysis of the DBAs will envelop the entire spectrum of credible accident initiators, sequences, and events.
 - d) Neither Applicants nor Staff have adequately identified and analyzed the ways in which human error can initiate, exacerbate, or interfere with the mitigation of CRBR accidents.

I. AN LMFBR REQUIRES A HIGHER STANDARD OF PROTECTION AGAINST CDAs THAN AN LWR, AND SHOULD INCLUDE CDAs WITHIN THE DESIGN BASIS

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 an energetic core disruptive accident (CDA) which can be described as a low-order nuclear

explosion. (Cochran, Int. Exh. 3, p. 9, Tr. 2818; 2776-81; Cochran, Int. Exh. 22, pp. 37-42, 6231-36; 6154-58, 6181-83). A CDA, or 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. (Cochran, Int. Exh. 3, p. 10, Tr. 2819; Staff Exh. 8, FSFES, App. J, p. J-8).

b. LMFBRs generally contain several times the core inventory of the highly toxic isotopes of plutonium than do LWRs. Release of plutonium into the environment following CDAs in LMFBRs potentially represents a far more serious ground 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. (Cochran, Int. Exh. 3, p. 10, Tr. 2819).

c. 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. (Cochran, Int. Exh. 3, p. 10, Tr. 2819).

d. It is not possible to model accurately the behavior of the CRBR core once cladding melting begins. (Cochran, Int. Exh. 3, pp. 10-11, Tr. 2819-20).

e. For those and other reasons, some experts in the technical community believe that LMFBRs require a higher standard for protection against CDAs compared to LWRs. (Cochran, Int. Exh. 3, p. 11, Tr. 2820). In fact, CDAs have occurred or were considered DBA's in other U.S. fast reactors. (Findings 55 to 60).

II. STAFF AND APPLICANTS' OWN ANALYSIS DEMONSTRATES THAT A CORE DISRUPTIVE ACCIDENT SHOULD BE INCLUDED WITHIN THE DESIGN BASIS OF THE CRBR.

A. Staff's Safety Objectives

2. In determining whether CDAs should be included within the design basis for the Clinch River Breeder Reactor (CRBR) 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. (Staff Exh. 7, p. 7-2; Staff Exh. 8, p. 7-1; Morris, Tr. 2277-79; Staff Exh. 5, p. 2; Int. Exh. 1, pp. 7-8). Applicants also set this criterion early in the project, though they no longer believe such a criterion is "necessary." (Clare, Tr. 1483).

3. Staff has taken the position that ... numerical evaluations of system reliability and accident risks ... are of significant value in indicating whether the safety objective "aiming point" is being adequately approached." (Staff Exh. 5, p.2).

4. The Commissions' Standard Review Plan for light water reactors (Staff Exh. 6, p. 2.2.3-2) also states:

[T]he identification of design basis events resulting from the presence of hazardous materials or activities in the vicinity of the plant is acceptable if the design basis events include each postulated type of accident for which the expected rate of occurrence of potential exposures in excess of the 10 CFR Part 100 guidelines is estimated to exceed the NRC staff objective of approximately 10^{-7} per year....[T]he expected rate of occurrence of potential exposures in excess of the 10 CFR Part 100 guidelines of approximately 10^{-6} per year is acceptable if, when combined with reasonable qualitative argument, the realistic probability can be shown to be lower. (emphasis added)

B. Staffs' Estimated Probability for CDA Initiation

5. In Appendix J of the Supplement to Final Environmental Statement related to construction and operation of Clinch River Breeder Reactor Plant (Staff Exh. 8, FSFES) Staff calculated the probability of four classes of CDA accident sequences, Classes 1 through 4. (Staff Exh. 8, FSFES, App. J, p. J-11). Staff admitted that the doses associated with CDA Classes 2 through 4 might exceed 10 CFR Part 100 guidelines, but claimed that the frequency of occurrence of those accidents was less than 10^{-6} per reactor year. (Staff Exh. 8, FSFES, App. J, p. J-11).

6. Staff's CDA Class 1 postulates CDA accident sequences of varying severity, but assumes that the containment system

functions as designed. (Staff Exh. 8, FSFES, App. J, p. J-11). Staff estimates the probability of this accident class at "less than 10^{-4} per reactor year," -- a "bounding estimate" using "realistic calculations for CDAs." (Id. at J-8, footnote p. J-11).

7. Staff is unable, based upon the level of analysis it performed in FSFES Appendix J, to distinguish between a probability of CDA initiation of 10^{-4} and that of 2×10^{-4} . (Rumble, Tr. 5614).

8. According to Staff, the probability that the wind will blow the CDA releases in any particular sector (i.e., any particular direction) is a factor of 10^{-1} . (Staff Exh. 18, p. 9, Tr. 5691). Staff's and Applicants' meteorological calculations are based on the use of sixteen 22.5 degree wind direction sectors. (See App. Exh. 2, PSAR, p. 2.3-10). Therefore, the probability of a CDA Class 1 accident sequence in which the releases are blown towards the worst-case direction is approximately 10^{-5} per reactor year.

C. The Doses Associated With Staff CDA Class 1 Accident

9. Staff claims that "the doses associated with Staff CDA Class 1 are not expected to exceed 10 CFR 100 guidelines." (Staff Exh. 8, FSFES, App. J, p. J-11) (emphasis added). Yet Staff has included no calculation in the record of the CDA Class 1 doses at the exclusion area or low population zone (LPZ)

boundaries which would support this conclusion, or justify the exclusion of this CDA accident sequence from the design basis.

10. To the contrary, the dose at the LPZ boundary to the maximally exposed individual resulting from a CDA Class 1 accident sequence would in fact greatly exceed the 10 CFR Part 100 thyroid dose guidelines using Staff's, as opposed to Applicants', modeling assumptions.

11. Although Staff has not calculated doses to the maximally exposed individual at the LPZ boundary for a Class 1 CDA, it has calculated the doses to an individual located at the Oak Ridge Gaseous Diffusion Plant (ORGDP) following a Class 1 CDA (Thadani, Tr. 5664). These doses include a dose of 100 rem to the thyroid (Staff Exh. 18, p. 7, Tr. 5689), a value close to the 10 CFR Part 100 guideline value of 150 rem to the thyroid at the CP (and LWA-1) stage. (Staff Exh. 1, 1982 SSR, p. III-9).

12. ORGDP is located approximately 3 miles north-northwest of the CRBR site. (Staff Exh. 1, 1982 SSR, p. III-6). The ORGDP is therefore just outside the outer boundary of the LPZ, which is 2.5 miles from the CRBR site. (Staff Exh. 1, 1982 SSR, p. III-3). The ORGDP, however, is not located in the wind direction sector with the worst-case meteorological conditions, that is, the wind direction sector with the highest X/Q values. (App. Exh. 34, ER pp. 2.6-10, -52; App. Exh. 2, PSAR, pp. 2.3-11, -12). This can be seen also by comparing Staff's Site Suitability Source Term (SSST) thyroid and whole body doses at

the LPZ boundary in the worst-sector direction (Staff Exh. 1, 1982 SSR, p. III-11) and the smaller SSST thyroid and whole body doses at ORGDP. (Staff Exh. 18, p. 6, Tr. 5688)

13. In order to calculate a first-order approximation of the CDA Class 1 dose to the maximally exposed individual at the LPZ boundary, one need only adjust the ORGDP dose to account for the worst-case LPZ meteorological conditions. In essence, one need only multiply the ORGDP thyroid dose by the factor representing the ratio of the thyroid doses at the two sites (at the LPZ boundary in the worst case wind direction relative to the ORGDP site). The ratio would result from differences in the X/Q values at the two sites, since the other parameters in the dose calculation (e.g., radioactive release rates, time of release, dose conversion factors, breathing rate) would be the same (see generally, Regulatory Guide 1.109).

Neither Staff nor Applicants reported the 50% X/Q values for their CDA dose calculations at ORGDP. Applicants could not recall the values, or whether the values were ever documented. (Hibbits, Clare and Strawbridge, Tr. 5197-98).

14. The difference in 50% X/Q values for the ORGDP site and the 50% X/Q values for the LPZ boundary in the worst-case sector results in a factor of 12 to 14 difference in thyroid dose. This can be seen by comparing Applicants' calculated thyroid doses at the LPZ boundary and ORGDP for Applicants' Case 2 HCDA. (App. Exh. 47, p. 5, Tr. 5425). This accident is comparable to Staff's

CDA Class 1 in that it involves a CDA release with no failure of CDA mitigating systems, e.g., the containment/confinement systems. In other words, Staff's CDA Class 1 and Applicants' HCDA Case 2 both assume the containment/confinement systems operate as designed. (App. Exh. 1, p. 69, Tr. 2053; Strawbridge, Tr. 5072-73). Applicants calculated a Case 2 thyroid dose of 85.4 rem at the LPZ in the worst-sector direction (App. Exh. 46, p. 34, Tr. 5410) versus 7.1 rem at the ORGDP (App. Exh. 47, pp. 13-14, Tr. 5433-34; Staff Exh. 18, p. 7, Tr. 5689), a factor of 12 difference. (The difference would be a factor of 14 if Applicants' previously calculated LPZ thyroid dose of 99.2 rem were used (App. Exh. 1, p. 71, Tr. 2060; see also Strawbridge, Tr. 5156, 5158, 5159-60, 5163-64)).

15. Similarly, there appears to be more than an order of magnitude difference in thyroid doses due to differences in the 95% X/Q values between the ORGDP and the LPZ boundary in the worst-sector direction. Staff calculated the SSST thyroid dose as 0.32 rem at the ORGDP (Staff Exh. 18, p. 6, Tr. 5688) and 7 rem at the LPZ boundary (Staff Exh. 1, p. III-11), a factor of 22 difference.

16. Applying the dose factor ratio of 12, due to differences in Applicants' 50% X/Q values, to Staff's CDA Class 1 thyroid dose of 100 rem at the ORGDP, the corresponding thyroid dose to the maximally exposed individual at the LPZ boundary is approximately 1200 rem ($100 \text{ rem} \times 12 = 1200 \text{ rem}$). This dose greatly exceeds

the 150 rem thyroid dose guideline value used by Staff in its CP review. (Staff Exh. 1, 1982 SSR, p. III-9).

17. In conclusion, according to Staff's own data, therefore, a CRBR core disruptive accident with an upper bound probability of approximately 10^{-5} per reactor year would most likely result in thyroid doses far exceeding the 10 CFR Part 100 dose guideline values. According to Staff's safety objective and the LWR Standard Review Plan, this CDA Class 1 accident sequence must be included within the design basis envelope for the CRBR.

D. Applicants' Estimates of CDA Probabilities and Consequences

18. Unlike Staff, Applicants have introduced into the record no independent estimates of the probabilities of various CDA accident sequences. Nor are Applicants relying on any of their own probability analyses for their conclusions regarding Staff's CDA probability analyses. (Strawbridge, Tr. 4997). Applicants' claim that Staff's FSFES, Appendix J probability estimates are conservative (App. Exh. 46, p. 21, Tr. 537) is based on observations of alleged conservatisms in Staff's analysis (App. Exh. 46, pp. 13-21, Tr. 5389-97) but omits substantial offsetting nonconservatisms which also exist. (Finding 123).

19. Applicants claim that Staff's CDA probability estimates are conservative because each of Staff's four CDA classes (Staff Exh. 8, FSFES, App. J, Table J-2) assumes a head release, which implies that all HCDAs are energetic. Applicants admitted,

however, that this criticism has little if any applicability to Staff's CDA Class 1 (Strawbridge, Tr. 5075-76). CDA Class 1 assumes that the containment functions as designed regardless of the severity of the primary system failure, and thus it is relatively insensitive to the magnitude of the head release. (Strawbridge, Tr. 5073-75).

20. Applicants have calculated directly the doses at the LPZ boundary in the worst-case wind direction sector for four HCDA accident sequences. (App. Exh. 1, Table 5-1, Tr. 2060; App. Exh. 46, p. 34, Tr. 5410). Applicants' estimated LPZ thyroid dose for HCDA Case 2 (corresponding to Staff's CDA Class 1), based on so-called 'realistic assumptions,' is about 85 rems. (App. Exh. 46, p. 34, Tr. 5410).

21. This 85 rem thyroid dose is approximately a factor of 14 below the 1200 rem thyroid dose estimated above based on Staff's data. Similarly, Applicants' and Staff's estimated thyroid dose at the ORGDP (7 rems and 100 rems respectively) also differ by approximately a factor of 14. (Staff Exh. 18, p. 7, Tr. 5689).

22. This difference in ORGDP thyroid doses is due primarily to the fact that Staff assumed more conservative filter efficiencies (99% efficiency for particulates and 95% for the iodines) than Applicants. (Thadani, Tr. 5665-66). The difference in the thyroid dose due to different assumed filter efficiencies is actually greater than a factor of 14, since it is offset in part by Staff's use of less conservative assumptions regarding

meteorology, principally the height of the radioactive release. (Staff Exh. 18, pp. 6-8, Tr. 5688-90; Thadani, Tr. 5665-67).

23. The difference between the two LPZ boundary thyroid dose estimates based on Staff's and Applicants' data, 1200 rem and 85 rem respectively, reflects differences in modeling assumptions based on independent calculations. (Staff Exh. 18, p. 8, Tr. 5690). Even if the mean value of 640 rem is taken as the "best estimate," this value is still greater than the 150 rem dose guideline value for thyroid exposure.

III. STAFF HAS PROVIDED NO ADEQUATE JUSTIFICATION
FOR ITS PLACEMENT OF CDAs OUTSIDE THE CRBR DESIGN BASIS

A. Staff's Reliance On Engineering Judgment To Exclude CDAs From
the CRBR Design Basis Is Misplaced.

24. Staff claims to have used no quantitative probability estimates of CDA initiation or CDA dose consequences for their conclusion that CDAs should not be within the CRBR design basis. (Morris, Tr. 2191-92, 2280-81; Rumble, Tr. 2173; Cochran, Int. Exh. 3, pp. 48-49, Tr. 2857-58). Staff claims that "engineering judgement" alone is sufficient in order to determine whether CDAs should be within the CRBR design basis. (Morris, Tr. 2281-82). This claim is not supported by the record.

a. The May 6, 1976 letter from Staff to Applicants, (Staff Exh. 5), indicates that

[The] numerical evaluations of
system reliability and accident
risks undertaken by the CRBR Project

and the ERDA LMFBR Development Program, as well as the systematic and disciplined evaluations of the plant design to identify potential causes and pathways for serious accidents so that any required design accommodation can be effectively implemented, are of significant value in indicating whether the safety objective "aiming point" is being adequately approached....

(Staff Exh. 5, p. 2).

b. 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. (Cochran, Int. Exh. 3, p. 57, Tr. 2866). This determination requires a demonstration of the reliability of "state of the art" prevention systems both individually and in combination, taking systems interaction into account. (Cochran, Int. Exh. 3, p. 43, Tr. 2852; Morris, 2280). Staff has not quantitatively analyzed the individual or combined reliabilities of the CRBR Safety Systems. (Morris, Tr. 2280). Staff admitted that in order to make a prudent engineering judgment, one should consider all relevant factors (Morris, Tr. 2176), and that there are cases in which techniques other than engineering judgment have been used to supplement engineering judgment. (Morris, Tr. 2175). Furthermore, Staff 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. (Rumble, Tr. 2185-86; see also Finding 120). The results of specific failure modes/effects analyses are available to Staff (Block, Tr. 1647-48; Clare, 1657, 1680, 1686), but Staff has not considered those results in its engineering judgment regarding the probability of CDA initiation. (Morris, Tr. 2178).

c. Staff's qualitative "engineering judgment" is contradicted by its own quantitative estimates of CDA probabilities and consequences presented in Staff Exhibits 8, 17, and 18. (Findings 2 to 17).

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

25. Staff has failed to establish and justify any general 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. (Cochran, Int. Exh. 3, pp. 44-48, Tr. 2853-57).

a. There are no general design criteria established for fast reactors. (Cochran, Int. Exh. 3, pp. 44-45, Tr. 2853-54).

b. Staff's review of Applicants' proposed general design criteria for the CRBR (Staff Exh. 1, 1982 SSR, Appendix A) is

not complete and will not be set out until the CRBR Safety Evaluation Report (SER) is published. (Cochran, Int. Exh. 3, p. 45, Tr. 2854; Morris, 2148-49, 2408).

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 Appendix A) against the proposed CRBR criteria. (Cochran, Int. Exh. 3, p. 45, Tr. 2854).

d. Applicants' proposed general design criteria for the CRBR (Staff Exh. 1, 1982 SSR, App. A) do not illustrate the feasibility of developing criteria suitable for a plant of this general size and type, since no demonstration has yet been made that the criteria are in fact suitable.

26. Staff's so-called "specific criteria" for the CRBR (Staff Exh. 2, pp. 13-23, Tr. 2458-68; Morris, Tr. 2406-10) are insufficient to render CDAs so improbable that they need not be considered design basis accidents. These criteria do not have specific detail (Morris, Tr. 2206-07), do not indicate what degree of conservatism is appropriate or sufficient, and do not demonstrably ensure that, if complied with, they will enable the CRBR to meet or even approach its probability design objective. (Findings 2 to 23).

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

27. Staff's assertion that the "safety functions which must be achieved for an LMFBR are not fundamentally different from the safety functions successfully implemented for LWRs" (Morris, Tr. 2458, 2205) is not a sufficient basis for excluding CBAs from the design basis. Implementation of a particular safety function could be very different from LMFBRs and for light water reactors. (Morris, Tr. 2206). 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. (Cochran, Int. Exh. 3, p. 37, Tr. 2846; Cochran, Tr. 2662; Morris, 2232-33). Yet Staff has not considered the results of specific failure modes/effects analysis in its engineering judgment regarding the probability of CDA initiation (Morris, Tr. 2178), even though such analyses are available (O'Block, Tr. 1647-48; Clare, 1657, 1680, 1686) and though it would be prudent to consider them. (Rumble, Tr. 2185-86; see also Finding 120). 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. (Morris, Tr. 2256-57). 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. (Cochran, Int. Exh. 3, p. 38, Tr. 2847; Strawbridge, Tr. 1705-07.) Yet 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. Id.

D. Staff Has Failed Adequately To Consider the Potential For CDA Initiation Resulting From Human Error at the CRBR

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

a. Human error could cause an undetected interdependence between various elements of the reactor, such as the two shutdown systems (Morris, Tr. 2255), and human error could be responsible for CDA initiation conditions in both LMFBRs and LWRs (Morris, Tr. 2263), such as failure to maintain sufficient coolant inventory and failure to respond to the loose parts monitoring system. (Morris, Tr. 2226-27).

b. Systematic fault tree/event tree analyses would be helpful in determining the effects of human error in a generic fashion (Rumble, Tr. 2420), but Staff has not performed any such systematic analyses at the CRBR. (Morris, Tr. 2243). Staff has not analyzed, and does not intend to analyze, the extent to which system interdependencies have

been discovered in LWRs for its conclusion that they are highly or very improbable for the CRBR. (Morris, Tr. 2256-57).

c. Staff claims that the potential for human error at the CRBR would not differ significantly from the potential for human error at an LWR (Morris, Tr. 2445), yet Staff has not used the estimates of the high contribution of human error to LWR Licensee Event Reports (LERs) in any way for its conclusion that accidents caused by human error would be very improbable at the CRBR. (Morris, Tr. 2246).

d. One basis for Staff's conclusion that CRBR accidents resulting from human error will be very improbable is the fact that after the TMI-2 accident, the Commission 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. (Morris, Tr. 2443; Staff Exh. 2, p. 23, Tr. 2468). Yet 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. (Morris, Tr. 2260-61).

E. Staff Has Failed to Demonstrate that CDAs are "Incredible."

29. Staff states that it determined which accidents to include within the CRBR design basis envelope by examining a range of

accidents to determine which are "credible." (Staff Exh. 2, p. 5, Tr. 2450). Yet Staff has failed to demonstrate that CDAs are in fact "incredible." Staff attaches no quantitative or qualitative probability to the word "credible" (Rumble, Tr. 2173; Morris, 2191-92), and states that its only definition of "credible" is one synonymous with "accidents within the design basis envelope." (Hulman, Tr. 2172; Staff Exh. 2, p. 8, Tr. 2453). Consequently, Staff attaches no quantitative significance to the term "design basis envelope," in contradiction to its own CRBR safety objective. (Findings 2-4.)

30. Staff has failed to demonstrate that it is feasible to design the CRBR so that CDAs are "incredible." This requires a demonstration of the reliability of "state of the art" prevention systems both individually and in combination, taking system interaction into account. (Cochran, Int. Exh. 3, p. 43, Tr. 2852; Morris, 2280). Staff has not quantitatively analyzed the individual or combined reliabilities of the CRBR safety systems. (Morris, Tr. 2280).

