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

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

TESTIMONY OF DR. THOMAS B. COCHRAN

PART IV

AS SUPPLEMENTED BY NEW INFORMATION IN CRBR FINAL ENVIRONMENTAL IMPACT STATEMENT SUPPLEMENT

(Intervenors' Contentions 1, 2, and 3)

DATED: November 12, 1982

Q.1: Please identify yourself and state your qualifications to present this testimony.

A.2: My name is Thomas B. Cochran. I reside at 4836 North 30th Street, Arlington, Virginia 22207. I am a Senior Staff Scientist at Natural Resources Defense Council, Inc. My background and qualifications to present this testimony are presented in previous testimony in this proceeding. (Tr. 2870-71, Cochran.)

Q.2: What is the subject matter of the present testimony?

A.2: Part IV of my testimony deals with the potential for severe accidents at CRBR and the adequacy of Applicants' and Staff's analyses of those accidents. These are matters that are raised in Intervenors' Contentions 1, 2, and 3. For purposes of this phase of the proceeding, those Contentions read as follows:

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.

   b) [deferred]

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 §100.11(a).

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 8(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 that 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.

3. Neither Applicants nor Staff have given sufficient attention to CRBR accidents other than the DBAs for the following reasons:

   a) [deferred]

   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.

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

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.

The accident discussion at this phase focuses on Appendix J of the Final Supplement to the FES, NUREG-0139, Supplement No. 1 (henceforth "FSFES").

Q.3: Dr. Cochran, are you familiar with Staff's NEPA analysis of the risks of potential accidents associated with the CRBR?

A.3: Yes.

Q.4: Where is this analysis set forth?

A.4: Primarily in Chapter 7 and Appendix J of the FSFES, although some paragraphs from Chapter 7 of the 1977 FES have been retained, including the conclusions in §7.1.4.

Q.5: Do you have general criticisms of Appendix J?

A.5: Yes. The methodology in Appendix J is crude by today's standards, and the assumptions behind it (and the input data) are not supported by any substantive analysis. While it presents estimates of the absolute probability of CRBR accidents, these estimates are backed up by no calculations and no event tree/fault tree analyses as one finds in risk assessment analyses such as the Reactor

Safety Study (WASH-1400) and CRBRP-1. No operating data are offered in support of its conclusions, and there are no quantified estimates of the uncertainty associated with the probability estimates. It must be remembered that WASH-1400, which contained an incomparably more detailed analysis of accident probabilities for two actual LWRs (and which is, incidentally, the direct progenitor of virtually all nuclear risk assessment work) was severely criticized for making unsupported assumptions, for failing to properly assess uncertainty and for its factual inscrutability. For these reasons, the NRC ultimately repudiated WASH-1400's absolute probability predictions. Yet, compared to Appendix J, WASH-1400 was a model of scientific analysis. Appendix J is not even supported by a plant-specific risk assessment. Its assumptions are not just unsupported by rigorous analysis; for the most part, they are not even presented for evaluation. If WASH-1400's probability estimates were unreliable, as the Commission correctly concluded, then the probability estimates in Appendix J are far more so. There is no reason to accept these on faith, and very little beyond faith is offered.

Moreover, the Staff attempt to quantitatively assess the uncertainty associated with the estimates for various quantitative accident probabilities and consequences presented in Appendix J is a one-sentence conclusory statement (FSFES, p. J-24) which is unsupported in the document by rigorous analysis. Probably the most serious criticism of WASH-1400 from the scientific community was its failure to assess or properly acknowledge the very large uncertainties attached to absolute probability predictions. Those uncertainties, which have been estimated to be as large as a factor of 100 in some cases, must be much greater for predicting CRBR accident probabilities, since the body of relevant operating data for LMFBRs is far less than for LWRs and since, for lack of a plant-specific assessment, the report is almost totally based on conclusory statements that can most charitably be characterized as "engineering judgment." Without some reasonable and scrutable assessment of the uncertainties inherent in these predictions, they are simply arbitrary and meaningless.

Q.6: Do you know whether the NRC Staff performed any calculations, reviewed operating data for other facilities, or did any plant-specific assessment of the reliability of the CRBR systems to back up the probability estimates presented in Appendix J?