31. Staff cannot logically reach a final determination as to whether CDAs or other accidents should be within the design basis envelope until it has completed a detailed CRBR safety review. Staff has not yet determined whether Applicants' list of proposed design basis accidents is sufficient (Tr. 2192-93, Morris), but might add to the list of design basis accidents after a detailed safety review. (Tr. 2193, Morris). Staff claims 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). There is no evidence in the record, however, to distinguish "changing the design" from "including the CDA within the design basis." According to Staff, furthermore, finding CDAs to be credible would automatically place them within the design basis envelope, since "credible" and "design basis" are considered synonymous. (Hulman, Tr. 2172).

F. Staff Originally Considered CDAs as DBAs for the CRBR and Has Demonstrated No Rational Basis For Its Change in Position.

32. In 1975, after the CRBR Preliminary Safety Analysis Report (PSAR) was submitted to Staff (Clare, Tr. 1837), Staff took the position that CDAs should be within the design basis for the CRBR. (Cochran, Tr. 2620-22, 2650-53; Int. Exh. 3, p. 25, Tr. 2834; Staff Exh. 5, p. 5). 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. (Strawbridge, Tr. 1503; Cochran, Int. Exh. 3, p. 22, Tr. 2831). On May 6, 1976, however, Staff changed its position, stating that "the probability of core melt and disruptive accidents can and must be reduced to a sufficiently low level to justify their exclusion from the design basis accident spectrum." (Staff Exh. 5, p. 5). Staff could offer nothing more than speculation regarding the change in Staff's position in

1976. (Morris, Tr. 2270). Applicants note that the change in Staff's position occurred soon after Applicants changed the proposed CRBR containment/confinement design. (Clare, Tr. 1837). This change, however, would not affect the probability of CDA initiation, nor does the present containment/confinement system ensure that the consequences of a CDA are below the 10 CFR Part 100 dose guidelines. (Findings 2 to 23).

IV. APPLICANTS HAVE NOT DEMONSTRATED
THAT THE LIKELIHOOD OF A CDA IS SO LOW
THAT IT CAN BE EXCLUDED FROM THE DESIGN BASIS

A. Applicants' Bases for Excluding CDAs from the CRBR Design Basis Envelope are Insufficient.

33. 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. (Cochran, Int. Exh. 3, p. 48, Tr. 2857).

34. Applicants' general design approach, characterized as "defense in depth" (Clare, Tr. 1501; Cochran, Int. Exh. 3, pp.

49-50, Tr. 2858-59), does not provide a basis for excluding the CDA from the DBA envelope. The three-level design safety philosophy (App. Exh. 8, Chapter 15.1.1) by itself does not dictate which accidents are within the design basis, since it was applied to the FFTF, SEFOR, and the CRBR Parallel Design, in which CDAs were treated in essence as design basis events, (Cochran, Int. Exh. 3, pp. 50-53, Tr. 2859-62; Brown, Tr. 1501-02; Strawbridge, Tr. 1503), and since the same safety philosophy would apply whether the CDA is deemed within or outside the design basis. (Cochran, Int. Exh. 3, p. 53, Tr. 2862; Strawbridge, Tr. 1509-10).

35. Applicants have not demonstrated that they have identified and considered all important classes of CDA initiators (O'Block, Tr. 1651; Clare, Brown, Strawbridge, Deitrich, Tr. 1476-78), a necessary condition to reasonably exclude CDAs from the design basis envelope. (Cochran, Int. Exh. 3, p. 53, 55, Tr. 2862, 2864). It is impossible to confidently list all the important initiators before the event tree and fault tree analyses have been performed. (Cochran, Int. Exh. 3, pp. 53-54, Tr. 2862-63). Staff has no basis for judging the completeness of Applicants' list of CDA initiators (Cochran, Int. Exh. 3, p. 54, Tr. 2863) and has not finalized its position regarding some of the potential CDA initiators identified by Applicants, e.g., double ended pipe break. (Staff Exh. 1, p. II-9; Cochran, Int. Exh. 3, p. 54, Tr. 2863).

36. Applicants have not demonstrated that the CRBR general design features "preclude" the occurrence of CDAs. All of the major safety features identified by Applicants and intended to prevent CDAs have some failure rate (Clare, Tr. 1382-83, 1387, 1391, 1393; Cochran, Int. Exh. 3, p. 55, Tr. 2864), and determination of these failure rates is crucial to the question of whether a CDA is "credible." (Cochran, Int. Exh. 3, p. 55, Tr. 2864). Applicants have not quantified the probability of failure of the major design features intended to prevent CDAs (Clare, Tr. 1461-62), but state only that the probability of such failure would be "very low." (Clare, Tr. 1462). Applicants' testimony, however, demonstrates that their use of terms such as "low," "very low level," "extremely unlikely," "prevent," and "high likelihood" are not clearly defined. (Clare, Tr. 1385-86, 1495-96, 1616, 1637, 1639).

B. By Failing to Consider Important Factors Applicants Cannot Provide Reasonable Assurance that CDAs are Not Credible Events.

37. In making their judgment that the likelihood of a CDA is so low that it can be excluded from the design basis envelope, Applicants do not rely upon: their Reliability Program (documented in PSAR Appendix C) (Cochran, Int. Exh. 3, p. 48, Tr. 2857); the probability of failure of the reactor shutdown systems or any of the general design features (Cochran, Int. Exh. 3, p. 48, Tr. 2857; Clare, 1461); tests of the reactor shutdown or shutdown heat removal or other CDA prevention systems

(Cochran, Int. Exh. 3, p. 49, Tr. 2858; Clare, 1479); quantified reliability threshold criteria (Cochran, Int. Exh. 3, p. 49, Tr. 2858; Clare, 1480, 1483, 1497); probabilistic risk assessments (Cochran, Int. Exh. 3, p. 49, Tr. 2858; Clare, 1484); analysis or evaluation of designs of plants other than the CRBR (Cochran, Int. Exh. 3, p. 3, Tr. 2858; Brown, 1684, 1727-28; Clare, 1487); sufficiency or completeness of the SSR Appendix A criteria, the Denise-Caffey letter criteria, or any known set of criteria (Cochran, Int. Exh. 3, p. 47-49, Tr. 2856-58; Clare, 1483, 1487-88); analysis of the CDA once initiated, including Section 5 of Applicants' Exhibit 1 (Cochran, Int. Exh. 3, p. 49, Tr. 2858; Clare, 1488-89); 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. (Clare, Tr. 1461-62).

38. Applicants have no analytical test for selection of DBAs and no basis for excluding CDAs from the DBA envelope.

a. Applicants originally selected those limiting design accidents not covered by NRC regulations, Regulatory Guides, or LWR licensing precedent through their Reliability Program. (Clare, Tr. 1475; Cochran, Int. Exh. 3, p. 56, Tr. 2865; Int. Exh. 1 pp. 1, 6-7). The Reliability Program was used to select DBAs as early as 1974, and only minor

adjustments have been made since then. (Clare, Tr. 1475, 1634). The Reliability Program was used to assure and confirm the low probability of specific initiators not covered by precedent or regulation and thereby allow exclusion of these initiators from the design basis. (Int. Exh. 1, p. 7).

b. Presently Applicants contend that they established the CRBR design basis accidents without the use of the Reliability Program or reliance on the adequacy of that program. This is inconsistent with Applicants' earlier assertions. (Finding 2). (Cochran, Int. Exh. 3, p. 56, Tr. 2865; Clare, 1463). But no alternative analytical test of Applicants' hypothesis that the CDA can be excluded from the design basis has been provided. (Cochran, Int. Exh. 3, p. 58, Tr. 2867). And, as Staff consultant at Los Alamos National Laboratory has stated, any cavalier approach justified by the hypothetical (often equated with impossible) status of these CDA accidents can degenerate quickly to judgments (perhaps hunches or guesses) instead of facts or quantified certainties. (Cochran, Int. Exh. 3, pp. 57-58, Tr. 2866-67).

39. 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." (Cochran, Int. Exh. 3, p. 59, Tr. 2868).

40. Applicants' argument that "We will require CDAs to be of low probability, hence they will be of low probability" (Cochran, Int. Exh. 3, p. 59, Tr. 2868) is circular. The Commission "required" the TMI-2 core not to be severely damaged, yet it was severely damaged. (Cochran, Int. Exh. 3, p. 59, Tr. 2868). The Atomic Energy Commission "required" that melting should occur in no more than one subassembly in the Fermi-1 core, yet there was melting in two subassemblies. (Cochran, Int. Exh. 3, p. 59, Tr. 2868).

41. Based upon the above facts, CDAs cannot be considered incredible for the CRBR, or for a reactor of the general size and type as the CRBR. (Cochran, Int. Exh. 3, p. 59, Tr. 2868).

V. THE DOUBLE-ENDED PIPE BREAK COULD CAUSE A CDA
IN THE CRBR, AND THERE IS NO BASIS FOR EXCLUDING IT
FROM THE DBA ENVELOPE

A. Introduction

42. Staff concluded that loss of coolant accidents ("LOCAs") caused by large primary coolant pipe breaks, which could lead to CDAs, should not be considered credible (i.e., design basis) events at CRBRP (Staff Exh. 1, p. II-9), and that the 10^{-4} per reactor year frequency assumed for loss of heat sink (LOHS) events adequately bounds the LOCA contributions to core disruption frequency. (Staff Exh. 8, FSFES, App. J, p. J-4). Staff cites the physical properties of the sodium coolant, implementation of an inspection program and a leak detection

system, and installation of guard vessels around the primary coolant as the bases for its conclusion that CRBR LOCAs are incredible. (Staff Exh. 8, FSFES, App. J, p. J-4.) This conclusion is unsupported by the record. A double-ended pipe break LOCA is considered a design basis accident for light water reactors (Strawbridge, Tr. 1509), and Staff has provided insufficient justification for departing from this approach for the CRBR.

B. The Harris Analysis

43. According to Applicants' consultant Harris (also a consultant to Staff), the CRBR pipe rupture frequency is comparable to that of a PWR within the uncertainty values demonstrated by his sensitivity analysis. (Cochran, Int. Exh. 22, pp. 19-22, Tr. 6213-16; Attachment 3, Tr. 6271-72). Harris' analysis demonstrates that within the present state of knowledge, it is not possible to ascertain the controlling parameters that govern the relative CRR/PWR pipe break frequency; that the failure rate of primary piping in CRBR is 0.1 to 1 times the corresponding value for PWRs; and that the frequency is a strong function of the number and characteristics of the pipe welds, which are design dependent. (Int. Exh. 22, pp. 20-22, Tr. 6214-16; Attachment 3, Tr. 6271; Cochran, Tr. 6179-81). Harris' analysis actually indicates that the CRBR pipe break frequency may be as much as 12 times higher than that for a PWR. (Int. Exh. 22, Attachment 3, Tr. 6269).

44. In an earlier interim report by Harris, CRBR pipe rupture probabilities were given as " 10^{-8} /plant-year for the cold leg, and 10^{-7} /plant year for the hog leg (See Ref. 1 [CRBRP-1], page III-116)." (Cochran, Tr. 6132; Tr. 6172-73). These absolute probabilities cannot serve as a basis for excluding the CRBR pipe break from the DBA envelope because (a) if the relative CRBR/PWR pipe break frequency is approximately 1 (Finding 43) it would argue against considering PWR pipe breaks as design basis events contrary to Commission policy, and (b) there is no basis in the record for determining whether Harris believes the absolute probabilities have significance as opposed to the comparative ratio (Cochran, Tr. 6172).

C. Staff's and Applicants' Reasons for Excluding Pipe Breaks from the DBA Envelope are Inadequate.

45. Implementation of a CRBR preservice and inservice inspection program or a comprehensive quality assurance program is an insufficient basis for concluding that CRBR pipe rupture frequency will be less than that for light water reactors. Such programs have been in place for light water reactors for some time (Cochran, Int. Exh. 22, p. 21, Tr. 6215; Clare, Strawbridge, Tr. 1552-54).

46. Implementation of a CRBR leak detection system is an insufficient basis for distinguishing CRBR pipe break frequency from that for LWRs. The leak detection system is designed to alert an operator and trigger operator action, and carries with

it the potential for human error. (Clare, Tr. 1547-48). It is thus not a totally passive system. (Strawbridge, Tr. 5051). Applicants do not know the failure rate for the leak detection system for either sodium or steam generator leaks. (Clare, Tr. 5294-95).

47. Use of sodium coolant near atmospheric pressure is not a sufficient basis for concluding that a double-ended pipe break should not be a design basis accident for the CRBR. (Clare, Tr. 1534-35). In some portions of the sodium loops the sodium may be at a pressure of approximately ten atmospheres. (Clare, Tr. 1536).

48. Use of stainless steel piping is not a sufficient basis 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. (Clare, Deitrich, Tr. 1538; Brown, Tr. 1540).

49. Placement of piping in nitrogen-inerted cells with low oxygen content is not a sufficient 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. (Brown, Tr. 1539-40). In fact, repair of the piping would be more difficult in the CRBR than if inerted cells were not used, such as in PWRs. (Strawbridge, Tr. 5016; Clare, Tr. 5010).

50. Use of guard vessels around the primary coolant piping is not a sufficient reason to conclude that a CRBR double-ended pipe break should not be a design basis accident. A pipe break in the primary coolant system could lead to a CDA even though the passive guard vessel system would maintain primary coolant inventory. (Strawbridge, Tr. 5050). Furthermore, the use of guard vessels in a breeder reactor would make it more difficult to visually inspect the coolant piping, relative to light water reactors. (Clare, Tr. 5001).

51. The existence of a material surveillance program is an insufficient 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. (Clare, Brown, Strawbridge, Deitrich, O'Block, Tr. 1540-41).

52. The assertion that CRBR piping would retain its integrity even if one or two snubbers were to fail during plant operational loadings (App. Exh. 1, p. 41, Tr. 2030) is not based on familiarity with requirements for snubbers in light water reactors (Clare, Brown, Strawbridge, Deitrich, O'Block, Tr. 1542-49), and should therefore be given little weight.

53. 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 (Clare, Tr. 1552), there is no evidence that Staff will require

Applicants to meet the more restrictive specifications. (Clare, Tr. 1555).

54. 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 (Clare, Tr. 1567-68), other than the fact that Applicants are unaware of any double-ended pipe ruptures that had occurred in LWRs. (Clare, Tr. 1568).

VI. CDAs OCCURRED OR WERE CONSIDERED AS DBAs
IN OTHER U.S. FAST REACTORS

55. 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, caused in part because automatic safety devices were intentionally disconnected. (Cochran, Int. Exh. 3, pp. 13-14, Tr. 2822-23; Tr. 2628-29). The Atomic Energy Commission had obvious concerns about the potential recurrence of that type of accident in subsequent fast reactors, and the accident was discussed in the context of the safety review of the EBR-II reactor. (Cochran, Tr. 2628).

56. The Enrico Fermi-1 plant was a 200 MWt demonstration LMFBR which was licensed by the Atomic Energy Commission 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." (Cochran, Int. Exh. 3, pp. 14-15, Tr. 2823-24,).

57. 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. (Cochran, Int. Exh. 3, Tr. 2824; 2638-39). The containment "design basis energy release" for SEFOR was 400 MW-sec., far more than the 100 MW-sec. the Atomic Energy Commission staff concluded was the "theoretical upper limit of the energy available as kinetic energy." (Cochran, Int. Exh. 3, p. 16, Tr. 2825). 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. (Cochran, Int. Exh. 3, p. 16 Tr. 2825; 2786-88). Applicants admitted that CDAs were within the equivalent of the third level of design safety (the design basis) for SEFOR. (Brown, Tr. 1502).

58. 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 Atomic Energy Commission staff. (Cochran, Int. Exh. 3, p. 16, Tr. 2825). It can be inferred from the Safety Evaluation Report for the FFTF that CDAs were treated as equivalent to design basis accidents for the plant. (Cochran, Int. Exh. 3, pp. 16-18, Tr. 2825-27, 2639-40, 2643, 2790-91). Applicants testified that a CDA was within the third level of

design safety for the FFTF. (Brown, Tr. 1502). 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. (King, Long, Tr. 2395-96).

59. EBR-II is an experimental 67.5 MWt fast neutron test reactor which was not licensed but did undergo an AEC safety review. (Cochran, Int. Exh. 3, p. 14, Tr. 2823). Its primary containment was designed to contain "without breaching" a "reasonable" upper limit on the explosive energy of about 300 lb. TNT. (Cochran, Tr. 2790; Cochran, Int. Exh. 3, p. 14, Tr. 2823).

60. Staff has made no systematic effort in its LWA-1 review to take into account foreign experience with breeder reactors (Morris, Tr. 2209), and does not have a good understanding of the design basis, design, or implementation of specific features in other domestic or foreign breeder reactors. (Morris, Tr. 2207-08, 2210, 2212-14, 2459; King, Tr. 2215).

Contention 2.

2. The analyses of CDAs and their consequences by Applicants and Staff are inadequate for purposes of licensing the CRBR, performing the NEPA cost/benefit analysis, or demonstrating that the radiological source term for CRBRP would result in potential hazards not exceeded by those from any accident considered credible, as required by 10 CFR §100.1(a), fn. 1.
 - a) The radiological source term analysis used in CRBRP site suitability should be derived through a mechanistic analysis. Neither Applicants nor Staff have based the radiological source term on such an analysis.
 - b) The radiological source term analysis should be based on the assumption that CDAs (failure to scram with substantial core disruption) are credible accidents within the DBA envelope, should place an upper bound on the explosive potential of a CDA, and should then derive a conservative estimate of the fission product release from such an accident. Neither Applicants nor Staff have performed such an analysis.
 - c) The radiological source term analysis has not adequately considered either the release of fission products and core materials, e.g. halogens, iodine and plutonium, or the environmental conditions in the reactor containment building created by the release of substantial quantities of sodium. Neither Applicants nor Staff have established the maximum credible sodium release following a CDA or included the environmental conditions caused by such a sodium release as part of the radiological source term pathway analysis.
 - d) Neither Applicants nor Staff have demonstrated that the design of the containment is adequate to reduce calculated offsite doses to an acceptable level.
 - e) As set forth in Contention 11(d), neither Applicants nor Staff have adequately calculated the guideline values for radiation doses from postulated CRBRP releases.
 - f) Applicants have not established that the computer models (including computer codes) referenced in Applicants' CDA safety analysis reports, including the PSAR, and referenced in the Staff CDA safety analyses are valid. The models and computer codes used in the PSAR and the Staff safety analyses of CDAs and their consequences have not been adequately documented, verified or validated by comparison with applicable experimental data. Applicants' and Staff's safety analyses do not establish that the

models accurately represent the physical phenomena and principles which control the response of CRBR to CDAs.

- g) Neither Applicants nor Staff have established that the input data and assumptions for the computer models and codes are adequately documented or verified.
- h) Since neither Applicants nor Staff have established that the models, computer codes, input data and assumptions are adequately documented, verified and validated, they have also been unable to establish the energetics of a CDA and thus have also not established the adequacy of the containment of the source term for post accident radiological analysis.

Contention 3(c):

Accidents associated with core meltthrough following loss of core geometry and sodium-concrete interactions have not been adequately analyzed.

Contention 11(d)

Guideline values for permissible organ doses used by Applicants and Staff have not been shown to have a valid basis.

- (1) The approach utilized by Applicants and Staff in establishing 10 CFR § 100.11 organ dose equivalent limits corresponding to a whole body dose of 25 rems is inappropriate because it fails to consider important organs, e.g. the liver, and because it fails to consider new knowledge, e.g., recommendations of the ICRP in Reports 26 and 30.
- (2) Neither Applicants nor Staff have given adequate consideration to the plutonium "hot particle" hypothesis advanced by Arthur R. Tamplin and Thomas B. Cochran, or to the Karl Z. Morgan hypothesis described in "Suggested Reduction of Permissible Exposure to Plutonium and Other Transuranium Elements," Journal of American Industrial Hygiene (August 1975).

**PART 1. Site Suitability Source Term Analysis
(10 CFR 100.11)**

**I. STAFF HAS NOT CORRECTLY PERFORMED
THE DOSE BONE SURFACE CALCULATIONS
IN THE SITE SUITABILITY SOURCE TERM ANALYSIS**

A. Introduction

61. The dose guidelines specified by Staff to evaluate the consequences of the postulated site suitability source term (SSST) release to an assumed individual at the exclusion area (EA) boundary and outer boundary of the low population zone (LPZ) are those specified in 10 CFR Part 100.11 (300 rem thyroid and 25 rem whole body) with the following additional guidelines for potentially critical organs: 75 rem for the lung, and 300 rem for bone surfaces, coupled with the additional guideline that the mortality risk equivalent whole body dose from any postulated design basis accident (on a calculated dose basis) for the CRBR should be no greater than the mortality risk equivalent whole body dose value of 10 CFR Part 100 for an LWR (i.e., 34 rem whole body risk equivalent at the operating license stage, and 24.5 rem whole body risk equivalent at the construction permit stage). (Staff Exh. 1, 1982 SSR, p. III-9).

62. The dose guidelines specified by Staff for use during the construction permit review are 150 rem to the thyroid, 20 rem to the whole body, 35 rem to the lung, and 150 rem to bone surfaces. (Staff Exh. 1, 1982 SSR, p. III-9).

63. Staff has calculated the dose consequences at the LPZ boundary resulting from the SSST release as 7 rem to the thyroid, 0.3 rem to the whole body, 0.4 rem to the lung, and 9 rem to the bone. (Staff Exh. 1, 1982 SSR, p. III-11).

64. As demonstrated below, Staff's bone (surface) dose calculations are in error in at least the following respects:

- a) failure to use current dosimetric and metabolic models;
- b) failure to use conservative plutonium isotopic concentrations;
- c) failure to consider the dose from the entire passage of the radioactive cloud (10 CFR 100.11(a)(2));
- d) failure to consider radioactive releases via the containment vent/purge system;
- e) failure to assume a boundary fuel release from the core;
- f) failure to consider the integrated dose commitment beyond 50 years.

B. The SSST Analysis Was Not Performed Using Current Dosimetric and Metabolic Models and the Analysis Failed to Properly Calculate Internal Organ Exposures.

65. Staff used the same bone dose commitment factor (DCF) for plutonium isotopes in the SSST analysis (Staff Exh. 1, 1982 SSR) that Staff was using in 1976. (Morgan, Int. Exh. 9, p. 16, Tr.

3134). This bone dose commitment factor, computed in NUREG-0172, was based on the dosimetric and metabolic models of ICRP Publications 2, 6 and 10. (Morgan, Int. Exh. 9, pp. 16-17, Tr. 3134-3135).

66. 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 (Morgan, Tr. 3163), 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. (Morgan, Tr. 2957-2958; McClellan, Tr. 1915).

67. The ICRP dosimetric and metabolic models employed by ICRP-30 are more appropriate for calculating organ doses to the bone surface, thyroid and lung, and represent a more up-to-date view of our knowledge than those in ICRP-2. (Thompson, Tr. 1902-3, 1907). Both Applicants and Staff now use the newer ICRP-30 models in calculating doses. (Bell, Tr. 2360-61; Branagan, Tr. 2389-90; Hibbits, Tr. 5218-19; App. Exh. 46, p. 33, Tr. 5409).

68. Although Staff specified a Construction Permit (CP) bone surface dose guideline of 150 rem (Staff Exh. 1, 1982 SSR, p. III-9), Staff failed to calculate the bone surface doses that would result from an SSST accident. Instead, Staff calculated an LPZ bone dose of 9 rems. (Staff Exh. 1, 1982 SSR, p. III-11). Use of the conversion factors reported in NUREG/CR-0150, which

are based on the newer ICRP models, is the more appropriate way to calculate bone surface doses. (Morgan, Tr. 3134-3136; Finding 99).

69. Calculating the LPZ bone surface dose from Staff's SSST release results in a dose of 27 rems, which is a factor of 3 higher than the LPZ bone dose calculated by Staff. (Morgan, Int. Exh. 9, p. 10, Tr. 3128, Table 1; Strawbridge, Tr. 5157; Staff Exh. 8, FSFES, App. J, p. J-2).

C. The SSST Analysis Fails 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.

70. In calculating the SSST dose at the exclusion area and LPZ boundaries, Staff assumed that the plutonium had the following isotopic concentrations (weight %): 1% Pu-238; 74% Pu-239; 20% Pu-240; 5% Pu-241; and 0% Pu-242. (Morgan, Int. Exh. 9, p. 10, Tr. 3128).

71. The assumptions made regarding plutonium isotopic concentration are of importance in the SSST analysis because these isotopic concentrations affect the doses calculated from the SSST release. (Cochran, Int. Exh. 13, p. 23-24, Tr. 4589-90; Strawbridge, Tr. 1748). In fact, the isotopic concentrations of Pu-238 and Pu-241 are controlling in terms of bone dose and bone surface dose. (Id.; Morgan, Int. Exh. 9, pp. 11-12; Tr. 3129-30).

72. Although 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 Mwd/MT. (Morgan, Int. Exh. 9, p. 12, Tr. 3130; Cochran, Int. Exh. 13, p. 23-24, Tr. 4589-90; Strawbridge, Tr. 1751, 5164). This can be seen from columns labeled 1-4 in Table 1 below:

TABLE 1

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

$^*\text{Pu}_f = ^{239}\text{Pu} + ^{241}\text{Pu}$

(Morgan, Int. Exh. 9, p. 13, Tr. 3131).

73. Assuming the use of high burnup LWR plutonium fuel in the CRBR would increase the plutonium source term in the two controlling isotopes, Pu-238 and Pu-241, by a factor of at least 2 to 4. (Yarbro, Tr. 4265). This would correspondingly increase the bone surface dose by a factor of up to 4.3. (Cochran, Int. Exh. 13, p. 24, Tr. 4590).

74. It is appropriate to assume, for purposes of the SSST dose calculations, that CRBR will be fueled at some point with plutonium recovered from high burnup LWR spent fuel with the higher concentrations of Pu-238 and Pu-241 comparable to those in column 3 in Table 1 above (Findings 154 to 166); and that Staff's bone/bone surface doses would be increased accordingly by a factor of 4.3.

75. Correcting this factor would lead to a bone surface dose at the LPZ boundary of 116 rems ($27 \text{ rems} \times 4.3 = 116 \text{ rems}$).

D. The SSST Analysis Fails to Consider the Dose From the
Radioactive Cloud "During the Entire Period of its Passage"
(10 CFR 100.11(a)(2))

76. In considering doses to an individual at the LPZ boundary Staff truncated the calculations at the end of 720 hours, or 30 days. (Staff Exh. 1, 1982 SSR, Table IV, p. III-11, Morgan, Int. Exh. 9, p. 9, Tr. 3127). The emissions from the postulated SSST accident, however, would continue after the 30-day period. (Bell, Tr. 2353).

77. In the case of the CRBR, and unlike LWRs, the LPZ dose would be significantly larger if the dose included the effect of post-30 day releases as modeled by a "puff release" at the end of 30 days (the worst-case condition), than doses calculated only for the first 30 days. (Bell, Tr. 2399). Unlike an LWR, the releases and dose consequences after 30 days cannot be considered negligible in a SSST analysis. (Bell, Tr. 2399). The bone surface dose at the LPZ boundary would increase by a factor of 4.3 if one accounts for post-30 day releases by adding a "puff release" to the 30-day dose calculated by Staff. (Morgan, Int. Exh. 9, p. 10, Tr. 3128; Bell, Tr. 2356).

78. The "puff release" used in the above calculation assumes that at the end of 30 days the emissions remaining in the containment are essentially instantaneously released (actually released over a 1 hour period), or "puffed" to the environment through the annulus filtration system. (Bell, Tr. 2356). This "puff release" calculation incorporates the appropriate degree of conservatism for the SSST analysis with respect to treatment of post-30-day releases (Bell, Tr. 2354), and is more appropriate and realistic than calculations that do not consider any emissions after a 30-day period. (Bell, Tr. 2350-51, 2355).

79. It is not appropriate to assume, for purposes of the SSST analysis, that 90 percent of the 30-day dose would occur as a result of releases in the first day, or that 98 percent of the 30-day dose would result from releases in the first week.

(Strawbridge, Tr. 1831). These calculations assume a "realistic" aerosol depletion rate that is not appropriately conservative for purposes of the site suitability source term analysis. (See generally, Staff Exh. 3, Attachment A, pp. 8-17, Tr. 2553-62). Staff's SSST analysis takes credit for aerosol depletion inside the containment only during the first 24 hours (Bell, Tr. 2358-59, 2401; Morgan, Int. Exh. 9, p. 10, Tr. 3128), whereas Applicants assume the aerosol depletion continues for the full 30 days. (Strawbridge, Tr. 1742).

80. Taking into account the release after 30 days modeled by a "puff release," the LPZ bone surface dose at the LPZ boundary would increase by a factor of 4.3, from 116 rems (Finding 75) to 500 rems, well above the 150 rem bone surface dose guideline value specified by Staff for use at the CP (and LWA-1) stage.

E. The SSST Analysis Fails to Include the Radioactive Releases From Containment Via the Containment Vent/Purge System.

81. Staff's SSST assumes a fission product release to containment of 100% of the noble gases, 50% of the iodines, 1% of the solid fission products, and 1% of the plutonium from the core. (Staff Exh. 1, 1982 SST, p. III-11). Loss of core coolable geometry is a necessary prerequisite in order to release these sizable fractions of halogens, iodine, fission products and plutonium fuel from the CRBR; in fact a CDA -- either a core meltdown or an energetic CDA -- involving the whole core or a substantial fraction of the core, has to occur. (Cochran, Int.

Exh. 4, pp. 23-25, Tr. 3072-3076). The SSST assumed for purposes of evaluating LWR sites is based on a CDA -- substantial core meltdown -- in an LWR. (Staff Exh. 3, Attachment A, p. 11, Tr. 2556) (10 CFR § 100.11(a), fn. 7). Staff also based its proposed CRBR source term on the occurrence of a CDA. (Cochran, Int. Exh. 4, p. 23, Tr. 3073).

82. In the event of a CDA in CRBR with substantial core involvement, the meltthrough of the reactor vessel would occur at approximately 1000 seconds (Cochran, Int. Exh. 4, p. 24, Tr. 3074; App. Exh. 1, pp. 66, 69, Tr. 2055, 2058), and about 1.1 million pounds of sodium would be dumped into the reactor cavity. (Cochran, Int. Exh. 4, p. 24, Tr. 3074; App. Exh. 1, p. 66, Tr. 2055).

83. The characteristics of a reactor of the general size and type as the CRBR are different from the characteristics of an LWR with regard to mitigation of CDAs. (Strawbridge, Tr. 5143). The containment capability of the two plant types is also different. (Strawbridge, Tr. 5143). CRBR has a number of specific active-component features, including an annulus cooling filtration system and a containment vent/purge system, which are designed to mitigate CDAs. (Strawbridge, Tr. 5144-45). The annulus cooling/filtration system and the containment vent/purge system are unique to the CRBR. (Strawbridge, Tr. 5144-47).

84. Given the proposed CRBR design, a CDA with substantial core involvement, would require activation of the containment

vent/purge system to avoid containment failure due to pressure and thermal effect resulting from sodium releases. (App. Exh. 46, p. 32, Tr. 5408; App. Exh. 47, p. 6 Tr. 5420; Cochran, Tr. 3075; Clare, Tr. 1880; App. Exh. 1, p. 66, Tr. 2054-55). The vent system pulls air from inside the containment through a radioactivity removal system directly to the atmosphere in the event of a core melt accident or energetic CDA. (App. Exh. 1, p. 55, Tr. 2044 and references therein, pp. 68-69, Tr. 2057-58; illustrated at App. Exh. 17, CRBRP-3, Vol. 2, Secs. 1, 2.1, 2.2, pp. 1-6, 2-1 - 2.10, 2-24, 76).

85. Staff's SSST analysis assumes that radiological releases to the environment, even from the most severe accident, will occur only via the annulus filtration system (App. Exh. 1, p. 50, Tr. 2039) and via bypass leakage at the design basis leak rate of 0.001% per day. (Staff Exh. 1, 1982 SSR p. III-11; Staff Exh. 3, p. 23, Tr. 2506). Staff has failed to incorporate the effects of the vent/purge system in the SSST analysis and has thus failed to calculate the LPZ boundary doses that would result from additional radioactive releases from the containment through the vent/purge system. (Staff Exh. 1, 1982 SSR, p. III-11; Thadani, Tr. 5664-65).

86. The LPZ doses from the SSST release are not very sensitive to the time the vent/purge system is first activated (between 10 to 36 hours) (App. Exh. 1, pp. 72-73, Tr. 2060-61), but the doses are very sensitive to whether the vent/purge system is activated

at all. (Strawbridge, Tr. 5217-18). This can be seen by comparing Staff and Applicants' so-called "conservative" SSST dose calculations for nearby facilities with the supposedly more "realistic" CDA dose calculations for the same facilities. (Staff Exh. 18, pp. 6-7, Tr. 5688-89; App. Exh. 47, pp. 8, 11, 13-14, Tr. 5428, 5437, 5433-34). The SSST analysis under 10 CFR Part 100 must use very conservative assumptions compared to a "realistic" CDA analysis. (Staff Exh. 3, Attachment A, pp. 8-17, Tr. 2553-62). The SSST analysis performed for CRBR used more conservative assumptions than the CDA analysis with regard to X/Q values (Thadani, Tr. 5665-66, 5672, Staff Exh. 1, 1982 SSR, p. III-10-11; App. Exh. 47, p. 6, Tr. 5426), aerosol depletion and fallout within the containment, (Finding 79), plutonium and fission product released to containment (source term) (Staff Exh. 3, pp. 10-15, Tr. 2493-98), and presumably other parameters as well. Yet the resulting SSST doses are generally lower than the more realistic CDA doses. (App. Exh. 47, p. 11, Tr. 5431). The resulting CDA thyroid dose at ORGDP, for example, is $312 (100 \div 0.32 = 312)$ times higher than the SSST thyroid dose, as calculated by Staff, and $12.9 (7.1 \div 0.55 = 12.9)$ times higher, as calculated by Applicants. Similarly, the CDA whole body dose at ORGDP is $15.8 (3 \div 0.19 = 15.8)$ times higher than the SSST whole body dose, as calculated by Staff, and $1.7 (0.17 \div 0.1 = 1.7)$ times higher, as calculated by Applicants. (Staff Exh. 18, pp. 6-7, Tr. 5688-89; App. Exh. 47, pp. 8, 13-14, Tr. 5428, 5433-34).

Staff did not calculate a SSST bone surface dose at ORGDP (Thadani, Tr. 5675, 5678) and therefore a comparison of CDA and SSST bone surface doses as calculated by Staff is not made. The CDA bone surface dose at ORGDP is only a factor of 4 less ($0.339 \div 1.364 = 0.25$) than the SSST bone surface dose as calculated by Applicants. By failing to include operation of the vent/purge system in the SSST analysis (Thadani, Tr. 5664; Hibbits, Tr. 5216), therefore, Staff and Applicants have offset virtually every conservative assumption built into the SSST analysis.

87. The LPZ bone surface dose for the SSST, therefore, should be increased by an unknown factor (500 rem x unknown factor, where 500 rem is from Finding 80). This dose, which already exceeds the 150 rem dose guideline value specified by Staff for use in the CP hearing, would exceed the 150 rem dose guideline by a wide margin. In addition, the record is inadequate to determine the effect of including the vent/purge system in the SSST analysis on other organ doses, such as lung, thyroid, and liver. Thus no judgment can be made as to whether those doses exceed 10 CFR Part 100 guidelines.

F. The SSST Assumed Fuel Release Is Not Bounding.

88. In the SSST analysis, Staff has assumed a radiological source term consisting of the usual LWR source term assumed to be released from the core, plus 1% of the plutonium in the core. (Staff Exh. 1, 1982 SSR, p. III-8). This source term does not

bound the consequences of a major core disruptive accident. (Cochran, Int. Exh. 4, p. 13, Tr. 3063; Staff. Exh. 8, FSFES, App. J, p. J-10).

89. Since core disruptive accidents are in fact credible (Findings 1 to 23, 55 to 60), the plutonium fraction of the site suitability source term must be increased to bound CDAs. Staff and Applicants have not performed the necessary analysis to determine the appropriate source term to bound credible CDAs. (Morris, Tr. 2274; Cochran, Int. Exh. 4, pp. 16-17, Tr. 3067-68).

90. The assumed plutonium release from the core must be increased by at least a factor of 10 in order to bound CDAs. (Cochran, Int. Exh. 4, p. 22, Tr. 3072). Staff's assumed fission product for Applicants' Parallel Design, in which a CDA was considered a credible accident within the design basis, included 10% of the plutonium from the core. (Cochran, Int. Exh. 4, p. 13, Tr. 3063). Applicants' own analyses of CDAs have postulated the release of up to 10% of the plutonium from the core. (Cochran, Int. Exh. 4, p. 22, Tr. 3072). Even larger SSST plutonium functions have been used in the past to bound CDAs in other reactors. (Cochran, Int. Exh. 4, pp. 13-14, Tr. 3063-64). The maximum capacity for harm from an LMFBR accident has been estimated to be an order of magnitude greater than that from an LWR, which is not reflected in Staff's choice of a 1% plutonium source term. (Cochran, Int. Exh. 4, p. 22, Tr. 3072). A plutonium fraction of at least 10% is necessary in

order to reach a sufficient level of conservatism in the site suitability analysis, to account for uncertainties in the novel design of the CRBR, and to account for later design modification and review of the CRBR design. (Cochran, Int. Exh. 4, pp. 18-19, Tr. 3068-69).

91. Even if it is not demonstrated that the CDA should be included within the DBA envelope at the LWA-1 licensing stage, the plutonium release fraction should still be increased by at least 10 to account for the substantial possibility that CDAs will be found credible after a full safety review. (Cochran, Tr. 3070-72).

92. The LPZ bone surface dose estimated by Staff in its site suitability analysis should be increased by a factor of 10 ((500 rem x unknown factor) x 10 = 5000 rem x unknown factor = more than 5000 rem, where 500 rem x unknown factor is from Finding 87), to account for release of up to 10% plutonium from the core during an SSST accident. This dose far exceeds the 150 rem bone surface dose guideline value specified by Staff for its CP (and LWA-1) review,

G. The SSST Analysis Fails to Consider the Entire Life of the Maximally Exposed Individual by Integrating Dose Commitment Beyond 50 Years.

93. 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. 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. (Morgan, Tr. 3173).

94. 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. (Morgan, Tr. 3174).

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

96. Applying this correction to Staff's estimate of the LPZ bone surface dose, and including the corrections in Findings 65 to 92, would yield an LPZ bone surface dose of more than 7500 rem $((5000 \text{ rem} \times \text{unknown factor}) \times 1.5 = 7500 \text{ rem} \times \text{unknown factor})$ where 500 rem x unknown factor is from Finding 92) -- over 50 times the 150 rem LPZ bone surface dose guideline specified by Staff for use in its CP (and LWA-1) review.

H. Conclusion

In sum, based on Findings 61 to 96 above, the SSST bone dose of 9 rem at the LPZ boundary as reported in the 1982 SSE (Staff Exh. 1, p. III-11), should be multiplied by the following factors to obtain the appropriate bone surface dose for comparison against the 10 CFR Part 100 guideline values:

<u>Factor</u>	<u>Basis</u>
3	to obtain bone surface dose, rather than bone dose, calculated with current dosimetric and metabolic models
4.3	to correct for potential use of plutonium from high burnup LWR spent fuel
4.3	to correct for emissions after 30 days (passage of the entire cloud)
unknown factor	to correctly model the releases of radioactivity through the vent system
10	to correct the plutonium release fraction to bound CDAs
1.5	to convert from a 50-year dose commitment to an 80-year dose commitment.

Thus, the appropriate value = 9 rem x 3 x 4 x 3 x 4.3 x unknown factor x 10 x 1.5 = (7500 x unknown factor) rem to bone surface, which is well above the dose guideline values specified by Staff for the CP (and LWA-1) review.

II. THE DOSE GUIDELINE VALUES SELECTED BY STAFF
FOR USE IN THE SITE SUITABILITY REVIEW
ARE INADEQUATE

A. Staff's Failure to Reduce the Dose Guideline Values at the
Construction Permit Stage to Account for Uncertainties is
Unsupported By the Record.

97. In the 1977 SSR, 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. Cochran, Int. Exh. 4, p. 31, Tr. 3081; Staff Exh. 3, p. 30, Tr. 2513).

98. In the 1982 SSR (Staff Exhibit 1, p. III-9), 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. (Branagan, Staff Exh. 3, pp. 30-31, Tr. 2513-14). This reduction is unwarranted by the record, since the uncertainties in the estimates of lung and bone surface doses due to plutonium (which is controlling) continue to exceed a factor of 10, as indicated below.