A.6: According to the NRC Staff, with only three exceptions (WASH-1400 for PWR auxiliary feedwater reliability and the probability of loss of offsite power, and NUREG-0460 for the frequency of anticipated transients without scram for typical LWRs), they did not. NRDC asked the Staff in discovery to identify the documents relied upon for each of the principal probability assessments in Appendix J. (See Staff Response to NRDC's 27th Set of Interrogatories, Oct. 1, 1982, pp. 53-70.) In almost every case, the Staff responded under oath that it relied on <u>no</u> "specific" documents for any of the conclusions presented, instead relying generally on the "cumulative knowledge" of the Staff and its consultants in general, or a similar response. While "engineering judgment" or "cumulative knowledge" is valuable for many purposes, it is not sufficient to support predictions of the probability of serious accidents in a plant as complex and untested as the CRBR.

Q.7: Have you been limited in your ability to independently assess the probability of accidents beyond the design basis for CRBR?

A.7: Yes, independent assessment has been greatly hindered. The probability of a catastrophic accident in any plant is a function of the plant design, the potential for equipment malfunction and human error, and the reliability of its many complex systems and components. The CRBR is the first plant of its kind. Applicants have done much work in assessing the reliability of the CRBR design, primarily as part of Applicants' Reliability Program (see PSAR, Appendix C). The document known as CRBRP-1 is another prominent example. Applicants have underway a comprehensive probabilistic risk assessment (PRA) of the CRBR and preliminary results have been presented to the ACRS and the Staff (cf., Letter from John R. Longenecker, CRBR Project to Paul S. Check, USNRC, June 21, 1982, subj: Probabilistic Risk Assessment (PRA) Program Plan). However, the scope of this LWA-1 proceeding has been limited to exclude inquiry into what are termed the "details" of the CRBR design. CRBRP-1 has been expressly excluded from consideration. In my judgment, no reliable estimate of CRBR accident probabilities can be made within the present scope of the LWA-1 proceeding and without reviewing the CRBR design in some detail. This has not been possible at this stage.

Q.8: Do you believe that the analysis in Appendix J is realistic and adequate to support Staff's conclusions regarding consequences of Class 9 accidents, namely "that CRBR accident risks would not be significantly different from those of current LWRs..." and that "the accident risks at CRBR can be made acceptably low." (Appendix J, p.

J-25)?

A.8: No.

Q.9: Please proceed to discuss some of the specific probability estimates. To begin, what frequency of occurrence did the NRC staff assign to core degradation due to LOHS (loss of heat sink) events for CRBR and what rationale did the staff give for its estimate?

A.9: Staff assigned a frequency of core degradation due to LOHS events of less than 10^{-4} per reactor year (i.e., one chance in 10,000 per reactor year). Staff cited three principal factors for this result:

1. A "general consideration of typical achievable PWR auxiliary feedwater system reliabilities;"
2. The "potential for common cause failures;"
3. The potential for achieving "high reliability in final design and operation through an effective reliability program." (FSFES, pp. J-3, -4.)

While the three factors above are all listed as the bases for the estimated LOHS probability, only the first -- PWR auxiliary feedwater system reliability -- serves as the basis for Staff's quantified estimate. The role the other two factors play in the choice of the 10^{-4}/year estimate is discussed only in the most general qualitative terms, e.g., "... unavailability estimates for ... heat removal systems have been set high enough to include allowance for potential common mode failures" (Appendix J, p. J-22). The choice of auxiliary feedwater system failure as the controlling failure mode is not justified. In other words, there is no reason to believe that failures in systems other than auxiliary feedwater may not contribute significantly to the LOHS probability. A fault tree analysis is necessary to justify limiting the discussion to auxiliary feedwater reliability.

In order to illustrate the complexity of this issue, consider the generalized fault model for the shutdown heat removal system for CRBR taken from CRBRP-1, Vol. 2, Appendix II, p. 2-14 to 2-22 (attached to my testimony as Exhibit 1). This fault tree, which is developed to the system (or subsystem) level rather than the more detailed component level as in the WASH-1400 case, can be considered applicable to a reactor of the general size and type as CRBR. Clearly, it takes a leap of faith to conclude that the failure rate of the auxiliary feedwater system controls the overall frequency of core degradation due to LOHS events.

Q.10: Setting aside your view that there is no basis for concluding that the failure rate of the auxiliary feedwater system is controlling, do you agree with the

Staff's estimate of the feedwater system reliability? Explain your answer.

A.10: First, I should note that Staff claims that its estimate of the probability of LOHS events was based on independent analyses, primarily by William Morris of the Staff and Staff consultant Edward Rumble of Science Applications Inc., (SAI), each using a different base of information (Deposition of William Morris, Oct. 12, 1982, pp. 24-25).