99. The first evidence of possible nonconservatism in plutonium dose estimates is set forth by Dr. Karl Z. Morgan in the American Journal of Industrial Hygiene (August 1975) (the "Morgan hypothesis").

a. The current plutonium-239 standard (based on ICRP-2) was established using 0.1 microcuries of radium-226 as the reference standard. (Morgan, Int. Exh. 9, p. 23, Tr. 3141; App. Exh. 25, p. 10, Tr. 2084). Deriving the bone surface dose directly from the radium-226 standard based on the Morgan hypothesis 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. (Morgan, Tr. 2960-2961; Morgan, Int. Exh. 9, pp. 21-24, Tr. 3139-3142).

b. Applying the Morgan hypothesis would increase Staff's estimate of the bone dose by a factor of 240. (Morgan, Int. Exh. 9, p. 23, Tr. 3141). By the same token, current Commission standards for plutonium exposure are too high by a factor of 240. (Morgan, Int. Exh. 9, p. 23, Tr. 3141; Cochran, Int. Exh. 4, p. 32, Tr. 3082). 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. (Morgan, Int. Exh. 9, p. 23, Tr. 3141; Cochran, Int. Exh. 4, p. 32, Tr. 3082).

c. Applicants 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 (Thompson, McClellan, Tr. 1912-1915). Application of the newer ICRP dosimetric and metabolic models leads to a bone surface dose that is 3 times the bone dose calculated using the older ICRP models (Findings 65 to 69). Using the dosimetric and metabolic models employed in ICRP-30 as a reference, and accepting Morgan's thesis, the quality factors used in the ICRP-30 methodology would have to increase by a factor of 80 ($240 \div 3$) in order to be fully consistent with the numerical result under the Morgan thesis.

d. Applicants claim 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. (App. Exh. 25, p. 11, Tr. 2085). This claim is incorrect, as evidenced by the factor of 3 difference between the bone dose (calculated assuming a 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 Finding 69.

e. Applicants claim that the Morgan thesis is "misplaced in the context of the NRC Staff's 1982 recommended dose guidelines since ... these were not derived from the ICRP-2 methodology but from the 10 CFR 100.11(a) thyroid dose guidelines and scaling factors from ICRP-26". (App. Exh. 25, pp. 10-12, Tr. 2084-86). This claim is incorrect. The ICRP weighting factors are a measure of the stochastic risk associated from a given organ or tissue exposure in rem, relative to the risk associated with uniform whole body exposure of the same amount. (App. Exh. 25, p. 5, Tr. 2079). Morgan's thesis implies the quality factors used to calculate bone surface dose as currently applied by ICRP are incorrect (Cochran, Int. Exh. 4, p. 32, Tr. 3082), and therefore the ICRP-26 weighting factor for bone surface is also incorrect.

100. A second example of possible nonconservatism in the plutonium dose estimates 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." (Cobb, Int. Exh. 8, pp. 1-2, Tr. 3101-02; Cochran, Int. Exh. 4, p. 32-33, Tr. 3082-83).

a. Martell noted, and his argument has been supported by others, in a series of Letters to the Editor appearing in the

New England Journal of Medicine, Vol. 307, 29 July 1982, 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 22 compared to the factor of 10-20 currently assumed." (Cochran, Int. Exh. 4, p. 33, Tr. 3083). Staff's witnesses had virtually no familiarity with this hypothesis. (Branagan, Tr. 2336).

b. Applicants claim that the warm particle hypothesis is "a working hypothesis that is not really with a proven foundation today." (McClellan, Tr. 4043-44). Although the warm particle hypothesis is not "proven," neither is the hypothesis currently accepted by Staff and Applicants -- that the risk associated with warm (or hot) particles can be conservatively treated by assuming the alpha irradiation is smeared uniformly throughout the organ. (Cochran, Int. Exh. 4, p. 34, Tr. 3084). The warm particle hypothesis nevertheless demonstrates continued uncertainty regarding the validity of the current hypothesis, as evidenced by the BEIR-III analysis of this issue. (Cochran, Int. Exh. 4, p. 35, Tr. 3085; (See Finding 104).

101. A third example of possible nonconservatism in the plutonium dose estimates is the evidence presented by Dr. John C. Cobb (Cobb, Int. Exh. 8, pp. 1-9, Tr. 3101-3109) 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. (Cobb, Int. Exh. 8, p. 1, Tr. 3101). Cobb's concern was based on the findings of recent research in four related areas:

a. The findings of his EPA-contracted study of plutonium burdens in the post-mortem tissues of people who had lived near the Rocky Flats plutonium weapons facility;

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

c. 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;

d. The findings of animal experiments showing that plutonium and other alpha-emitters cause mutations and genetic defects as well as cancers. (Cobb, Int. Exh. 8, p. 2, Tr. 3102).

e. Cobb concluded, based on his findings (Cobb, Int. Exh. 8, pp. 3-5, Tr. 3103-3105) that "we may have underestimated the toxicity of plutonium by a large factor and we have probably overestimated our ability to control it, as shown by our experience with the Rocky Flats plutonium weapons facility." (Cobb, Int. Exh. 8, p. 8, Tr. 3109).

f. The plutonium burden in humans near Rocky Flats, a plutonium facility (Cobb, Tr. 2898), suggests 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. (Cobb, Tr. 2888, 2919).

102. A fourth example of possible nonconservatism in current plutonium dose estimates is evidenced by the work of Dr. Carl J. Johnson, which questions the adequacy of the scientific basis for the existing plutonium standards, namely ICRP-2 and the proposed EPA guidance for plutonium soil contamination (EPA 520/4-77-016).

a. Dr. Johnson believes the maximum permissible body burden of 0.04 microcuries specified in ICRP-2, which forms the basis for current NRC regulation of plutonium, is not sufficiently protective, based upon the work of Drs. Kocher, Morgan, Meyers, Cross, et al., and Barr (Johnson, Int. Exh. 21, pp. 9-11, Tr. 6026-28; Johnson, Tr. 5859, 5869-70, 5922-25).

b. Dr. Johnson also believes that the proposed EPA guidance regarding maximum permissible soil concentration limits for plutonium (EPA 520/4-77-16) is inadequate, since it is ten times less protective than an Interstate Commerce Commission guideline limiting contamination of trucks hauling

radioactive materials to less than 4.4 dpm (2 picocuries) of alpha radiation per square centimeter. Dr. Johnson believes the EPA proposed guidance conflicts with the usual public health practice of reducing an occupational concentration limit by up to 100 times, when setting a limit for the general public. (Johnson, Int. Exh. 21, pp. 12-13, Tr. 6029-30).

103. A final example of possible nonconservatism in current plutonium dose estimates is the "hot particle hypothesis" proposed by Arthur R. Tamplin and Thomas B. Cochran in a series of NRDC reports. (Cochran, Int. Exh. 4, pp. 33-34, Tr. 3083-84).

104. While none of the hypotheses cited in Findings 99 to 103 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. (Cochran, Int. Exh. 4, p. 34, Tr. 3084).

a. As 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:

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. (Cochran, Int. Exh. 4, p. 34-35, Tr. 3084-85).

b. Based on the foregoing facts (Findings 97 to 103): where the exposure is due to plutonium (or other alpha emitter) the guideline value for bone surface dose should be reduced by a factor of 80 (See Finding 99); and the guideline value for lung dose should be reduced by a factor of 50 (the difference between an assumed quality factor of 1000 (Finding 100, and a quality factor of 20 assumed by ICRP-30. (Morgan, Tr. 3163)).

B. Staff's Proposed Dose Guideline Levels for Lung and Bone Surface are Too High to Provide Adequate Public Protection.

105. Current Commission regulations (10 CFR Part 100) do not contain any dose guideline values for lung or bone surface. (Cochran, Tr. 3013). There is no internally consistent method for selecting guideline values for bone surface and lung dose that does not conflict with either the whole body or thyroid guideline value in 10 CFR Part 100. From among several alternative means available for selecting such guideline values (Branagan, Tr. 2511; Cochran, Tr. 3013), Staff used the 300 rem

thyroid dose guideline value in 10 CFR § 100.11 coupled with the ICRP-26 weighting factors for lung and bone surface (Branagan, Tr. 2511, Cochran, Int. Exh. 4, p. 28, Tr. 3078). This method is insufficiently conservative to ensure that siting of the CRBR would not result in "serious injury to individuals offsite if the unlikely, but still credible accident should occur." (Cochran, Int. Exh. 4, p. 29, Tr. 3079; Morgan, Tr. 3142).

106. There are three alternative approaches to establishing guideline values for bone and lung which provide better public health protection than the levels selected by Staff. First, the dose guideline values for lung and bone surface could be set at a level consistent with the ICRP-26 limits of 50 rems/year or the EPA proposed dose commitment limit of 30 rems/year, which are intended to prevent non-stochastic effects. (Cochran, Int. Exh. 4, p. 28-29, Tr. 3078-79). Although these non-stochastic limits relate to annual occupational doses, annual dose-equivalent limits can be used to give some indication of where one should properly establish dose guideline values for lung and bone surface to protect public health by avoiding serious injury. (Cochran, Tr. 3004).

107. Alternatively, the lung and bone surface dose guideline values could be established consistent with EPA's regulations regarding annual releases for the uranium fuel cycle during normal operations (40 CFR § 190.10(a)). Applying these EPA regulations, the 25 rem whole body dose guideline value in 10 CFR

Part 100 would correspond to a dose guideline value of 25 rems for lung and bone surface. (Cochran, Int. Exh. 4, p. 30, Tr. 3080).

108. In setting the dose guideline values for lung and bone surface, Staff could establish limits that are consistent with the Environmental Protection Agency's "Proposed Guidance on Dose Limits for Persons Exposed to Transuranium Elements in the General Environment", EPA 520/4-77-160, Sept. 1977 (Morgan, Int. Exh. 9, p. 21, Tr. 3139; Cobb, Tr. 2884, 2890-2893), 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." This guidance 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. (Cobb, Tr. 2913; Morgan, Int. Exh. 9, p. 21, Tr. 3139). 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. (Cobb, Int. Exh. 8, pp. 1, 7, Tr. 3101, 2907; Johnson, Int. Exh. 21, pp. 12-13, Tr. 6029-30). Staff should also consider the Colorado State guidelines for permissible levels of plutonium in the environment (2 disintegrations per minute per gram of soil), which are even stricter than the EPA guidelines, by a factor of about 25. (Cobb, Tr. 2098, Int. Exh. 8, p. 3, Tr. 3103).

**Part 2. Staff and Applicants Have Failed to
Demonstrate That CRBR Accident Risks
Are Comparable to LWR Accident Risks**

I. INTRODUCTION

109. In licensing the CRBR, Staff has taken the basic position that the CRBR should achieve a level of safety comparable to current generation light water reactor (LWR) plants, according to all current criteria for evaluation; that there be no greater than one chance in one million per year for potential consequences greater than the 10 CFR 100 dose guidelines for the CRBR; and that the CRBR containment system should be protected from the unique effects of CRBR core disruptive accidents in order to maintain comparability with LWR safety. (Staff Exh. 5, pp. 1-2,5).

110. In Appendix J of Staff's CRBR Final Environmental Impact Statement Supplement (Staff Exh. 8), Staff estimated the probabilities and consequences of CRBR core disruptive accidents, and, based upon those estimations, concluded that "CRBRP accident risks would not be significantly different from those of current LWRs." (Staff Exh. 8, FSFES, p. J-25). This conclusion is unsupported by the record, because the uncertainties in Staff's CDA Appendix J probabilistic risk assessment are too great, and because Staff's CDA consequences analysis is limited to calculations of "average risk values." Staff's failure to analyze and compare doses to the maximally exposed individual

provides an insufficient basis for assuring comparability with LWR safety.

II. THE UNCERTAINTIES IN STAFF'S CDA PROBABILISTIC RISK ASSESSMENT

A. General Uncertainties

111. Staff judges that the uncertainty bounds in its Appendix J estimates of CDA probabilities and consequences could be "well over a factor of 10 and may be as large as a factor of 100, but is not likely to exceed a factor of 100." (Staff Exh. 8, FSFES, App. J, p. J-24). By way of comparison, in WASH-1400, a comprehensive probabilistic risk assessment of two light water reactors, the uncertainties associated with its probability estimates ranged as high as a factor of 100. (Cochran, Int. Exh. 22, p. 7, Tr. 6201). In fact, the Commission found that the WASH-1400 accident probability estimates are "unreliable." (Cochran, Int. Exh. 22, p. 43, Tr. 6237). Yet the WASH-1400 analysis and methods are far superior to those in Appendix J. Appendix J was prepared hurriedly and is not supported by any calculations, fault tree/event tree analyses, other plant-specific risk assessments or explicit assumptions, and fails to reference its background assumptions. (Cochran, Int. Exh. 22, p. 6, Tr. 6200). Staff's attempt to qualitatively assess the uncertainty associated with its risk estimates is a one-sentence conclusory statement (Staff Exh. 8, FSFES, p. J-24), unsupported

by rigorous analysis. (Cochran, Int. Exh. 22, pp. 5-8, Tr. 6199-6202). Staff, for example, could not state whether the relative uncertainties in its 10^{-4} /reactor year probability estimate for Anticipated Transients Without Scram ("ATWS") events is less or more than the probability estimates in WASH-1400. (Morris, Tr. 5633).

112. Staff has failed to back up its probability estimates for CDA initiators, the conditional frequency of energetic CDAs, and containment failure. (Findings 113 to 119). Staff has also failed to analyze adequately CRBR common cause failures and systems interaction. (Findings 120 to 122). Applicants' claim that Staff's estimates are conservative is based upon observations of alleged conservatism in Staff's analysis, but omits substantial offsetting nonconservatism which also exist. (Findings 123 to 126).

B. Estimates of Loss of Heat Sink (LOHS) Probability

113. Staff estimated the probability of a CRBR core disruptive accident (CDA) due to loss of heat sink (LOHS) events at less than 10^{-4} per reactor year, based upon a general consideration of typical achievable PWR auxiliary feedwater system reliabilities (believed to be an important contributor to LOHS events), the potential for common cause failures, and the assumption of an effective reliability program. (Staff Exh. 8, p. J-4; Rumble, Tr. 5450, 5512-13, 5519). This probability estimate is

unsupported by the evidence in the record, and constitutes an unacceptably high risk to the public. (Cochran, Int. Exh. 22, pp. 10-16, Tr. 6204-6210).

114. Staff's choice of auxiliary feedwater system (AFWS) failure as the controlling failure mode for LOHS events is not justified. There is insufficient evidence in the record to establish that failures in systems other than auxiliary feedwater do not contribute significantly to the LOHS probability. A fault tree/event tree analysis, which Staff has not performed (Rumble, Tr. 5579), is necessary to justify limiting the discussion to CRBR auxiliary feedwater reliability. (Cochran, Int. Exh. 22, p. 11, Tr. 6205; Attachment 1, Tr. 6240-49). For example, neither Applicants nor Staff have analyzed the contribution of steam generator failure to the overall risk of LOHS events, or considered the possible mechanisms or modes of steam generator failure. (Cochran, Int. Exh. 22, pp. 15-16, Tr. 6209-10. Attachment 2, Tr. 6250-60).

115. There are several major differences between the CRBR and a PWR in the cooling systems design, and consequently differences in potential LOHS accident sequences and failure modes. (Rumble, Tr. 5577-78). Because of these differences, there is no obvious correlation between PWR system reliabilities and CDA frequency due to LOHS accident scenarios in CRBR. (Cochran, Int. Exh. 22, pp. 12-14, Tr. 6206-08). Without a detailed CRBR fault tree/event tree analysis, Staff is not justified in relying upon

estimates of PWR auxiliary feedwater system reliabilities as a basis for a CRBR LOHS frequency estimate of 10^{-4} per reactor year. (Cochran, Int. Exh. 22, pp. 12-14, Tr. 6206-08).

Consequently, Staff erred in applying WASH-1400 PWR probability estimates to the question of unavailability of decay heat removal systems for CRBR. (Cochran, Int. Exh. 22, pp. 13-14, Tr. 6207-08).

C. Other Probability Estimates

116. Staff's conclusion that the estimated 10^{-4} /year LOHS core degradation sequence adequately bounds the flow blockage contribution to core disruption frequency (Staff Exh. 8, FSFES, p. J-4) is unsupported by the record. Staff has no quantitative probability estimate of CDAs from flow blockage (Morris, Tr. 5612-13), and Staff's testimony on the potential for mechanistic deposition of debris or loose parts, the characterization of the mitigation system as active or passive, and the probability of flow blockage from loose parts, is inconsistent. (Morris, Tr. 5610-5612).

117. Staff's conclusion that the 10^{-4} /year frequencies attributed to LOHS, UTOP, and ULOF events adequately bound the contribution to core disruption frequency from fuel failure propagation (FFP) (Staff Exh. 8, FSFES, p. J-5) is unsupported by the record. Staff has not introduced any quantitative probability estimate for fuel propagation (Morris, Tr. 5587-

88). Staff cannot estimate whether the probability of fuel failure propagation is an order of magnitude lower, or even equal, to that of ATWS and LOHS events. (Morris, Tr. 5589-91). Staff's reliance on the tag gas system is unwarranted, since Staff does not know the failure rate of this system. (Morris, Tr. 5592). Staff's reliance on the CRBR quality assurance program, detection systems, and redundant systems to conclude that the CRBR fuel failure propagation probability is lower than LWR ATWS probability is unwarranted, since these systems are also used to prevent ATWS events. (Morris, Tr. 5595-97).

118. Staff estimated the continued frequency of core disruption from LOHS, ATWS, LOCA, and FFP events as less than 10^{-4} per reactor year. (Staff Exh. 8, FSFES, p. J-5). Without any quantitative failure estimates for LOCAs and fuel failure propagation events, even within an order of magnitude (Morris, Tr. 5589-91), however, this combined probability estimate is unsupported by the record. Staff's degree of analysis does not permit it to distinguish between a combined probability of 10^{-4} and that of 2×10^{-4} . (Rumble, Tr. 5614).

119. There is no adequate basis in the record for Staff's conclusion that only 0.1 percent of all CDAs, once initiated, would be highly energetic. (Rumble, Tr. 5615-16; Cochran, Int. Exh. 22, pp. 29-30, Tr. 6223-24).

D. Staff's Treatment of Common Mode Failures and Systems Interaction

120. In its probability estimates for CDA initiation, Staff has not taken adequate account of the potential for common mode failures and systems interaction. (Cochran, Int. Exh. 22, pp. 22-25, Tr. 6216-19). Staff analyzes failure rates for specific CRBR systems such as the shutdown heat removal system, by comparing them to specific LWR systems (see, e.g., Staff Exh. 17, p. 9, Tr. 5756), even though these CRBR systems are not totally independent from other CRBR systems (Rumble, Tr. 5567), and there are significant differences in the CRBR and LWR systems (Finding 27). Comparing failure rates for CRBR and LWR systems has no safety significance unless the systems being compared are truly independent of other systems at the plant and have an equivalent role in performing a post-accident function. For purposes of comparing safety, the appropriate place of comparison is the accident sequence, since it is at this point where all system interdependencies are considered. (NUREG-CR-1659-3 of 4, "The Reactor Safety Study Methodology Applications Program, Calvert Cliff, No. 2 Power Plant," Sandia National Laboratory, Tr. 5560-63).

121. Staff has not performed a fault tree/event tree analysis for the CRBR shutdown heat removal system or other CRBR systems (Rumble, Tr. 5582; Cochran, Int. Exh. 22, pp. 7-8, Tr. 6201-02), and could not have conducted an adequate systems interaction for

CRBR at this time, given the lack of a final CRBR design and of an adequate systems interaction method. (Cochran, Int. Exh. 22, pp. 22-24, Tr. 6216-18). Performing a comprehensive fault tree/event tree analysis, a systems interaction review, or a probabilistic risk assessment might reveal previously undiscovered failure modes or systems interactions. (Rumble, Tr. 5582-83; Clare, Tr. 4975).

122. In conclusion, Staff's CDA probability estimates have large associated uncertainties because of Staff's failure adequately to consider common mode failures and systems interactions. For example, Staff has failed to analyze the probabilities or consequences of a core disruptive accident in which spray fires or missiles also caused immediate containment failure (5188; Rumble, Tr. 5617) and therefore do not know whether Staff's 10^{-4} /year failure probability bounds the probability of common mode failure of the two systems. The evidence shows, however, that such an event has not been considered. Staff's CDA Class 4, which includes a highly energetic CDA plus containment failure, has an estimated probability of 10^{-7} per reactor year. This estimate is based on the probability of 10^{-4} /year that a CDA will occur, the conditional probability of 10^{-1} that such a CDA will be highly energetic, and an independent probability of 10^{-2} that the containment will fail. (Staff Exh. 8, FSFES, p. J-8). Yet if the containment failed directly as a result of the highly energetic CDA, the containment failure would not be independent, and thus the probability would be closer to 10^{-5} than 10^{-7} /year.

E. Applicants' Evaluation of Staff's Probability Estimates

123. Applicants presented evidence of alleged nonconservatism in Staff's estimates of Shutdown Heat Removal System (SHRS) failure (App. Exh. 46, pp. 14-21, Tr. 5390-97), but fail to present offsetting nonconservatism that could impact the ability of CRBR to remove decay heat relative to LWRs. These include:

a. exothermic reactions caused by a water-to-sodium leak, which could cause the loss of a steam generator loop as a decay heat removal path (Clare, Tr. 5000, 5002-03; Strawbridge, Tr. 5029-30);

b. sodium fires caused by contact between sodium and air, which could result in the loss of one or more of the three shutdown heat removal loops (Clare, Tr. 5004-06, 5032; Strawbridge, Tr. 5023, 5030, 5036, 5041);

c. the fact that there is much less operating experience with breeder reactor shutdown heat removal systems involving sodium than with light water reactor shutdown heat removal systems (Clare, Tr. 5006-07);

d. the fact that CRBR natural circulation capability has not been tested, and in fact cannot be tested until the plant is built (Strawbridge, Tr. 5060-61; Cochran, Tr. 6176), causing Staff to allow no credit for CRBR natural circulation capability at this time (Staff Exh. 1, 1982 SSR, p. II-13).

124. Applicants have included in the record no independent probability estimates of CDAs or CDA initiators. (Clare, Tr. 4973-74).

125. Although Applicants claim to have performed a key systems review, some of which is referenced in PSAR Appendix C, Reliability Program, (Clare, Tr. 5247), Applicants have not introduced the results of such review into evidence, and have not relied upon such reviews in any way for their testimony and conclusions regarding the environmental risks and consequences of core disruptive accidents. (Clare, Tr. 5288-91).

126. Applicants have no idea what the uncertainties are in each of Applicants' CDA case probability estimates. (Strawbridge, Tr. 5185). Applicants state that Staff's probability estimates for its four CDA classes are conservative, because each CDA class assumes the most severe primary system failure category (an energetic CDA) and does not analyze each primary system failure category separately. (App. Exh. 46, p. 36, Tr. 5412). This supposed conservatism does not apply to Staff CDA Class 1, in which releases are relatively insensitive to the magnitude of the head release, or to Staff CDA Class 4, in which only one primary system failure category is initially assumed. (Strawbridge, Tr. 5073, 5076, 5082). Applicants have not analyzed the extent of this claimed conservatism in Staff's CDA Classes 2 and 3. (Strawbridge, Tr. 5083).

F. Staff's Estimates of CDA Consequences

127. The radioactive source terms assumed by Staff for its estimate of CDA consequences are highly design dependent, are not supported by analysis or documentation, and could be at least a factor of 3 higher. (Cochran, Int. Exh. 22, pp. 32-33, Tr. 6226-27).

128. The CRAC model utilized by Staff assumes an LD_{50/60} (lethal dose to 50% of the exposed population within 60 days) of 510 rads, a value that is subject to considerable controversy. (Cochran, Int. Exh. 22, pp. 33-34, Tr. 6227-28). By using this assumption, Staff may have underestimated the number of fatalities by a factor of two to four. (Cochran, Int. Exh. 22, p. 34, Tr. 6228).

129. Staff's CRAC consequences code contains several hidden unrealistic assumptions regarding the cancer risk estimator for latent cancers, which appear substantially to reduce the estimate of latent cancer fatalities, exclusive of thyroid cancers, by a factor of 2 to 2.5 compared to the estimate one would obtain using the cancer risk coefficient the Staff purports to use elsewhere in the FSFES. (Cochran, Int. Exh. 22, pp. 35-36, Tr. 6229-30).

III. STAFF'S FAILURE TO ANALYZE AND COMPARE
CRBR AND LWR ACCIDENT RISKS
TO INDIVIDUALS LIVING NEAR THE PLANTS

130. Staff estimated the average risk to populations within 80 kilometers, i.e. probability of early fatalities, latent cancer fatalities, and other costs, based on selected CRBR accidents. (Staff Exh. 8, FSFES, App. J, pp. J-12 - J-16, Table J.5).

131. Staff did not present dose estimates for each of its CDA classes, or indicate in more than a general way how it arrived at the figures in FSFES Appendix J, Table J.5.

132. Staff did not present dose estimates for each of its CDA classes at the EA or LPZ boundaries to analyze whether the risk to the maximally exposed individual, and populations close to the CRBR site, are comparable to or less than exposures from severe LWR accident risks. (See generally Staff Exh. 8, pp. J-10 to J-16; Findings 2 to 17).

IV. CONCLUSION

Based upon the above facts, the record does not support a conclusion that the risks associated with CRBR accidents are comparable to the risks from current light water reactors, or that the risks from CRBR accidents can be made acceptably low.

Contention 5(b):

Since the gaseous diffusion plant, other proposed energy fuel cycle facilities, the Y-12 plant and the Oak Ridge National Laboratory are in close proximity to the site an accident at the CRBR could result in the long term evacuation of those facilities. Long term evacuation of those facilities would result in unacceptable risks to the national security and the national energy supply.

I. STAFF AND APPLICANTS HAVE NOT ADEQUATELY ANALYZED
THE IMPACTS OF CRBR ACCIDENTS UPON THE Y-12 FACILITY

A. Introduction

133. The Y-12 plant is a major facility within the Department of Energy's nuclear weapons production complex. The plant produces components and subassemblies in support of the production of nuclear weapons delivered by DOE to the Department of Defense. The plant also produces components used in the nuclear weapons development and testing programs carried out by the three DOE nuclear weapons design laboratories. The plant is located about 9-11 miles from the site of the proposed CRBRP and employs about 7300 persons. (App. Exh. 47, pp. 3, 9, Tr. 5423, 5429).

134. The Y-12 plant is vital to national security, (Hibbits, Tr. 5243), and the consequences of long term evacuation of Y-12 would be unacceptable in terms of national security risk. (Hibbits, Tr. 5193).

B. Staff and Applicants Failed to Consider the Impacts Upon Y-12 of Major Core Disruptive Accidents at the CRBR

135. Staff and Applicants calculate the radiological consequences at the Y-12 plant of only two types of CRBR accidents; a CRBR site suitability source term (SSST) accident and the most benign category of CRBR core disruptive accidents. Staff limited its CDA consequences analysis to a CDA Class 1 (Thadani, Tr. 5664), which results in the lowest releases and dose consequences of all the CDA Classes analyzed by Staff. Applicants limited their CDA analysis to an HCDA Case 2 core disruptive accident (Hibbits, Tr. 5234, App. Exh. 47, p. 5, Tr. 5425), which is comparable to Staff CDA Class 1 (Finding 14), and which does not result in the highest dose consequences of all the CDA Cases analyzed by Applicants for each of the organs of interest. (Strawbridge, Tr. 5186).

136. Staff failed to calculate the dose consequences to Y-12 personnel that would result from CRBR core disruptive accidents more severe than Staff's CDA Class 1 or Applicants HCDA Case 2, e.g. Staff's CDA Classes 2 through 4. Consequently, Staff failed to analyze whether long term evacuation of Y-12 personnel or other consequences would be likely following such accidents. (Soffer, Tr. 5668-69). Although Staff estimated the probability of occurrence of such accidents at the CRBR, it failed to analyze adequately the risk of such accidents, which involves equal consideration of both probability and consequences. (Soffer, Tr. 5668).

137. Applicants failed to calculate the dose consequences to Y-12 personnel that would result from more severe CRBR core disruptive accidents, for example, Staff's CDA Classes 2 through 4, or even Applicants' HCDA Cases 3 and 4 which are more benign than Staff CDA Classes 2 through 4, in that they assume that the containment/confinement systems operate as designed. (App. Exh. 1, p. 69, Tr. 2058; Strawbridge, Tr. 5072-73, Tr. 5188) even though Staff and Applicants included those accidents in their NEPA analysis. (Staff Exh. 8, FSFES, App. J; App. Exh. 46, pp. 33-34, Tr. 5409-10). Consequently, Applicants failed to analyze whether long term evacuation of Y-12 personnel or other consequences would be likely following such accidents.

138. Applicants' claim that the "radiological risk" from each of Staff's four CDA Classes is the same, and that therefore the consequences of only one type of CDA need be considered, (App. Exh. 46, pp. 38-39, Tr. 5414-15), is unfounded.

a. Applicants reach this conclusion by multiplying the probability of occurrence of each Staff CDA Class by the radiological releases to the environment from each Class, in order to calculate radiological risk. (App. Exh. 46, p. 38, Tr. 5414). This method ignores the fact that a CDA Class 2-4 accident may require long-term evacuation (Staff Exh. 18, pp. 8-9, Tr. 5690-91, Strawbridge, Hibbits, Tr. 5192-93; Hibbits, Tr. 5195), whereas Applicants concluded that a CDA Class 1 accident might not require long term evacuation.

(Hibbits, Tr. 5192-93, 5195). Given this fact, a threshold level apparently exists for the consequences of various CDA accidents, above which level the consequences are unacceptable in terms of national security risk. When multiplying probabilities and consequences in situations where a threshold level exists, it is appropriate to apply a risk aversion weighting factor (larger than 1) to events with consequences above the threshold level, in order to more accurately represent societal risk. (NUREG-0739, "An Approach to Quantitative Safety Goals for Nuclear Power Plants, Oct. 1980, Tr. 5195-96. Applicants have not applied such a risk aversion weighting factor (Strawbridge, Tr. 5190-96), and have thus failed to demonstrate that the national security risks of CDA Classes 1-4 are comparable or acceptable in all cases.

b. Applicants' risk calculation is also in error in that it multiplies the probability of occurrence of a CDA Class times the radiological releases to the environment from such a CDA, rather than the doses to Y-12 personnel from such a CDA. The Y-12 doses are not necessarily proportional to the releases, since the releases from Classes 3 and 4 occur immediately, whereas the Class 2 doses do not occur for 24 hours. (Staff Exh. 8, pp. J-7, J-11).

139. Based on the above facts, Staff and Applicants have failed adequately to analyze the impacts on Y-12 of CDAs with more serious consequences.

C. Staff and Applicants' Calculations of SSST and CDA Releases Are Based Upon Faulty Assumptions

140. Staff and Applicants' estimates of the releases to Y-12 from a site suitability source term (SSST) accident are faulty in that they do not include releases from the containment vent/purge system (Strawbridge, Tr. 5216-18; Thadani, Tr. 5664-65), which would be activated in the event of a core melt accident, the type of accident upon which the SSST is based. (Findings 81 to 87).

141. Staff and Applicants' estimates of the releases to Y-12 from SSST and CDA accidents are faulty in that they do not take into account the use in CRBR of plutonium recovered from LWR high burnup spent fuel, which has higher isotopic concentrations of Pu-238 and Pu-241 and, therefore, more serious dose consequences. (Findings 154 to 166).

142. Applicants' estimates of the releases to Y-12 from SSST and CDA Case 2 accidents are at least a factor of 14 less conservative than Staff's estimates, based upon more optimistic assumptions regarding filter efficiencies. (Staff Exh. 18, pp. 6, 8, Tr. 5688, 5690; Thadani, Tr. 5665-66; Finding 22).

D. Staff and Applicants Failed Adequately to Analyze the Radiological Consequences at Y-12 of SSST or Core Disruptive Accidents

143. Neither Staff nor Applicants considered the effects of wet deposition, i.e. rainfall, upon their calculations of SSST and CDA dose consequences at the Y-12 plant. (Hibbits, Tr. 5233-34; 5332; Thadani, Tr. 5656). The most appropriate treatment of wet

deposition is to perform two dose calculations; one assuming no rainfall, and the second assuming continuous rainfall. (Hibbits, Tr. 5332). The issue of wet deposition introduces substantial uncertainty in any calculations of dose consequences and ground contamination. (Hibbits, Tr. 5332).

144. Neither Staff nor Applicants adequately analyzed the extent, and effects upon Y-12 of plutonium ground contamination from CRBR accidents.

a. Staff did not consider the extent or effects of ground contamination at all in its examination of the impacts upon Y-12 of CRBR accidents.

b. Applicants failed to consider the extent of plutonium ground contamination beyond the first 7 days of an accidental release (Hibbits, Tr. 5210), or the extent and implications of total ground deposition levels at the Y-12 plant following an SSST or CDA release at CRBR. (Hibbits, Tr. 5232-33).

c. There is nothing in the record to indicate what ground contamination levels might trigger evacuation of Y-12 personnel or how long such evacuation might last, including whether the ground contamination levels calculated by Applicants (App. Exh. 47, pp. 14-15, Tr. 5434-35) would trigger such evacuation.

E. Staff and Applicants Failed Adequately to Consider the Likelihood of Long-Term Evacuation at Y-12 Following a CRBR Accident

145. In determining whether long-term evacuation at the Y-12

plant would likely follow an accident at CRBR, Scaff relied solely on whether the whole body and thyroid doses from a particular accident would exceed the Environmental Protection Agency's Protective Action Guidelines (PAGs). (Staff Exh. 18, pp. 7-8, Tr. 5689-90). Evacuation, however, would likely occur at dose levels much lower than those contained in the EPA PAGs. (Thadani, Tr. 5673-74; Hibbits, Tr. 5221, 5276-77; Soffer, Tr. 5660-61). In addition, there are no EPA PAGs for bone dose or bone surface dose (Hibbits, Tr. 5296-97; Thadani, Tr. 5663-34), even though bone surface dose may be controlling in terms of plutonium release. (Hibbits, Tr. 5297).

146. Based upon the above facts, relying solely upon the EPA Protective Action Guidelines to determine whether evacuation of Y-12 is likely to occur following a CRBR accident is improper.

147. Staff and Applicants also failed to consider the extent and implications for Y-12 evacuation of the tracking of ground contamination to the homes of Y-12 personnel by means of automobiles, clothes, and the like. (Hibbits, Tr. 5201-04).

148. Staff and Applicants therefore failed to consider adequately the likelihood of evacuation at Y-12 following CRBR accidents. Staff also failed to consider the impacts upon national security of long-term evacuation at Y-12. (Thadani, Tr. 5657, 5677; Lowenberg, Tr. 5693, Staff Exh. 18, p. 11, Tr. 5693).

Contentions 11(b) and 11(c)

11. The health and safety consequences to the public and plant employees which may occur if the CRBR merely complies with current NRC standards for radiation protection of the public health and safety have not been adequately analyzed by Applicants or Staff.
 - b) Neither Applicants nor Staff have adequately assessed the genetic effects from radiation exposure including genetic effects to the general population from plant employee exposure.
 - c) Neither Applicants nor Staff have adequately assessed the induction of cancer from the exposure of plant employees and the public.

I. STAFF AND APPLICANTS' ANALYSIS OF THE SOMATIC EFFECTS ASSOCIATED WITH CRBR ROUTINE RELEASES IS INADEQUATE

A. Staff's Cancer Risk Estimators

149. Staff analyzed the somatic cancer effects associated with CRBR routine releases by multiplying the estimated annual whole body dose to an individual ("standard man") by a somatic risk estimator of 135 potential fatal cancers per million person-rem, based in part on a 1972 National Academy of Sciences report entitled "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation (the "BEIR I" Report). (Staff Exh. 8, FSFES, pp. 5-14, 5-15, 5-21; Staff Exh. 13, pp. 4-7, Tr. 4147-50).

150. Expert opinion on the appropriate value for the cancer risk estimator, or cancer risk coefficient, differ, in some cases markedly, from the 135 fatal cancers/ 10^6 person rem value assumed by Staff. A number of experts, including Edward Radford, Karl Morgan, John Gofman, Alice Stewart, Thomas Mancuso, George Kneale, and Arthur Tamplin, believe that the Staff cancer risk

estimator is low, or probably low. Their own estimates vary, but range from a factor of 3 (Radford) to a factor of 7 (Morgan) to a factor of 28 times greater than Staff's estimate of 135 fatal cancers/ 10^6 person-rem due to low-LET whole body exposure. (Cochran, Int. Exh. 22, pp. 35-36, Tr. 6229-30).

151. Staff failed adequately to display and consider the substantial uncertainties surrounding its cancer risk estimator. This would be best accomplished by presenting a range of values for such estimators rather than a single point estimator (McClellan, Tr. 4031-32; Bender, Tr. 4083-84), and by representing in such range of values the views of other experts in the field. (McClellan, Tr. 4022-25).

152. Staff's use of the BEIR I report "relative risk" model to indicate the "reasonable" upper bound of uncertainty in its cancer risk estimator (Staff, Exh. 8, FSFES, p. 5-15) is insufficient. The findings of the 1980 National Research Council's Report of the Biological Effects of Ionizing Radiation (BEIR III) Committee, and by implication, earlier findings of the BEIR I committee as well, are still subject to uncertainty and evolution, based on new information and analysis. (Bender, Tr. 4076, 4092, 4118; McClellan, Tr. 4022-25; Thompson, Tr. 4025-31). Staff should therefore not limit its discussion of uncertainties to values within the BEIR I or BEIR III reports, but should consider as well the range of expert opinion contained outside those reports (Cochran, Int. Exh. 22, pp. 35-36, Tr. 6229-30), as well as more recent data. (Thompson and McClellan, Tr. 4025-31).

B. Applicants' Cancer Risk Estimators

153. Applicants employed a range of cancer risk estimators in their calculation of the somatic risks associated with routine CRBR radiological releases. These estimators were based on the "absolute risk" and "relative risk" models of the BEIR III Committee and the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR). (App. Exh. 42, p. 27, Tr. 4293). Although use of a range of values for cancer risk estimators is appropriate, Applicants fail to consider the range of expert opinion outside those reports (Finding 150) as a more complete representation of the uncertainty in those cancer risk estimators.

Contentions 6 The ER and FES do not include an adequate analysis of the environmental impact of the fuel cycle associated with the CRBR for the following reasons:

* * * *

- b) The analysis of fuel cycle impacts in the ER and FES are inadequate since:
 - 1) The impact of reprocessing of spent fuel and plutonium separation required for the CRBR is inadequately assessed;
* * * *
 - 3) The impact of disposal of wastes from the CRBR spent fuel is inadequately assessed[.]

I. STAFF AND APPLICANTS' ANALYSIS IS BASED
UPON INSUFFICIENTLY CONSERVATIVE ASSUMPTIONS FOR
PLUTONIUM FUEL ISOTOPIC CONTENT

A. Introduction

154. The origin of the plutonium used to fuel CRBR and the manner in which it has been, and is being, recycled determines the isotopic concentrations of the plutonium isotopes that are released to the environment from the CRBR and its fuel cycle under normal and accident conditions. (Cochran, Int. Exh. 13, pp. 7-8, Tr. 4573-74). These plutonium isotopic concentrations, in turn, determine to a large extent the somatic risks, genetic risks, and other environmental effects associated with plutonium releases from the CRBR and its fuel cycle. (Cochran, Int. Exh. 13, pp. 8, 19-20, Tr. 4574, 4585-87).

155. The isotopes Pu-238 and Pu-241 are controlling in terms of bone surface dose (Cochran, Int. Exh. 13, p. 21, Tr. 4587), although, for ease of reference, one can generally assume that the higher the percentage of Pu-240, the higher the percentage of controlling isotopes Pu-238 and Pu-241 (Cochran, Tr. 4530). In

light water reactors (LWRs), the burnup of a particular type of fuel (i.e., the amount of time spent inside a reactor prior to removal and reprocessing) can be used as a starting point for estimating plutonium isotopic content, since, in light water reactors, the higher the burnup of a fuel, the higher the percentage of Pu-238 and Pu-241. (Clark, Tr. 4378-79; Sherwood, Tr. 4263-64). For fast reactors, such as CRBR, the percentage of Pu-238 and Pu-241 slowly decreases over a 13 cycle (13 year) period according to Applicants' analysis. (App. Exh. 36, PSAR, p. 14.4A.5; Cochran, Tr. 4539-50; Sherwood, Tr. 4311-12).

156. Applicants performed two estimates of CRBR fuel cycle impacts; one using so-called "LWR-grade" plutonium (20% Pu-240; 0.5% Pu-238; 6% Pu-241; 1.5% Pu-242) in a once-through fuel cycle in the CRBR (App. Exh. 35, ER, Section 5.7) and one using so-called "FFTF-grade" plutonium (12% Pu-240) which was recycled repeatedly in the CRBR without being commingled with LWR spent fuel. (App. Exh. 36, ER, p. 14.4A-2). Applicants' witness believed that the 20% Pu-240 plutonium used in Applicants' Exh. 35 had a burnup in the range of 25,000 megawatt days per metric ton (Mwd/MT). (Sherwood, Tr. 4259-61).

157. Staff, in its Site Suitability Source Term (SSST) analysis (Staff Exh. 1) assumed a fuel plutonium isotopic concentration of 20% Pu-240, 1% Pu-238, and 5% Pu-241. (Cochran, Int. Exh. 13, p. 20, Tr. 4586). For purposes of its analysis of CDA risks and consequences (Staff Exh. 8, FSFES, App. J), Staff assumed similar plutonium isotopic concentrations. (Cochran, Int. Exh. 13, p.

21, Tr. 4587). In its estimates of routine releases and effluents associated with CRBR fuel reprocessing (Staff Exh. 8, FSFES, App. D), Staff assumed a plutonium composition of roughly 18% Pu-240, 1% Pu-238, and 6% Pu-241 (Cochran, Int. Exh. 13, p. 22-23, Tr. 4588-89). Staff assumed the latter isotopic concentrations are associated with a burnup of 20,000-30,000 megawatt days per metric ton but did not know for sure. (Clark, Tr. 4370, Lowenberg, Tr. 4383-84).