Dr. Morris claimed his estimate is based on the reliability of auxiliary feedwater systems in PWRs over the years as documented in the Standard Review Plan for LWR feedwater systems (Morris, Deposition of Oct. 12, 1982, pp. 23-24).

Mr. Rumble also claimed his estimate was based on reliability studies of PWR auxiliary heat removal systems, the Accident Delineation Studies (Phases 1 and 2) (NUREG-CR-1407 is Phase 1) prepared by Sandia for NRC-NRR, and the study CRBRP-1 (which is beyond the scope of the LWA-1 proceeding). Mr. Rumble said these estimates were what he believed should be achievable, not necessarily what has been achieved to date (E.R. Rumble, private telephone communication, July 27, 1982, as noted in T.B. Cochran Memo to Files, July 27, 1982).

I do not agree with Staff's estimate or Staff's underlying analysis. First, LOHS fault trees for CRBR developed in CRBRP-1 differ from those of a PWR as developed in WASH-1400, and consequently there is no obvious correlation between PWR system reliabilities and the core degradation frequency due to LOHS accident scenarios in CRBR. This can be seen by comparing the generalized fault models for CRBR shutdown heat removal (see CRBRP-1, Vol. 2, Appendix II) with the fault models for a PWR (see WASH-1400, App. II).

Staff claims that its estimate of 10^{-4}/year is based on "typical achievable PWR auxiliary feedwater system reliabilities" (Appendix J, p. J-4). If this is so, there must be wide variations in achievable feedwater system reliability. For example, the RSSMAP (Reactor Safety Study Methodology Applications Program) report for Calvert Cliffs (NUREG/CR-1569) concluded that the probability of core melt for Calvert Cliffs was 1 chance in 2400 per reactor year, largely due to unreliabilities in the auxiliary feedwater system and failure of backup heat removal methods. This result is a factor of 4 larger than the Staff's alleged "upper bound" result for CRBR. No justification has been presented for concluding that he CRBR auxiliary feedwater system will be more reliable than Calvert Cliffs by at least a factor of four. Furthermore, there is a serious question about the comparability of PWR operating data in this area to the CRBR. It should be noted in this connection that the authors of the Applicants' risk assessment work felt that the WASH-1400 data could not be applied to the question of unavailability of decay heat removal systems for CRBR. Instead, a fault tree analysis was conducted to determine the system availability. (CRBRP-1, Vol. 2, at III-3.)

There is no basis for concluding that CRBR's auxiliary feedwater system will be "typical" in its reliability. The conservative assumption to make at this juncture might be to assume that CRBR's auxiliary feedwater system will be no better than Calvert Cliffs' system. Moreover, since CRBR's Decay Heat Removal System (DHRS) is dependent upon AC electrical power, it cannot be assumed to be significantly more reliable than PWR DHRSs; according to Staff (FSFES, pp. J-3,4), a principal unreliability in PWR decay heat removal systems is not in system failures per se but in loss of offsite and onsite AC power. Thus, if Staff is correct, the ability of the CRBR DHRS to operate at "normal" temperature and pressure (whereas PWR DHRSs can operate only at low pressure) should not have a major impact on overall risk.

Q.11: Are there other CRBR heat removal systems that are important in terms of the comparability between the frequencies of core degradation in CRBR and PWRs due to loss of heat sink (LOHS)?

A.11: What I noted above was that one cannot tell the degree of contribution that various component failures have on the overall failure rate without a detailed fault tree analysis. However, it is evident that there are other CRBR heat removal components whose failure rates are not necessarily comparable to PWR systems. The steam generators are an example. There is no discussion whatever in Appendix J of the contribution of steam generator failure to the overall risk of LOHS, nor of the possible mechanisms or modes of failure considered. Unlike an LWR, the steam generators in an LMFBR, such as CRBR, represent a location where significant amounts of sodium and water are in close proximity. CRBR event sequences can be postulated, e.g., propagation of steam generator tube failures, where sufficient water and sodium can be brought together in such a manner as to create a sodium-water reaction coupled with a hydrogen reaction, resulting in loss of the shutdown heat removal function (see generally CRBRP-1, Appendix VIII).

The General Accounting Office in a recent letter to Congress was highly critical of DOE's failure to conduct complete and thorough tests of the steam generators to be used in the CRBR, in spite of the fact that steam generators for LMFBRs have had a history of serious technical problems and the fact that development and demonstration of reliable steam generators have been and still are one of the most significant technical problems facing the CRBR project. (Letter from Charles A. Bowsher, Comptroller General, to Congressman John D. Dingell, May 25, 1982, GAO/EMD-82-75, attached as Exhibit 2).

In sum, because of the inherent differences in the shutdown heat removal systems, e.g., steam generators, between PWRs and LMFBRs introduced by the use of sodium coolant in an LMFBR, it does not directly follow that the frequency of core degradation due to LOHS events in PWRs is directly transferrable to LMFBRs.

Q.12: How did Staff treat the contribution of pipe rupture failure as a contributor to the core disruptive frequency?

A.12: The frequency of large pipe breaks (loss-of-coolant accidents, or "LOCAs") is pivotal to an assessment of the risk of accidents at CRBR or a reactor of the general size and type. A large pipe break in the cold leg (and perhaps the hot leg, as well) would likely lead to core disruption and serious offsite consequences. It is an important determinant in whether the CRBR site is suitable. Staff states:

> Because of the high boiling point of sodium, the CRBRP primary coolant system would

> operate at significantly lower pressures than LWR primary coolant systems. This reduces the frequency of large ruptures in the primary coolant system. To further ensure that large breaks cannot occur and cause core damage, implementation of preservice and inservice inspection of the primary coolant boundary and a leak detection system will be required. In addition, a guard vessel will be included to prevent unacceptable leakage from large portions of the primary coolant system. For these reasons LOCAs are not considered credible (i.e., design-basis) events at CRBRP. The frequency assumed for LOHS adequately bounds the LOCA contributions to core disruption frequency.

(FSFES, p. J.4, emphasis supplied.) When asked to identify every document relied upon by Staff for its conclusion above that "LOCAs are not considered credible ... events at CRBRP," Staff stated:

> The cumulative knowledge of the Staff and its consultants rather than a specific document were relied upon by the Staff for its conclusions in Appendix J regarding whether LOCAs are DBAs for CRBR. This issue was also discussed in the SSR and the Staff's prefiled testimony for the site suitability hearings.

(Staff Response to Interrogatory 33, 27th Set, Oct. 1, 1982, p. 58.) I take this answer to mean that Staff has no documentation or written analysis demonstrating that a LOCA is a low probability event for the CRBR.

In the 1982 SSR, Staff stated:

> It is the staff's opinion, based on the following considerations, that the heat transport system can be designed for a high level of integrity and for continued assurance of this integrity throughout the operating history of the plant. The specifications include stringent

nondestructive examination requirements. The material is characterized by high fracture toughness and corresponding large critical flaw size, a negligible growth rate of postulated defects and the probability of throughwall growth rather than elongation of defects. The system has low stored energy and is monitored by sensitive leak detection instruments. The staff preliminary conclusion is that double ended rupture of the CRBRP primary cold leg piping (an event that could potentially lead to a CDA unless otherwise mitigated) need not be considered a design basis event. This conclusion is conditioned on an acceptable preservice and inservice inspection program, a material surveillance program, continued research and development verifying material degradation processes, and verification of leak detection system performance. The staff considers it feasible to implement programs to satisfy these requirements. The staff intends to continue its review of the sodium cold leg piping to insure that the issues are resolved properly.

Because of its higher operating temperature, the same conclusions have not yet been reached concerning the hot leg piping (995° vs 730° F). The staff has studies underway to evaluate the potential for and consequences of hot leg piping ruptures. Preliminary results obtained so far indicate that this event has more benign consequences with respect to core thermal conditions than the cold leg rupture. For example, a hot leg pipe rupture followed by a scram and a pump trip and normal flow coastdown does not appear to lead to boiling in the core. Analyses of this event are continuing and the results will be factored into any future requirements to assure that hot leg pipe ruptures, like the cold leg case, need not be considered as events that would lead to a CDA.

(1982 SSR, pp. II-8 to II-9.)

Q.13: Do you agree with Staff's assessment, as stated above, of the pipe rupture probability, and, if not, what is the basis for your disagreement?