158. Both Staff and Applicants' analyses assumes that the CRBR will be fueled by low burnup LWR spent fuel, in either a once-through fuel cycle or recycle in the CRBR without commingling with other LWR spent fuel. (App. Exhs. 35 and 36; Lowenberg, Tr. 4360). Both Staff and Applicants fail to analyze the environmental impact of the reasonably foreseeable use in CRBR of

- (a) plutonium obtained directly from high burnup LWR spent fuel (e.g., from Barnwell or the Developmental Reprocessing Plant (DRP));
- (b) plutonium obtained from LWR high burnup spent fuel after this plutonium has been recycled several times in LWRs, prior to use in CRBR; or
- (c) plutonium obtained from CRBR recycled fuel commingled with either (a) or (b) above.

(Cochran, Int. Exh. 13, pp. 17-18, Tr. 4583-84; Lowenberg, 4363).

B. The Use in CRBR of Plutonium from LWR High Burnup Spent Fuel at Some Point in the Operating Lifetime of the Plant is Reasonably Foreseeable.

159. There is insufficient evidence in the record to establish that there are sufficient amounts of plutonium with isotopic concentrations of 12-20% Pu-240 to fuel the CRBR during its entire operating lifetime. Staff and Applicants, assuming that 12-20% Pu-240 fuel is associated with burnups of 20,000-30,000 Mwd/MT, claim that enough fuel at burnups less than approximately 25,000 Mwd_{MT} is available in the DOE stockpile and LWR spent fuel to fuel the CRBR for at least 15 years. (Yarbro, Tr. 4260; Hartman, Tr. 4313-14). But as demonstrated by Intervenor Exh. 14 and associated testimony, the Pu isotopic concentrations assumed by Staff and Applicants in their SSST and fuel cycle analyses (i.e., 20% Pu-240, 1% Pu-238, 5-6% Pu-241) is more reasonably associated with a burnup of 12,000-14,000 Mwd/MT. (Cochran, Int. Exh. 14, Tr. 4617; Cochran, Tr. 4531-33, 4561, 62-4617). There is insufficient evidence in the record to establish that there would be sufficient LWR fuel with burnups of less than 12,000-14,000 Mwd/MT (the fuel type assumed by Staff and Applicants) to fuel the CRBR for "at least 15 years," much less for its entire projected 30 year operating lifetime.

160. Staff claims it is "realistic" to assume that CRBR will employ a once-through, or open, fuel cycle during the early years of CRBRP operations, followed by a closed fuel cycle utilizing repeated recycle of plutonium during later CRBRP operations (Staff Exh. 8, FSFES, App. D, p. D-35). Staff and Applicants'

claim that there will be sufficient low burnup LWR fuel available to fuel the CRBR is necessarily based on the assumption that the CRBR fuel cycle will be closed prior to running out of available low burnup LWR spent fuel, since Applicants' (and Staff's) belief that there are sufficient stocks of low-burnup LWR spent fuel for "at least 15 years" (Finding 159) will not fulfill the CRBR fuel requirements for its full projected 30-year operating life if operated on an open fuel cycle.

161. The record is insufficient to support a conclusion that the CRBR fuel cycle will be closed prior to expending the available low burnup LWR fuel.

a. The Developmental Reprocessing Plant (DRP), or a presently existing but modified reprocessing facility, which would be a necessary part of a closed CRBR fuel cycle, might not be available until 1996 (Yarbro, Tr. 4235-36, 4241; Clark, Tr. 4353-54), whereas CRBR might come on line as early as 1989. (Yarbro, Tr. 4236), a difference of 7 years.

b. Since the so-called "LWR-grade" plutonium used in Staff and Applicants' analysis actually corresponds to 12,000-14,000 Mwd/MT and not 25,000 Mwd/MT (Finding 159), the record does not support a conclusion that there is sufficient plutonium from spent fuel with burnup of less than that level to fuel the CRBR for the seven years or so prior to the operation of the DRP.

162. It is reasonably foreseeable that CRBR will utilize plutonium recovered from high burnup LWR spent fuel that has been

reprocessed in the DRP (or elsewhere), regardless of the availability of low-burnup LWR spent fuel.

a. The DRP capacity, according to its conceptual design, is 150 metric tons per year, only 11 metric tons (or 8%) of which would consist of CRBR spent fuel. Approximately 137 metric tons per year (92%) of DRP capacity could be used to process light water reactor fuel. (Staff Exh. 8, FSFES, App. D, p. D-12). The DRP could handle any of the fuels expected to be discharged from LWRs, with no constraints for processing LWR fuel with high burnup. (Yarbro, Tr. 4305, 4308). There is no evidence that this reprocessed LWR fuel is earmarked for use anywhere other than in the CRBR. (Yarbro, Tr. 4308-09; Sherwood, Tr. 4253-54).

b. The LWR spent fuel reprocessed in the DRP could be of high burnup. Some light water reactors have achieved a burnup of 33,000 megawatt days per metric ton (Sherwood, Tr. 4262), and light water reactors are moving to higher burnups at this time. (Sherwood, Tr. 4263-64). Applicants could not estimate how high the burnups of LWR fuels might reach during the lifetime of the CRBR. (Sherwood, Tr. 4264). There is no evidence in the record to indicate that if LWR spent fuel were reprocessed it would not be recycled in LWRs, or used in CRBR after recycling in LWRs. (Sherwood, Tr. 4253).

c. There is no assurance that the Tennessee Valley Authority (TVA), which has the option of buying and managing

the CRBR following its 5-year demonstration period (Staff Exh. 8, p. iii), would utilize only plutonium from low burnup spent fuel, equal to or less than 12,000-14,000 Mwd/MT, even if it were available. There is nothing in the record to indicate, and it is unreasonable to assume, that TVA would purchase low burnup LWR spent fuel from other utilities rather than utilize high burnup spent fuel from other light water reactors in the TVA system, or commingle CRBR fuel with other TVA LWR spent fuel. (Sherwood, Tr. 4310; Yarboro, Tr. 4311).

163. For a reactor of the general size and type as the Clinch River Breeder Reactor, Staff and Applicants should assume that it will be fueled at some point in its operating lifetime by plutonium recovered from LWR high burnup spent fuel, and should analyze the environmental impacts of such fuel use. In particular, Staff must analyze whether the site is suitable under 10 CFR Part 100 for a reactor of the general size and type as CRBR, if such LWR high burnup spent fuel is used.

C. Effect on CRBR Analysis of Assuming Use of LWR High Burnup Spent Fuel.

164. Assuming that CRBR will be fueled by plutonium recovered from LWR high burnup spent fuel would increase the plutonium bone surface dose contribution in Appendix D by a factor of 2 to 4.3 over the respective dose estimates of Staff and Applicants. (Cochran, Int. Exh. 13, pp. 20-25, Tr. 4586-91). Applicants

conceded that assuming use of LWR high burnup spent fuel would increase the plutonium source term in the two controlling isotopes, Pu-238 and Pu-241, by a factor of 2 to 4. (Yarbro, Tr. 4265).

165. Correcting for this factor would also increase Staff's site suitability source term bone surface (and bone) dose estimates in Staff Exh. 1, 1982 SSR, p. III-11 by a factor of 4.3 (Findings 74, 157), and would increase Staff's core disruptive accident bone surface dose estimates in Staff Exh. 8, FSFES, App. J by the same factor. (Finding 157).

166. This increase in bone surface dose contribution would occur regardless of how the plutonium isotopic concentration would change after the fuel is placed in the CRBR. Even if the concentrations of Pu-238 and Pu-241 would slowly decrease as the fuel is burned in the CRBR over a 13 year period (Finding 155), the initial concentrations would be much higher than assumed by Staff and Applicants. (Sherwood, Tr. 4311-12; Cochran, Int. Exh. 13, p. 25, Tr. 4591).

II. STAFF AND APPLICANTS HAVE NOT ANALYZED
ALL REASONABLY FORESEEABLE ALTERNATIVES
TO THE DEVELOPMENTAL REPROCESSING PLANT (DRP)

A. Modification and Use of the Savannah River Plant to
Reprocess CRBR Spent Fuel is a Reasonably Foreseeable
Alternative to the DRP.

167. Applicants have selected, and Staff has analyzed, the Developmental Reprocessing Plant (DRP), which is still in the conceptual design stage, as a basis for evaluating the

environmental impacts of the reprocessing step of the CRBR fuel cycle. (Staff Exh. 8, FSFES, App. D, p. D-12). One reasonably foreseeable option to the DRP is the modification, by the addition of a "breeder head-end" facility, of the existing DOE reprocessing facility at the Savannah River Plant (SRP). (Staff Exh. 8, FSFES, App. D, pp. D-15 to D-17; App. Exh. 35, ER, p. 5.7-7; Yarbrow, Sherwood, Tr. 4194-95, 4204-12).

B. The Estimated Environmental and Radiological Impacts from the DRP Do Not Bound the Anticipated Impacts From Reprocessing CRBR Fuel at the Savannah River Plant.

168. Staff estimated that most of the U.S. radiological impacts from the CRBR fuel cycle (170 person-rem annual U.S. population whole body dose commitment) (Staff Exh. 8, FSFES, p. 5-20, App. D, p. D-34), would result from spent fuel reprocessing activities, (Staff Exh. 8, FSFES, App. D, p. D-34); that over 99% of that U.S. population whole body dose commitment would be due to releases of carbon-14 and tritium (Staff Exh. 14, p. 22, Tr. 4465), which would be released from the head-end facility if an alternative reprocessing plant were chosen (Lowenberg, Tr. 4405-06); and that only 1% of the annual U.S. population whole body dose commitment from reprocessing facilities would be due to plutonium and other transuranics. (Staff Exh. 14, p. 22, Tr. 4465). These estimates are neither applicable to nor bounding of the radiological impacts from reprocessing CRBR fuel at the Savannah River Plant.

169. The amount of plutonium and other transuranics that would be released into the atmosphere at the SRP, and the corresponding dose impacts, are much greater than estimated for the DRP.

a. The recent plutonium gaseous releases from the SRP are approximately a factor of 10 higher than those estimated for the DRP. The lifetime plutonium gaseous releases from the SRP, which may include both accidental and routine releases, are approximately 4000 times higher than those estimated for the DRP. (Cochran, Int. Exh. 13, pp. 31-33, Tr. 4597-99; see also Sherwood, Tr. 4220-21; Lowenberg, Tr. 4397-98, 4490-10).

b. Although the DRP is assumed to release no liquid effluents, the Savannah River Plant does release liquid radioactive effluents to the environment. (Staff Exh. 8, FSFES, App. D, p. D-17). Staff has not analyzed what the SRP liquid effluents would be from reprocessing CRBR fuel. (Branagan, Tr. 4411-12).

170. There is insufficient evidence in the record to establish that if the Department of Energy (DOE) selected the Savannah River plant to reprocess CRBR fuel, that DOE would reduce the current levels of plutonium and transuranic releases to meet those assumed for the DRP.

a. Construction of a "breeder head-end" facility at the SRP would not reduce transuranic releases, since most of the transuranics would be released from the balance of the SRP rather than the head-end facility. (Lowenberg, Tr. 4409-10).

b. Addition of another bank of HEPA filters (Lowenberg, Tr. 4430-31) would not reduce the transuranic releases that are due to accidents or bypass leakage around the entire filter system. (Clark, Tr. 4436). Neither Staff nor Applicants have determined what percentage of the SRP transuranic releases are due to accidental or bypass leakage (Clark, Tr. 4436), nor have they analyzed the risks and consequences of accidental releases at the DRP (Cochran, Int. Exh. 13, pp. 33-34, Tr. 4599-4600), which is an important contributor to radiological releases from plutonium handling and reprocessing facilities. (Johnson, Int. Exh. 21, pp. 3-4, 6-9, Tr. 6020-21, 6023-26; Cobb, Int. Exh. 9, pp. 4-5, 9, Tr. 3104-05, 3109).

c. Addition of another bank of HEPA filters, which are used to reduce atmospheric releases (Lowenberg, Tr. 4430), would not reduce the amount of transuranics released in liquid effluents. There is no evidence that Applicants would commit to releasing zero liquid effluents at SRP if the plant were also used to reprocess CRBR fuel. (Clark Tr. 4429; Staff Exh. 8, FSFES, App. D, p. D-17).

d. There is insufficient evidence that DOE would make any changes in the SRP if it were chosen as the CRBR reprocessing alternative, other than construction of the head-end facility, which is a necessary component for reprocessing breeder fuel. The Savannah River Plant is not required to comply with any Commission regulations, and there is no

evidence that DOE regulatory requirements for SRP are co-extensive or equivalent to those of the Commission. (Yarbro, Sherwood, Tr. 4247-48). DOE has not upheld its commitments in the past; for example, the commitment to bottle rather than release any noble gases from the CRBR plant itself. (Yarbro, Tr. 4179; Sherwood, Tr. 4181-82), and DOE has not always been completely candid in measuring and reporting radioactive releases from its facilities. (Johnson, Tr. 6022-23). Staff's conclusions regarding SRP releases should be based on its independent analysis of experience to date at the SRP and similar facilities, rather than relying on DOE commitments which may not materialize. (Cochran, Int. Exh. 13, pp. 29-34, Tr. 4595-4600).

171. The health risk from plutonium releases at the DRP, and, consequently, from alternative reprocessing facilities, is greater than that assumed by Staff and Applicants. Staff's claim that only one percent of the population dose commitment from the DRP is due to plutonium and transuranic releases refers only to U.S. population whole body dose commitment. (Staff Exh. 8, FSFES, App. D, Table D.17, p. D-34; Staff Exh. 14, p. 22, Tr. 4465; Johnson, Tr. 5930-33). Staff has not reported the population bone surface dose commitment that would result from reprocessing activities at any facility, even though bone surface dose is controlling for plutonium releases. (Johnson, Int. Exh. 21, p. 7, Tr. 6024; Hibbits, Tr. 5297). The health risk from plutonium

reprocessing releases could be significantly higher if one included the internal organ (including bone surface) dose contributions rather than reporting only whole body dose contributions. (Cochran, Tr. 4594-4600).

172. Although Staff estimates that the contribution to the population bone surface dose commitment due to plutonium is on the order of 0.5% (4 + 875 person-rem (Cochran, Tr. 4594), the combination of potential errors introduced by underestimating the plutonium isotopic concentrations of Pu-238 and Pu-241 (Findings 154 to 166), the plutonium confinement factors at reprocessing plants (Finding 169), and understating the quality factor for bone surface (and lung) dose calculations (Finding 104), could lead to a substantial underestimate, by several orders of magnitude, of the health effects due to plutonium releases. (Cochran, Inc. Exh. 13, p. 34, Tr. 4600).

C. Staff Has Not Analyzed Adequately the Environmental Impacts of Reprocessing CRBR Fuel at the Savannah River Plant.

173. Other than concluding that its analyses of DRP environmental and radiological releases envelope the impacts from any alternative reprocessing facility, Staff has not analyzed the environmental and radiological impacts of reprocessing CRBR fuel at the Savannah River Plant. (Lowenberg, Tr. 4389, 4402; Cochran, Int. Exh. 13, p. 7, Tr. 4573). Specifically, Staff did not analyze liquid effluents (Branagan, Tr. 4411-12); transuranic releases (Lowenberg, Tr. 4397-98, 4409-10); accidental or bypass leakage (Clark, Tr. 4436; Cochran, Int. Exh. 13, pp. 5, 33-34,

Tr. 4591, 4599-4600); or confinement factors (Clark, Tr. 4395-96) at the SRP or other alternative reprocessing facilities, such as the Purex facility at Hanford.

III. STAFF AND APPLICANTS HAVE NOT ADEQUATELY ANALYZED
THE ENVIRONMENTAL AND RADIOLOGICAL IMPACTS ASSOCIATED WITH
CRER WASTE MANAGEMENT ACTIVITIES.

174. Staff concluded that the effluents from the CRER high level waste (HLW) stored in a geologic repository would be zero or negligible, and that the only non-zero radiological effluents are the releases of radon and its decay products associated with construction of the repository, e.g., mining the repository cavity. (Staff Exh. 8, FSFES, App. D, p. D-23). Staff also concluded that these releases, associated with mining the cavity, are negligible by comparison with similar effects from other fuel cycle steps. (Staff Exh. 8, FSFES, App. D, pp. D-9, D-23; Cochran, Int. Exh. 13, p. 35, Tr. 4601).

175. Staff's conclusions do not reasonably reflect the uncertainties associated with HLW disposal. The Draft EPA Proposed Environmental Standards and Federal Radiation Protection Guidance for Management and Disposal of High-Level and Transuranic Radioactive Wastes (EPA, Working Draft, 6/14/82) (recently issued for public comment as a Proposed Rule, Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes, 47 Fed. Reg. 58196 (Dec. 29, 1982)), establishes limits on radioactivity released to the "accessible environment" which are

designed to limit long-term risks to 1000 health effects over 10,000 years for a 100,000 MTHM Repository. (Cochran, Int. Exh. 13, p. 36, Tr. 4602). Staff assumes that CRBR high-level waste over a 30-year operating period will represent on the order of, or less than, 1/100 of the total repository volume. (Staff Exh. 8, FSFES, App. D, p. D-20). Thus, under proposed EPA standards, the CRBR contribution to the total health effects commitment in the accessible environment during the first 10,000 years after closure is meant to be limited to $1000/100=10$ health effects, or approximately 0.3 health effects for each year of CRBR operation. This level of risk is an order of magnitude greater than other fuel cycle risks as estimated by the Staff, i.e., 0.023 potential cancers/year. (Staff Exh. 8, FSFES, p. 5-21).

176. Because the health effects associated with CRBR waste management activities are not negligible, Staff has failed adequately to analyze these effects, including the uncertainties associated with its estimates of health effects.

IV. STAFF HAS FAILED TO INCLUDE
THE IMPACTS OF OTHER FUEL CYCLE ACTIVITIES
IN ITS ENVIRONMENTAL COST/BENEFIT ANALYSIS

177. Staff has failed to analyze, and to include in its NEPA cost/benefit analysis, the environmental impacts of obtaining plutonium for CRBR by reprocessing LWR spent fuel. (App. Exh. 43, pp. 16-17, Tr. 4339-40). It is possible that LWR spent fuel reprocessing would occur only to supply plutonium for CRBR (Sherwood, Tr. 4232), but Staff has not included as an

environmental cost, in its cost/benefit analysis, the portion of the impacts of LWR reprocessing facilities or waste management facilities that might be attributable to obtaining CRBR plutonium. (Sherwood, Tr. 4238).

178. Staff has failed to analyze and weigh the environmental impacts of interim storage of CRBR high level wastes or transuranic wastes from the Secure Automated Facility (SAF) line or the DRP. Away-from-reactor (AFR) interim storage of wastes from these facilities might reasonably be required (Yarbro, Tr. 4235-37), but was not discussed or included as an environmental cost.

Contentions 5(a) and 7(c)

5. Neither Applicants nor Staff have established that the site selected for the CRBR provides adequate protection for public health and safety, the environment, national security, and national energy supplies; and an alternative site would be preferable for the following reasons:
 - a) The site meteorology and population density are less favorable than most sites used for LWRs.
 - (1) The wind speed and inversion conditions at the Clinch River site are less favorable than most sites used for light-water reactors.
 - (2) The population density of the CRBR site is less favorable than that of several alternative sites.
 - (3) Alternative sites with more favorable meteorology and population characteristics have not been adequately identified and analyzed by Applicants and Staff. The analysis of alternative sites in the ER and the Staff Site Suitability Report gave insufficient weight to the meteorological and population disadvantages of the Clinch River site and did not attempt to identify a site or sites with more favorable characteristics.
7. Neither Applicants nor Staff have adequately analyzed the alternatives to the CRBR for the following reasons:
 - c) Alternative sites with more favorable environmental and safety features were not analyzed adequately and insufficient weight was given to environmental and safety values in site selection.
 - (1) Alternatives which were inadequately analyzed include Hanford Reservation, Idaho Reservation (INEL), Nevada Test Site, the TVA Hartsville and Yellow Creek sites, co-location with an LMFBR fuel reprocessing plant (e.g., the Development Reprocessing Plant), an LMFBR fuel fabricating plant, and underground sites.

I. THE HARTSVILLE, YELLOW CREEK, HANFORD, INEL, AND SAVANNAH RIVER SITES ARE SUBSTANTIALLY BETTER FOR THE LMFBR DEMONSTRATION PROJECT THAN THE CLINCH RIVER SITE

A. The Hartsville Site

179. The meteorological accident X/Q values for the Hartsville site average about a factor of two lower than those for Clinch River, and thus the accident diffusion conditions are preferable. (Staff Exh. 15, p. 14, Tr. 4878; Spickler, Tr. 4803-04). The 0-30 mile population density projection for the year 1990 is about a factor of three lower at Hartsville than at Clinch River. (Staff Exh. 15, pp. 20,22, Tr. 4884, 4886; Ferrel, Tr. 4806-07). Considering these two factors together as a surrogate for radiological risk to the public (Spickler, Tr. 4801), the radiological risk at the Hartsville site would be about a factor of 6 lower than at the Clinch River site.