A.13: I disagree with the Staff assessment. In this regard, it is extremely instructive to compare Staff's analysis with the analyses conducted by D. O. Harris of the Palo Alto office of Science Applications, Inc. (SAI), for the CRBR Project office in the 1977-78 period. SAI was a consultant to the CRBR Project in the development and application of the fault tree/event tree methodology for assessing the reliability of CRBR systems as published in CRBRP-1, March 1977, and continued work for DOE on a variety of CRBR risk assessment issues through early 1979 and perhaps beyond. Staff consultant Rumble is a Vice President of SAI at the same Palo Alto office and has stated to me that he relied in part on CRBRP-1 for his assessment of the core degradation frequency which appears in Appendix J of the DSFES (and therefore the FSFES).

I have not been permitted to address that work in this hearing because, of course, it involves the "details" of the CRBR design. Only the most general conclusions have been presented in Appendix J.

In what appears to be a final risk assessment task report, obtained by NRDC under the Freedom of Information Act, D.O. Harris of the SAI Palo Alto office summarized the result of SAI's assessment of the CRBR pipe rupture probability (Harris, D.O., "Relative Pipe Rupture Probability for the Primary Heat Transport System of CRBRP," Nov. 13, 1978, attached as Exhibit 3 to this testimony).

Harris's analysis appears to be based on the assumption that the primary large pipe failure mechanism is fatigue crack growth due to cyclic stress imposed on defects introduced prior to service, hence other potential sources of failure were not considered. In this respect, Harris's analysis appears similar to that conducted in CRBRP-1 (Vol. 2, App. III, p. III-112). In the Harris analysis, calculated relative probability of pipe rupture in CRBR compared to that of PWRs was primarily a function of

a) probability of having a defect, which in turn was a function of the number and characteristics of the weld joints, Because the appropriate normalization was not known, separate calculations were made using weld volume, weld area, and weld length as the basis of normalization.

b) the initial crack size and depth distribution. Because the appropriate crack distribution was not known, separate calculations were made using four crack distribution expressions.

The differences between Staff's assertions and the SAI anlysis are important. Staff's conclusion that the CRBR cold leg pipe break is incredible (i.e., beyond the design basis) is based in part on the fact that there will be preservice and inservice inspection programs. Such programs have been in place for light water reactors for some time. The SAI analysis assumed equivalent effectiveness for the inspection programs for both CRBR and PWR in each calculation of the relative probability of pipe break failure of the two. This is the approriate way to treat the subject. Staff offers no evidence that any relative difference in the CRBR and PWR surveillance programs would have a significant effect on the crack distributions in CRBR piping relative to that in PWRs.

SAI found that "[w]ith the present state of knowledge, it is not possible to ascertain the controlling parameters" that govern the relative CRBR/PWR pipe break frequency. SAI found a wide range of values varying from 0.0186 to 11.62 (i.e., three orders of magnitude) in the ratio of CRBR pipe failure to PWR pipe failure depending on the assumptions made. In fully 13 out of 36 cases (36%) analyzed, the probability of CRBR pipe failure exceeded the probability of PWR pipe failure. Furthermore, the probability of PWR failure was found to be strongly design dependent, varying by as much as a factor of 14 among the three PWRs analyzed.

In conclusion, the Staff analysis of the pipe break probability is nothing more than a series of unsupported assumptions that appear to be in conflict with a more rigorous CRBR-specific analysis. The SAI analysis does not support the conclusion that a LOCA is "incredible" for the CRBR. Moreover, as evidenced by the SAI analysis, i.e., the lack of understanding of the controlling factors, the fact that the CRBR pipe break frequency may be as much as 12 times higher than that in a PWR, and the fact that the frequency is a strong function of the number and characteristics of the pipe welds, which are design dependent, the Staff conclusion that a cold (or hot) leg pipe rupture is not credible in a reactor of the general size and type of CRBR is not substantiated by rigorous analysis. It should be rejected.

Q.14: Do you agree with Staff's analysis of common mode failures?

A.14: The one sentence devoted to common cause failure hardly qualifies as "an analysis." LOHS failures due to common causes are but one manifestation of a larger class of failures that fall under the general category of systems interaction (SI). Systems interaction is presently the subject of two unresolved safety issues (USIs) -- namely A-17, "Systems Interaction in Nuclear Power Plants," and

A-47, "Safety Implications of Control Systems." The NRC has sponsored four separate evaluations of systems interaction in an attempt to develop an acceptable methodology for reviewing final designs for adverse systems interactions. These four studies are:

1. NUREG/CR-1321, "Final Report -- Phase I Systems Interaction Methodology Applications Program," G. Boyd, et al., Sandia National Laboratories, April 1980.

2. NUREG/CR-1896, "Review of Systems Interaction Methodologies," P. Cybulskis, et al., Battelle Columbus Laboratories, January 1981.

3. NUREG/CR-1859, "Systems Interaction: State-of-the-Art Review and Methods Evaluation," J.J. Lim, et al., Lawrence Livermore Laboratory, January 1981.

4. NUREG/CR-1901, "Review and Evaluation of System Interactions Methods," A.J. Buslik, et al., Brookhaven National Laboratory, April 1981.

The NRC Staff's evaluation of these four reports is summarized in the periodic "TMI Action Plan Tracking System Report" as follows:

> State-of-the-art review concluded that no single method presently exists in a form that can be used to perform an adequate review for adverse SI.

Thus, it can be fairly concluded that an adequate systems interaction review of CRBR could not have been conducted. Moreover, such a review requires a final design, which is not yet available for CRBR. It should be noted that three of the SI reviews above attempted unsuccessfully to evaluate SI in actual past events involving SI, including the Browns Ferry fire in 1975, the TMI-2 accident in 1979, the Browns Ferry partial scram failure in 1980, the pressurizer relief valve failure at Beznau in 1974, the temporary loss of decay heat removal at Davis-Besse in 1980, the loss of DC control power and diesel generator fire at Zion in 1976, and the Crystal River LOCA in 1980.

In addition, common mode failures and other forms of systems interaction involve more than just hardware failures. Also involved are external events (such as seismic events and hurricanes), human error (including errors of omission and commission, and including not only operations but design, fabrication, installation, maintenance, and testing), and design flaws. The design of the control room and any auxiliary control panels or remote shutdown locations, and actual operating, emergency, maintenance, and test procedures can also impact on systems interactions.

In sum, the effect of potential common mode failures on CRBR accident probabilities involves complex issues that the technical community has been wrestling with for years, thus far without notable success. There is no substantive basis for Staff's broad-brush assertion that "[t]he foregoing estimates of frequencies and risk associated with CRBR have included allowances for uncertainties. For example, unavailability estimates for shutdown and heat removal systems have been set high enough to include allowances for potential common cause failures." (Appendix J, p. J-22.)

Q.15: In estimating the quantitative probability of CRBR accidents, can credit be assigned for an "effective reliability program"?

A.15: In my opinion, it is not possible to assign any particular value to the level of "reliability" to be achieved. No CRBR-specific program has been presented by Staff; no precedent is cited for an "effective reliability program" for any other plant and no criteria are presented.

Finally, such assertions about the achievability of high reliability must be taken in the context of the most recent construction and design experience. This body of experience includes widespread problems at Diablo Canyon, Zimmer, and Midland. This experience is scarcely cause for confidence.

For all the reasons given above, I conclude that the NRC Staff's estimate of the frequency of core degradation due to LOHS events is optimistic, unsupported by rigorous analysis, and fails to properly account for uncertainties.

Q.16: Turning now to other contributors to the probability of core disruption, what assumption did the Staff make with regard to the probability of simultaneous failure of both reactor shutdown systems?

A.16: The Staff assured that "there are sufficient inherent redundancy, diversity, and independence in the overall shutdown system designs to expect an unavailability of less than 10^{-5} per demand," and concluded that "the combined frequency of degraded core accidents initiated by ULOF and UTOP events is less than 10^{-4} per reactor" (FSFES, p. J-4,5).

Q.17: What is the basis for the Staff estimate?

A.17: Beyond the explanation on pages J-4,5 of the FSFES, Staff claimed the value of 10^{-4} per year was a bounding value based primarily on LWR experience as published in NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors." In Vol. 1, Section 4.3 of NUREG-0460, an estimate of $2x10^{-4}$ per year for the frequency of ATWS for typical LWRs was given. Staff also stated, "Because the [CRBR shutdown systems] design and the reliability program are not final they have not been definitive in making the reliability estimate." (Staff Response to Interrogatories 36, 37, 38, 27th Set, Oct. 1, 1982, p. 60.)

Staff Witness Morris claimed that Mr. Rumble of SAI may have had a different basis for arriving at the value of 10^{-4} per year (Deposition of Staff Witness Morris, Oct. 12, 1982, p. 43).