180. The Hartsville water availability, overall aquatic ecology and terrestrial ecology are all environmentally preferable to the Clinch River site. (Staff Exh. 8, pp. L-5, L-8 - L-10). The Hartsville geology and seismology; overall hydrology; water quality; proximity to industrial, military, and transportation facilities; and utility participation impacts are all comparable to those at the Clinch River site. (Staff Exh. 8, pp. L-5, pp. L-5 - L-12). Socioeconomic factors would be less preferable at the Hartsville site. (Staff Exh. 8, p. L-11). The cost of moving to the Hartsville site would be about 1-2% of total cost, and possibly less if existing facilities and site preparation work were utilized. (Staff Exh. 8, pp. 9-12 - 9-14; Kripps, Tr. 4709-11).

B. The Yellow Creek Site

181. The meteorological accident X/Q values for the Yellow Creek site range from slightly worse than the Clinch River site for the 0-2 hour dose at the exclusion area boundary (EAB) to about a factor of six better than the Clinch River site for the 4-30 day dose at the low population zone (LPZ) boundary. (Staff Exh. 15, p. 14, Tr. 4878; Spickler, Tr. 4804). The meteorological diffusion conditions are slightly preferable than at the Clinch River site. (Staff Exh. 8, p. L-6). The 0-30 mile population density projection for the year 1990 is about a factor of four lower at Yellow Creek than at the Clinch River site. (Staff Exh. 15, pp. 20, 22, Tr. 4884, 4886; Ferrell, Tr. 4806-07). Considering these two factors together as a surrogate for radiological risk to the public (Spickler, Tr. 4801) the radiological risk at the Yellow Creek site would vary from about a factor of 3 better than Clinch River for the 0-2 hour dose at the EAB, to a factor of about 24 better than the Clinch River site for the 4-30 day dose at the LPZ boundary.

182. The Yellow Creek overall hydrology, water availability, drinking water, groundwater, water quality, and terrestrial ecology impacts are all preferable to those at the Clinch River site. (Staff Exh. 8, pp. L-26 - L-29). Aquatic construction effects would be preferable at the Yellow Creek site if the project utilized the existing barge unloading facility, water intake, and other site preparation work. (Staff Exh. 8,

p. L-28). Overall aquatic environmental impacts would be preferable at the Yellow Creek site since construction of the proposed LWR units at that site has been cancelled. (Staff Exh. 8, pp. L-28 - L-29). Yellow Creek geology and seismology; flooding; proximity to industrial, military, and transportation facilities; and utility participation factors are all comparable to those at the Clinch River site. (Staff Exh. 8, pp. L-26 - L-27, L-31 - L-32). Socioeconomic factors would be less preferable at the Yellow Creek site. (Staff Exh. 8, p. L-30). The cost of moving to the Yellow Creek site would only be 1-2% of the total cost, and possibly less if existing facilities and site preparation work were utilized. (Staff Exh. 8, pp. 9-12 - 9-14; Kripps, Tr. 4709-11).

C. The Hanford Site

183. The meteorological accident X/Q values for the Hanford site average about a factor of 3 lower than those for the Clinch River site, and thus the accident diffusion conditions are preferable. (Staff Exh. 15, pp. 14-15, Tr. 4878-79). The Hanford 0-30 mile population density projection for the year 1990 is 66 persons per square mile, about a factor of three lower than that for the Clinch River site. (Staff Exh. 15, pp. 20, 22, Tr. 4884, 4886). Considering these two factors together as a surrogate for radiological risk to the public (Spickler, Tr. 4801), the radiological risk at the Hanford site would be about a factor of 8-9 lower than at the Clinch River site.

184. The Hanford water availability, drinking water, overall hydrology, water quality, and aquatic construction impacts are all environmentally preferable to those at the Clinch River site. (Staff Exh. 8, pp. L-33 - L-35). The aquatic impacts of plant operation are also preferable at the Hanford site, since there are no potential impacts on striped bass or other aquatic biota at Hanford, as there are at Clinch River. (Staff Exh. 8, pp. L-34 - L-35). Hanford groundwater; flooding; terrestrial ecology; and proximity to industrial, military and transportation facilities impacts are comparable to those at Clinch River. (Staff Exh. 8, pp. L-33 - L-38). Geology, seismology, and socioeconomic impacts are less favorable than at Clinch River. (Staff Exh. 8, pp. L-33, L-37). The cost of relocating at the Hanford site is within the range of relocation costs for moving to another TVA site, taking into account increased revenue from the sale of power at Hanford. (Staff Exh. 8, pp. 9-12 - 9-13). There is insufficient evidence to determine the availability of utility participation at the Hanford site.

D. The Savannah River Site

185. The meteorological accident X/Q values for the Savannah River site average about a factor of 4 lower than those for the Clinch River site, and the accident diffusion conditions are significantly preferable. (Staff Exh. 15, pp. 14-15, Tr. 4878-79). The 0-30 mile population density projection for the year 1990 is 93 persons per square mile, more than a factor of 2 lower

than for the Clinch River site. (Staff Exh. 15, pp. 20, 22, Tr. 4884, 4886). Considering these two factors together as a surrogate for radiological risk to the public (Spickler, Tr. 4801), the radiological risk at the Savannah River site would be about a factor of 8-11 lower than at the Clinch River site.

186. Savannah River water availability, drinking water, groundwater, overall hydrology, water quality impacts, and aquatic impacts from plant construction, are all environmentally preferable to the Clinch River site. (Staff Exh. 8, pp. L-44 - L-46). Savannah River aquatic construction impacts, terrestrial ecology, flooding potential, socioeconomics, and proximity to industrial, military, and transportation facilities are all comparable to the Clinch River site. (Staff Exh. 8, pp. L-44, L-46 - L-49). Geology and seismology would be less preferable at the Savannah River site. (Staff Exh. 8, p. L-43). The cost of relocating at the Savannah River site is within the range of relocation costs for moving to another TVA site, taking into account the increased revenue from the sale of power at Savannah River. (Staff Exh. 8, pp. 9-12 - 9-13). There is insufficient evidence to determine the availability of utility participation at the Savannah River site.

E. The Idaho National Engineering Laboratory (INEL) Site

187. The meteorological accident X/Q values for the Idaho National Engineering Laboratory (INEL) site average about a factor of 3 lower than those at the Clinch River site, and the

accident diffusion conditions are preferable. (Staff Exh. 15, pp. 14-15, Tr. 4878-79). The INEL 0-30 mile population density projection for the year 1990 is 36 persons per square mile, more than a factor of 5 lower than at the Clinch River site. (Staff Exh. 15, pp. 20, 22, Tr. 4884, 4886). Considering these two factors together as a surrogate for radiological risk to the public (Spickler, Tr. 4801), the radiological risk at INEL would be a factor of 15-18 lower than at the Clinch River site.

188. INEL aquatic impacts from plant construction and operation, and terrestrial ecology, are environmentally preferable to the Clinch River site. (Staff Exh. 8, pp. L-40 - L-41). INEL geology; water availability; flooding potential; and proximity to industrial, military, and transportation facilities impacts are environmentally comparable to the Clinch River site. (Staff Exh. 8, pp. L-39, L-43). Seismology, hydrology, water quality, and socioeconomics are less desirable at INEL. (Staff Exh. 8, pp. L-39 - L-42). The cost of relocating at the INEL site is within the range of relocation costs for moving to another TVA site. (Staff Exh. 8, pp. 9-12 - 9-13). There is insufficient evidence to determine the availability of utility participation at the INEL site.

II. STAFF AND APPLICANTS HAVE GIVEN
INSUFFICIENT WEIGHT TO RELATIVE RADIOLOGICAL RISK
AND POPULATION DENSITY IN COMPARING ALTERNATIVE SITES

A. Radiological Risk

189. In the 1977 Final Environmental Statement related to construction and operation of Clinch River Breeder Reactor Plant (Staff Exh. 7), Staff considered the radiological risk of accidents at the Clinch River and alternative sites, as a function of population density and meteorological characteristics (Staff Exh. 7, pp. 9-11, 9-22). Staff concluded that the Hanford, Savannah River, and INEL sites

are better than the proposed site or any of the other alternative sites because the isolation provided would result in lower radiation dose in the event of an accidental release of radioactivity, in terms of both the nearest receptor and the total number of people exposed.

(Staff Exh. 7, p. 9-22). Staff found that the accidental radiological doses at those three DOE sites would be roughly 50 times lower than at the Clinch River site.

190. In the 1982 Supplemental Impact Statement (Staff Exh. 8,) Staff did not consider population and meteorological considerations jointly in its alternative siting analysis, although considering population and meteorology together is a more appropriate, and less crude, surrogate for radiological risk than considering population density alone, as Staff has done. (Soffer, Tr. 4795, 4798-99; Spickler, Tr. 4801).

191. In Staff Exhibit 8, Staff concluded that the Hanford, INEL, and Savannah River sites were no longer environmentally preferable, even though the population, meteorology, and relative risk parameters had not changed appreciably. (Staff Exh. 8, p. 9-15; Soffer, Tr. 4787-88, 4793-94; Spickler, Tr. 4790-93).

192. Staff's changed conclusion regarding alternative site preferability was due primarily to a reassessment of the weight to be given relative radiological risk. Staff now concludes that, since the radiological risk at the Clinch River site is acceptably low, any decrease in radiological risk at alternative sites must be considered inconsequential. (Staff Exh. 15, pp. 23, Tr. 4887; Soffer, Tr. 4788, 4794, 4818-19). As a result, Staff now believes that any reduction in radiological risk from LMFBR accidents that would occur as a result of moving to an alternative site must be considered insignificant, even if an alternative site had no resident population within 10 miles of the site. (Leech, Tr. 4808-09; Soffer, Tr. 4818-19; Staff Exh. 15, p. 23, Tr. 4887).

193. Applicants based their conclusion that the Hanford, INEL, and Savannah River sites are not environmentally preferable to the Clinch River site solely on the fact that they considered a factor of 50 difference in radiological risk consequences insignificant in terms of expected environmental impact. (Kripps, Tr. 4697-98). As a result, Applicants were unable to state whether they would consider a factor of 500 difference in radiological risk significant, if both sites met the requirements

of 10 CFR Parts 50 and 100. (Kripps, Tr. 4702). Applicants were unable to state whether, among all sites meeting 10 CFR Parts 50 and 100 criteria, any of them were preferable to others in terms of radiological risk. (Kripps, Tr. 4701).

194. Based upon the above facts, the analysis of alternative sites by Staff and Applicants gives insufficient weight to relative radiological risk.

B. Relative Population Density

195. In terms of population density, the Hartsville, Yellow Creek, Hanford, Savannah River, and INEL sites are all at least twice as good as the Clinch River site and up to 5 times lower in population. (Findings 179, 181, 183, 185, and 187).

196. Staff nevertheless concluded that none of the alternative sites is environmentally preferable to the Clinch River site with regard to population density. (Staff Exh. 8, pp. L-12, L-31, L-37 - L-38, L-42, L-48; Staff Exh. 15, pp. 22-23, Tr. 4886-87). According to Staff, even a site with no residential population within 10 miles of the LMFBR site would not be preferable to the Clinch River site in terms of population density. (Soffer, Tr. 4819).

197. Applicants also concluded that none of the alternative sites were environmentally preferable to the Clinch River site in terms of population density, despite the large actual differences in population density. (App. Exh. 45, pp. 12-15, Tr. 4744-47).

198. Based upon the above facts, Staff and Applicants gave insufficient weight to relative population density in their alternative siting analysis.

Contentions 7(a) and 7(b)

7. Neither Applicants nor Staff have adequately analyzed the alternatives to the CRBRP for the following reasons:
- a) Neither Applicants nor Staff have adequately demonstrated that the CRBRP as now planned will achieve the objectives established for it in the LMFBP Program Impact Statement and Supplement.
 - (1) It has not been established how the CRBR will achieve the objectives there listed in a timely fashion.
 - (2) In order to do this it must be shown that the specific design of the CRBR, particularly core design and engineering safety features, is sufficiently similar to a practical commercial size LMFBP that building and operating the CRBR will demonstrate anything relevant with respect to an economic, reliable and licensable LMFBP.
 - (3) The CRBR is not reasonably likely to demonstrate the reliability, maintainability, economic feasibility, technical performance, environmental acceptability or safety of a relevant commercial LMFBP central station electric plant.
 - b) No adequate analysis has been made by Applicants or Staff to determine whether the informational requirements of the LMFBP program or of a demonstration-scale facility might be substantially better satisfied by alternative design features such as are embodied in certain foreign breeder reactors.

I. THERE ARE ALTERNATIVE DESIGN APPROACHES WHICH ARE SUBSTANTIALLY BETTER THAN THOSE PRESENTLY PROPOSED

A. A More Cautious Steam Generator Testing Program Would be a Substantially Better Design Approach in Terms of Minimizing the Technological Risk of the CRBR Project.

199. Applicants do not plan to conduct complete and thorough tests of the steam generator design to be used in the CRBR prior to installation. Instead, Applicants plan to conduct

(1) a series of limited tests on a steam generator which differs significantly from those designed for use in the CRBR, (2) a vibration test on a one-third scale model steam generator, and (3) some in-plant testing on a CRBR steam generator after all CRBR steam generators have been fabricated. (Int. Exh. 22, Attachment 2, Tr. 6250).

200. The U.S. General Accounting Office ("GAO"), in a report entitled "Revising the Clinch River Breeder Reactor Steam Generator Testing Program Can Reduce Risk, (GAO/EMD-82-75, May 25, 1982), concluded that Applicants are not minimizing risks in their steam generator testing program in that:

--model steam generator testing and prototype fabrication were conducted concurrently, thus deficiencies found in the models were not corrected in the prototype;

--prototype testing involves testing a design which is significantly different from the design for the CRBR steam generators;

--prototype testing will not include simulating important operating conditions; and

--the steam generator design to be used in the CRBR will not be completely and thoroughly tested prior to fabrication and installation of all CRBR steam generators.

(Id. at 5, Tr. 6254; Staff Exh. 21, p. 6, Tr. 6527).

201. Applicants' proposed test program would not provide complete and thorough information regarding two critical breeder reactor steam generator problems--structural integrity and ability to withstand sharp temperature transients. (Id. at 9, Tr. 6258).

202. A prime contractor on the CRBR project, Westinghouse, also recognizes that the planned steam generator tests would not provide data concerning structural integrity or temperature transients. (Id. at 8, Tr. 6257).

203. The firm which designed and fabricated the prototype steam generator for CRBR, Atomics International, supports GAO's conclusion that more thorough testing of the steam generators is called for to minimize risk. (Id.)

204. Applicants' position on steam generator testing is inconsistent with its programs to test other critical CRBR components, such as sodium pumps. (Id.)

205. A cautious, conservative, and prudent approach is called for in the critical steam generator decision, and changes in the likely need for breeders makes the delay that would entail a viable option. (Id. at 9, Tr. 6258).

206. Minimizing risks through a more complete and thorough steam generator testing program is far more attractive than the risk associated with purchasing steam generators which may not operate as required, and constitutes a substantially better alternative than the proposed program. (Id. at 10, Tr. 6259).

207. Staff relies in part on steam generator experience at other LMFBRs for its confidence that the CRBR steam generators will perform as required. (Becker, Tr. 6528). However, Staff admitted that none of those other breeder steam generators are physically similar to the proposed CRBR steam generators. (Becker, Tr. 6474, 6475).

B. Inclusion of a Core Catcher in the CRBR Design Would Be a Substantially Better Alternative in Terms of Protection Against Core Melt Accidents.

208. Intervenors' Findings of Fact on Contentions 1, 2, and 3 indicate that the likelihood and consequences of a core melt accident at CRBR is substantially greater than that projected by Applicants and Staff, and that such an accident should be considered as within the design basis for CRBR. (Findings 1 to 132).

209. If a core melt accident is considered within the design basis for CRBR, a core catcher would be a prudent design feature to include (as it was included in the former Parallel Design) (Long, Tr. 6491).

210. Staff's testimony that a core catcher would not likely provide any useful information since the probability of its being used is extremely low (Long, Tr. 6547-48), is inconsistent with the contentions of both Applicants and Staff that design and construction of CRBR will themselves provide useful information for the LMFBR program. Staff admitted on cross-examination that useful information could be obtained from design, construction, and testing of a core catcher. (Long, Tr. 6488). Therefore, inclusion of a core catcher in the CRBR design would be a substantially better alternative in terms of protection against core melt accidents, and in terms of providing informational benefits to the LMFBR program, than not including this design feature.

C. Inclusion in the CRBR of a No-Vent Containment Would Be a Substantially Better Alternative in Terms of Reducing Radiological Consequences of Accidents to the Public.

211. The CRBR as proposed includes a vented containment; i.e., a vent and purge system. (Findings 83 and 84).

212. Applicants' and Staff's site suitability source term analyses fail to include the implications of the vent purge system. (Findings 81 to 87).

213. The current CRBR containment/confinement system is inadequate to reduce the radiological dose consequences from CRBR accidents to a sufficiently low level. (Findings 1 to 132). Therefore, inclusion in the CRBR of a no-vent containment able to withstand the pressure and thermal consequences of a CDA would be a substantially better alternative than the present containment/confinement design in terms of reducing the radiological risk to the public from a CDA.

II. STAFF'S ANALYSIS OF PROGRAMMATIC OBJECTIVES AND WHETHER CRBR IS SUBSTANTIALLY LIKELY TO MEET THEM IS INADEQUATE

A. Under the Present Timing "Objective", It Cannot be Determined Whether CRBR Will Meet its Programmatic Objectives in a Timely Fashion.

214. The current "timing objective" for the LMFBR demonstration plant is to complete its construction "as expeditiously as possible," (Leech, Tr. 6523) or "as soon as possible." (Longenecker, Tr. 6410).

215. The 1976 Commission decision in this case held that, although "timing" of the LMFBR demonstration plant is a programmatic given, whether the objectives would be achieved in a timely fashion is a legitimate issue for litigation in this proceeding. (CLI-76-13, 4 NRC 67, at 78, 92.)

216. Staff did not conduct a review to determine whether CRBR will be constructed as expeditiously as possible. (Leech, Tr. 6470). Staff does not believe it is necessary to make such a determination in order for it to make its recommendation on issuance of an LWA-1. (Id.)

217. Applicants and Staff have asserted that because no alternatives could meet the objectives in a more timely fashion than the proposed CRBR, the alternatives would not meet the timing objective. (Leech, Tr. 6524; Longenecker, Tr. 6319). Thus, satisfaction of the timing objective is determined by whether or not the action under consideration is the proposed action.

B. Staff's Analysis of the Economic Feasibility Objective is Meaningless.

218. Staff interprets the CRBR's economic feasibility objective to be met by the existence of a detailed cost accounting system to collect information. (Long, Tr. 6476, 6484).

219. In analyzing whether CRBR demonstrates that an LMFBR is economically feasible, Staff does not consider the actual costs of CRBR to be relevant. (Long, Tr. 6477, 6485).

220. In other words, no matter how high the actual cost of CRBR gets, Staff will consider the project to have met its objective of demonstrating economic feasibility simply because it has provided information.

C. Staff's Analysis of Other Programmatic Objectives is Meaningless.

221. Staff believes CRBR would meet its technical performance and reliability objectives even if the plant were a technical failure. For example, Staff testified that even if CRBR had a steam generator explosion during the five-year demonstration period, it would still meet its technical performance and reliability objectives by providing that information. (Long, Tr. 6470). Likewise, Staff testified that if CRBR had a core disruptive accident exceeding the 661-megajoule primary containment design, the objective would still be met because of the relevant information provided. (Long, Tr. 6472).

222. In other words, no matter how bad the actual performance is, Staff considers that the technical performance and reliability objectives will be met because of the information provided.

Contentions 4 and 6(b)(4)

4. Neither Applicants nor Staff adequately analyze the health and safety consequences of acts of sabotage, terrorism or theft directed against the CRBR or supporting facilities nor do they adequately analyze the programs to prevent such acts or disadvantages of any measures to be used to prevent such acts.
 - a) Small quantities of plutonium can be converted into a nuclear bomb or plutonium dispersion device which if used could cause widespread death and destruction.
 - b) Plutonium in an easily usable form will be available in substantial quantities at the CRBR and at supporting fuel cycle facilities.
 - c) Analyses conducted by the Federal Government of the potential threat from terrorists, saboteurs and thieves demonstrate several credible scenarios which could result in plutonium diversion or releases of radiation (both purposeful and accidental) and against which no adequate safeguards have been proposed by Applicants or Staff.
 - d) Acts of sabotage or terrorism could be the initiating cause for CDAs or other severe CRBR accidents and the probability of such acts occurring has not been analyzed in predicting the probability of a CDA.
- 6(b)(4) The impact of an act of sabotage, terrorism or theft directed against the plutonium in the CRBR fuel cycle, including the plant, is inadequately assessed, [as] is the impact of various measures intended to be used to prevent sabotage, theft or diversion.

I. THE CRBR AND ITS SUPPORTING FUEL CYCLE
PRESENT NEW, DIFFERENT AND GREATER SAFEGUARDS RISKS

A. Plutonium Availability

223. Substantial quantities of plutonium will be associated with the CRBR and related fuel cycle facilities. The initial core-loading of the CRBR will be approximately 1.7 metric tons of

plutonium. (Staff Exh. 8, FSFES, p. D-2). At equilibrium, the CRBR will utilize approximately .9 metric tons of plutonium in its fuel and blanket assemblies per year, discharging spent fuel elements containing approximately 1000 kilograms of plutonium per year. (Staff Exh. 8, FSFES, p. D-6). Similar quantities of plutonium, i.e., approximately 1000 kilograms per year, will constitute the throughput at fuel cycle facilities associated with the CRBR. (Cochran, Tr. 3847; Int. Exh. 12, pp. 5-6, Tr. 3892-93; Staff Exh. 10, p. 10, Tr. 3742).

224. These figures, i.e., the annual CRBR plutonium requirements, are a substantial fraction of the entire amount of plutonium produced annually in the weapons program in years past. (Cochran, Tr. 3847).

225. The quantities of plutonium associated with the CRBR and related fuel cycle facilities are unique in the context of commercial power generation. (Dube, Tr. 3730; Hammond, Tr. 3433, 3437, 3440). Indeed, operation of the CRBR will mark the first time that significant quantities of separated plutonium have been used in a power reactor system. (Dube, Tr. 3730).

B. Special Safeguards Risks of Plutonium Use

226. While there is some expert difference of opinion over the difficulty of doing so, there is no dispute that it is possible to use plutonium and fresh CRBR fuel for illicit weapons purposes. (Hockert, Tr. 3702-03, 3708-09).

227. A clandestine fission explosive ("CFE") can be made directly from fresh CRBR fuel without the need for chemical separation. (Cochran, Int. Exh. 12, p. 7, Tr. 3894). Only 6 to 12 kilograms of plutonium would be necessary to construct such a device. (Cochran, Int. Exh. 12, p. 7, Tr. 3894). Similarly, a plutonium dispersal device could be fabricated by terrorists or saboteurs directly from fresh CRBR fuel, perhaps with only a few grams of such fuel. (Cochran, Int. Exh. 12, pp. 7-8, Tr. 3894-95). A CFE would cause both physical and radiological effects, while a plutonium dispersal device could be used to produce cancers (principally lung) in humans and to contaminate buildings, large areas of land, etc. (Cochran, Int. Exh. 12, p. 8, Tr. 3895; Int. Exh. 12, pp. 15-17, Tr. 3902-04).

228. Relatively small quantities of plutonium fuel are of safeguards significance. Any amount of plutonium larger than two kilograms is a "formula quantity" as defined under 10 CFR Section 73.2(bb), which triggers safeguards requirements under the Commission's regulations, 10 CFR Pt. 73. One formula quantity is less than that generally considered necessary to construct a CFE. (Cochran, Int. Exh. 12, p. 7, Tr. 3894).

229. A CFE could be designed and constructed by a single individual or a small group of people, none of whom has ever had access to the classified literature and none of whom has advanced nuclear training, using generally available equipment, supplies and techniques. (Cochran, Int. Exh. 12,

p. 7, Tr. 3894; Staff Exh. 10, p. 9, Tr. 3741).

230. The Commission's "operating assumption" is that a CFE could be successful on the first try and produce a yield substantially greater than the yield of an equivalent quantity of high explosives. (Staff Exh. 8, FSFES, p. E-4; Hockert, Tr. 3711). Construction of a CFE with an equivalent yield of 1000 tons of TNT is well within the range of possibility. (Cochran, Int. Exh. 12, pp. 15-17, Tr. 3902-04).

231. For purposes of constructing an illicit weapon, fresh CRBR fuel is "preferable" to anything in the conventional LWR fuel cycle. (Cochran, Int. Exh. 10, p. 14, Tr. 3901).

232. Because of the nature of plutonium fuel, the safeguards risks associated the CRBR fuel cycle are greater than the risks associated with the conventional LWR fuel cycle. (Hammond, Tr. 3434-35).

C. Consequences of Diversion or Theft

233. The environmental consequences of detonation of a CFE would be severe, including immediate physical destruction and radiation health hazards. (Cochran, Int. Exh. 12, pp. 15-17, Tr. 3902-04; Staff Exh. 7, p. 7-23; Staff Exh. 10, p. 5, Tr. 3737). This means something comparable to 0.1 to ten times the destruction experienced at Nagasaki with the detonation of a plutonium device. (Cochran, Int. Exh. 12, p. 15, Tr. 3902). "[I]f a workable illicit device of even

modest yield were cleverly placed and detonated, thousands of people could be killed and millions of dollars worth of property could be destroyed." (Staff Exh. 7, p. 7-23).

234. Likewise, the consequences of use of a plutonium dispersal device could be severe. (Cochran, Int. Exh. 12, pp. 7-18, Tr. 3904-05; Staff Exh. 7, pp. 7-23; Staff Exh. 10, p. 5, Tr. 3737). Plutonium dispersion, in addition to having toxic effects by causing long-term cancers, could cause wide-spread contamination, the clean-up of which could be extremely costly, i.e, several hundred million dollars. (Cochran, Int. Exh. 12, pp. 15-16, Tr. 3902-03). Dispersal of a plutonium device into a ventilation system of a large office building might produce 70-80 latent cancer fatalities per gram of plutonium effectively dispersed. (Hockert, Tr. 3665).

235. While other environmental contaminants might also cause severe damage if released (Hockert, Tr. 3664), there is no evidence that equivalent quantities of these contaminants are associated with any civilian programs or otherwise move in commerce. (Hockert, Tr. 3666).

236. It is the position of Staff that the environmental consequences of the successful use of either a CFE or a plutonium dispersal device are "unacceptable." (Hockert, Tr. 3586, 3591).

237. In addition to environmental and health effects, civil liberties consequences may be associated with the successful theft of plutonium at the CRBR or its supporting

fuel cycle facilities. (Cochran, Tr. 3849; Int. Exh. 12, pp. 18-109, Tr. 3905-06). A variety of civil liberties restrictions, including search without warrant, arrest without warrant, widespread searches, and even marshal law, are possible. (Cochran, Tr. 3849, Int. Exh. 12, pp. 18-19, Tr. 3905-06). These civil liberties restrictions would not necessarily involve violations of law; they could be imposed consistent with the requirements of existing law. They would, however, entail social costs which have not been analyzed by Applicants or Staff. (Staff Exh. 8, FSFES, App. E).

D. The Threat to the CRBR and Its Supporting Fuel Cycle Facilities

238. There have been numerous attacks on, and both theft and sabotage attempts at, nuclear power plants and fuel cycle facilities in the past twenty years. (App. Exh. 40; Cochran, Int. Exh. 12, pp.12-13, Tr. 3899-3900). From 1966 through 1979, approximately 39 incidents in the United States and abroad have been documented. (App. Exh. 40). This history indicates that such facilities are attractive targets.

239. It is difficult (and perhaps impossible) to design a personnel system that will assuredly detect potential saboteurs. (Penico, Tr. 3274). In particular, psychological testing may not be sufficient to pick up the incipient or potential saboteur. (Penico, Tr. 3274).

240. While there is no evidence of an attack or

sabotage attempt at a facility with protections equivalent to those at the CRBR and its supporting fuel cycle facilities (Cochran, Tr. 3800-17), the nature of the safeguards and physical security features at a plant is not conclusive as to the nature of the threat which might be directed against it. There is evidence that, in mounting an attack or sabotage attempt, a terrorist or saboteur will take the level of protection into account and then develop the forces necessary to overcome those protections. (Jones, Tr. 3593). Further, there may be an element of irrationality in the terrorist mentality which could lead to attacks against particular facilities, despite rational assessments that the chances of success were slim. (Jones, Tr. 3596).

241. The CRBR and its supporting fuel cycle facilities may be higher risk targets than conventional nuclear facilities. This is so because plutonium used in the CRBR represents a preferred material for the construction of atomic bombs, as opposed to material that would be extracted from high burnup fuel in conventional light water reactors, and because the CRBR, as the first demonstration use of plutonium in the United States, has high visibility and symbolic importance. (Cochran, Int. Exh. 12, pp. 14-15, Tr. 3901-02).

II. STAFF'S REVIEW OF FUEL
CYCLE RISKS WAS INSUFFICIENT

A. Inadequate Review Criteria

242. With respect to safeguards, "the objectives of both the Commission and DOE are 'to provide high assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety.'" (App. Exh. 35, p. 5.7-37).

243. In assessing safeguards risks and consequences at the CRBR and its supporting fuel cycle facilities, Staff adopted three criteria:

1. Do DOE's proposed safeguards systems provide a potential for deterring attempts at theft or diversion of plutonium and attempts at sabotage of facilities or materials to be used in the CRBRP fuel cycle?
2. Are DOE's proposed safeguards systems likely to detect attempts at sabotage, theft, or diversion?
3. Do DOE's proposed systems for responding to attempted theft, diversion, or sabotage provide reasonable assurance that such attempts would not be successful?

(Staff Exh. 8, FSFES, p. E-1).

244. The three criteria set forth above were the primary basis for Staff's safeguards judgments (Dube, Tr. 3644-45; Staff Exh. 10, p. 7, Tr. 3739).

245. The three criteria utilized by Staff do not provide "high assurance" that safeguards objectives will be met or, in fact, that the Commission's safeguards regulations will be satisfied. (Dube, Tr. 3682-83; Cochran, Int. Exh. 12, pp. 23-24, Tr. 3910-11). Indeed, Staff's ultimate

conclusion is simply that there is a "potential" for providing adequate safeguards at CRBR fuel cycle facilities. (Staff Exh. 10, p. 14, Tr. 3746).

B. Insufficiency of Staff's Comparability Analysis

246. In analyzing safeguards risks at CRBR fuel cycle facilities, Staff made certain comparative judgments in order to arrive at a conclusion that risks associated with the CRBR and its fuel cycle are not greater than risks associated with other, similar licensed and non-licensed facilities. (Staff Exh. 8, FSFES, p. 12-34, E-9; Staff Exh. 10, pp. 12-13, Tr. 3744-45).

247. In making these judgments, Staff did no more than examine DOE safeguards regulations (DOE Orders 5630, 5631 and 5632) and determine if they were comparable to Commission regulations. (Hurt, Tr. 3604-05).

248. Staff did not go beyond DOE orders and examine actual risks at fuel cycle facilities to determine if they were comparable. (Dube, Tr. 3605).

249. In making its safeguards assessments, Staff did not take into account critiques made by the General Accounting Office of the efficacy of safeguards at DOE facilities. (Dube, Tr. 3601).

250. Staff did not examine or assess actual safeguards systems now in place or planned for possible future installation at CRBR fuel cycle facilities. (Dube, Tr. 3601-02).

251. Staff did not examine the records of safeguards compliance at possible CRBR fuel cycle facilities. (Dube, Tr. 3601).

252. In sum, Staff's approach necessarily resulted in only an incomplete and inadequate assessment of fuel cycle safeguards risks.

C. Lack of Independent Assessment of Applicants' Submissions

253. In carrying out its analysis of safeguards risks and consequences, Staff relied primarily on representations made by Applicants with respect to the nature of the fuel cycle facilities. (Staff Exh. 10, p. 6, Tr. 3738; Dube, Tr. 3642-43, 3684). Staff made no specific examination of the safeguards systems at either other DOE nuclear facilities or the specific facilities proposed to be part of the CRBR fuel cycle. (Dube, Tr. 3642-43, 3684).

254. Staff also relied primarily upon representations made by Applicants with respect to safeguards effectiveness at proposed fuel cycle facilities. (Dube, Tr. 3642-43, 3684). No confidence levels were attached to the figures provided by Applicants (Cochran, Int. Exh. 12, pp. 24-25, Tr. 3911-12), and no independent information was developed with respect to system capabilities (Hurt, Tr. 3600-01). Staff's review, in other words, was "based entirely on the Applicants' environmental report and documents specifically referenced in that report, which include the DOE orders." (Hurt, Tr. 3600-01).

III. STAFF CANNOT PROPERLY RELY UPON
APPLICANTS' ASSURANCES WITH RESPECT
TO FUEL CYCLE SAFEGUARDS

A. Uncertainties With Respect to Effectiveness of
Reprocessing Safeguards at the DRP

255. There currently exist significant uncertainties with respect to the nature and scope of safeguards systems, and their effectiveness, at the facilities which will reprocess CRBR fuel. (Cochran, Int. Exh. 12, p. 22, Tr. 3909, Int. Exh. 12, p. 35, Tr. 3922).

256. The Developmental Reprocessing Plant ("DRP"), where Applicants would like to reprocess CRBR fuel, is now only in the "conceptual design" stage; there are not actual designs for such facility. (Hammond, Tr. 3387; Hurt, Tr. 3678-79).

257. Applicants, at this time, have not quantified safeguards goals for the DRP in terms of errors in inventory balances. (Hammond, Tr. 3387).

258. Further, Applicants are not in a position to state whether the DRP design goals can actually be met (Hammond, Tr. 3379, 3381, 3387, 3407-08), and it is not yet known just what levels of performance any system can achieve. (Cochran, Int. Exh. 12, p. 35, Tr. 3922).

259. Even assuming the DRP design goals could be met, considering the entire DRP throughput (and not just the throughput attributable to the CRBR), the confidence levels on inventories (1.4% of throughput) are such that it may not be possible reliably to detect a two kilogram ("formula

quantity") diversion. (Dube, Tr. 3682; Cochran, Int. Exh. 12, p. 26, Tr. 3913). This means that there is no assurance that other safeguards systems (physical protection and material control) will be effective and that losses of safeguards significance will be detected. (Cochran, Int. Exh. 12, p. 30, Tr. 3923).

260. In sum, it cannot reasonably be concluded at this time that safeguards objectives will be met at the DRP.

B. Uncertainties with Respect to Safeguards At Reprocessing Facilities Other Than the DRP

261. The DRP may never be built. (Hammond, Tr. 3389). Reprocessing could take place elsewhere, i.e., DOE's Savannah River Plant or its Purex Plant in Hanford, Washington, or a small facility that would be built into the FMEF, depending upon the outcome of budget and appropriation decisions with respect to the DRP (Cochran, Int. Exh. 12, p. 28, Tr. 3915)(Finding 167).

262. Despite the fact that the DRP may never be built, the Staff, in analyzing reprocessing safeguards, only considered the safeguards, as described by Applicants, at the DRP; it did not look at alternative fuel cycle facilities or the capabilities of such facilities to meet safeguards objectives. (Dube, Tr. 3601, 3642-43; Hurt, Tr. 3680; Cochran, Int. Exh. 12, p. 28, Tr. 3915).

263. Whether safeguards objectives can be met at alternative reprocessing facilities is not known; the technical capabilities of other facilities are uncertain and

projected DRP performance may not be technically feasible for such facilities. (Cochran, Int. Exh. 12, p. 22, Tr. 3909).

264. As expressed in the report of the General Accounting Office, Nuclear Fuel Reprocessing and the Problems of Safeguards Against the Spread of Nuclear Weapons (EMD-80-38, March 18, 1980)(Int. Exh. 11), current safeguards systems at DOE reprocessing plants "cannot assure that diversions of weapons usable material for non-authorized purposes can be detected in a timely manner. Diversion or theft of materials sufficient to construct a nuclear weapon is possible and could go undetected." (Cochran, Int. Exh. 12, p. 35, Tr. 3922).

265. In sum, it cannot reasonably be concluded at this time that safeguards will be effective if CRBR fuel is reprocessed at existing DOE facilities.

C. Uncertainties in Needed Research and Development With Respect to Reprocessing Safeguards

266. In order for the safeguards systems at future reprocessing facilities to meet their objectives, certain R&D successes are required. (App. Exh. 39, p. 73, Tr. 3547).

267. The measurement capability of the safeguards system proposed for reprocessing of CRBR fuel has not yet been demonstrated. (Hammond, Tr. 3417; Hurt, Tr. 3690-91; Staff Exh. 8, FSFES, p. 12-70, E-13).

268. Budgetary constraints could hamper needed research effort. (Hammond, Tr. 3303).

269. Even if the money were there, research and

development payoffs cannot be guaranteed. (Hammond, Tr. 3334).

270. The scope and direction of the DOE reprocessing safeguards R & D program have been subject to substantial criticism by the General Accounting Office which, in its report, Nuclear Fuel Reprocessing and the Problems of Safeguarding Against the Spread of Nuclear Weapons (EMD-80-38, March 18, 1980)(Int. Exh. 11), has stated that the DOE program "lacks direction and control." (Hammond, Tr. 3314-15, 3325; Int. Exh. 11, p. 51). As a result, while upgrade work may improve safeguards effectiveness, it is "uncertain how much the diversion risks will be reduced." (Cochran, Int. Exh. 12, p. 29, Tr. 3916; Int. Exh. 11 at 10).

271. In sum, it cannot reasonably be concluded at this time that R & D efforts will be successful and obstacles to creating an effective reprocessing safeguards system overcome.

D. Future Fuel Cycle Compliance Uncertainties

272. The Commission exercises no regulatory authority over DOE fuel cycle facilities, and, if DOE commitments relative to fuel cycle safeguards are not implemented, there is nothing the Commission can do about it. (Cochran, Int. Exh. 12, pp. 33-34, Tr. 3920-21).

273. While Applicants make certain commitments in the Environmental Report (App. Exh. 35) with respect to safeguards programs (and their effectiveness) at fuel cycle

facilities, there are no additional written assurances that "commitments" will be honored. (Hammond, Tr. 3307). Staff, moreover, has no criteria for concluding that compliance with applicable safeguards regulations is likely. (Cochran, Int. Exh. 12, p. 29, Tr. 3917, Int. Exh. 12, p. 34, Tr. 3921).

274. No analysis has been undertaken of the empirical likelihood that Applicants' commitments will be met, i.e., by examining compliance at current facilities. (Dube, Tr. 3684; Hurt, Tr. 3692).

275. DOE orders, moreover, are general in nature; they do not indicate precisely which systems or technologies should be employed. Indeed, they do not provide for incorporation of "best available technology." (Hammond, Tr. 3308-09).

276. Under the DOE orders, the Operations Office, not Headquarters, will make final decisions with respect to incorporation of particular technologies and systems. (Hammond, Tr. 3309).

277. What particular systems or technologies will be incorporated in the future at fuel cycle facilities will largely depend upon future priorities and which advocate will be most persuasive in future deliberations about such priorities. (Pencio, Tr. 3467; Hammond, Tr. 3455). The future or shape of fuel cycle safeguards thus cannot be predicted with certainty at this time.

278. Given this uncertainty, there is no assurance that the latest or best safeguards systems or technologies will be

utilized at CRBR fuel cycle facilities.

279. Moreover, even if advanced systems or technologies are utilized, the Commission may not have reliable data to judge their effectiveness. (Cochran, Int. Exh. 12, p. 39, Tr. 3926).

280. In sum, it cannot reasonably be concluded at this time that fuel cycles systems and technologies will be utilized which will ensure an acceptably low safeguards risk.

E. Changing Threat Levels and the Level of Preparedness

281. Threat levels against the CRBR and its fuel cycle facilities may change over time. (Cochran, Int. Exh. 12, p. 38, Tr. 3925; Hammond, Tr. 3421).

282. Small changes in the threat level, i.e., on the order of one or two persons, are difficult to detect. (Penico, Tr. 3422-23).

283. Further, it may be difficult to detect a threat before it actually materializes. In other words, there is no assurance that there will be advance warning of a threat change unless group size becomes very large. (Cochran, Int. Exh. 12. pp. 38-39, Tr. 3925-26).

284. However, upgrading of regulations may take from several months to several years. (Cochran, Tr. 3835; Penico, Tr. 3427; Dube, Tr. 3686-87). The result is that for at least some period of time facilities may be unprepared to deal effectively with increased threat levels.

285. In sum, it cannot reasonably be concluded at this

time that the system will respond with sufficient speed to assure that safeguards risks are acceptably low.

F. Lack of Independent Effectiveness of Material Control and Accounting and Physical Security Systems

286. Material control and accounting systems and physical security systems, as planned and intended to be implemented by Applicants, are not independently effective in deterring, detecting and thwarting safeguards threats. (Hammond, Tr. 3363-64, 3432; Dube, Tr. 3695).

287. However, material accounting "provides the only means for assuring that the physical protection and material control systems are effective and that no significant losses or diversions have gone undetected." (Cochran, Int. Exh. 12, p. 36, Tr. 3923).

288. Given the lack of independent effectiveness of the material control and accounting and physical security systems, it cannot reasonably be concluded at this time that safeguards objectives, i.e., high assurance of deterrence, detection and apprehension of diversion or theft of formula quantities of special nuclear material, can or will be achieved at CRBR fuel cycle facilities.