Staff Witness Rumble said the basis for his estimate of the scram reliability of 10^{-5}/demand at DSFES, p. J-4, was based primarily on NUREG-0460; however, several other studies were mentioned as well. Mr. Rumble stated he was not familiar with the Commission's ATWS Policy Statement. (Edward Rumble, private communication, July 27, 1982, as recorded in Memo to files of T.B. Cochran, July 27, 1982.)

Q.18: Do you agree with the Staff conclusion that 10^{-4} per year is a conservative "upper bound" frequency of degraded core accidents initiated by ULOF and UTOP events in CRBR and, if not, what is the basis for your disagreement?

A.18: I do not agree. I believe 10^{-3} per year would be a conservative upper bound based on the Commission's LWR analysis in the Commission's Proposed ATWS rule for LWRs (46 Fed. Reg. 57521, Nov. 24, 1981)(see Tr. 2845, Cochran). While 10^{-4}/year might ultimately be shown to be appropriate, in light of the current absence of the detailed CRBR failure mode and effects analysis for the shutdown systems and consideration of effects of common mode failure, including, for example, seismic induced scram failures, there is at this time no basis for selecting a value larger than 10^{-3} per year.

Q.19: What assumptions did Staff make with regard to the probability of core degradation as a consequence of fuel failure propagation?

A.19: Staff assumed that "the CRBR fuel design will be required to have an inherent capability to prevent rapid propagation of fuel failure from local faults" (FSFES, p. J-4) and that the frequencies attributed to LOHS, UTOP, and ULOF events adequately bound the contribution to core disruption frequency from fuel failure propagation (FSFES, p. J-5).

Q.20: Has Staff provided adequate justification for this assertion, and what is the basis for your conclusion.

A.20: I do not believe there is an adequate basis for this conclusion. Staff has not developed the specific requirements or any associated criteria or confirmatory programs to prevent rapid propagation (details of the systems to prevent propagation of fuel failure are not final at this time), and Staff could cite no documentation for the conclusion that the core disruption frequency due to fuel failure propagation is bounded by 10^{-4} per year (Response to Interrogatory 39, 27th Set, Oct. 1, 1982, pp.

62-63).

Q.21: What assumption did Staff make with regard to the conditional frequency that a CDA once initiated would be energetic?

A.21: Staff developed four categories of primary system failure as a function of the energy associated with disruption (FSFES, p. J-5,6) and assigned a probability of primary system failure by excessive mechanical and/or thermal loads resulting in continuous open venting into the upper containment through failed seals (Category IV) of approximately 0.1 per CDA (FSFES, p. J-6).

Q.22: What basis did Staff give for this assumption?

A.22: In response to interrogatories asking for all documents relied on to support this conclusion, Staff claimed that this estimate was based on "the Staff's general knowledge of and experience with the extensive research on the phenomena that may occur in a core disruptive accident ...", but refused to cite any documents. (Staff Response to Interrogatory 43, 27th Set, Oct. 1, 1982, pp. 66-67.)

Q.23: Do you have any basis for disagreeing with Staff estimate?

A.23: There is inadequate documentation to support the Staff's estimate, which may be correct, incorrect, conservative, or nonconservative.

Q.24: What assumptions did the NRC Staff make regarding containment integrity in its analysis of CDAs?

A.24: Staff assumes that mitigating systems, principally the containment annulus cooling and vent/purge systems, will have an unavailability of less than or equal to 1 in 100 per demand. Staff also assumes that the unavailability of containment isolation will be equal to or less than 1 in 100 per demand. (FSFES, pp. J-6, -7.)

Q.25: Do you agree with these estimates and, if not, why not?

A.25: If Staff is correct that loss of offsite and onsite AC power dominates the failure probability for LOHS events, such a failure could also cause the failure of the mitigating systems. Staff has not accounted for this common failure mode.

Staff Witness Rumble stated that the basis for the 10^{-2} per demand for containment failure was based on estimates of LWR containment failure of $3x10^{-3}$ (Edward Rumble, private telephone communication, July 27, 1982, as summarized in Memo to Files of T.B. Cochran, July 27, 1982). As noted in the Union of Concerned Scientists' comments on the DSFES (letter from Steven C. Sholly to Paul Check, 13 Sept. 1982; FSFES, p. N-50), the operating history of PWRs and BWRs in the United States does not support the assumed unavailability result of 10^{-2} per demand. A review of actual experience through 1980 was reported in Nuclear Safety (Michael B. Weinstein, "Primary Containment Leakage Integrity: Availability and Review of Failure Experience," Nuclear Safety, Vol. 21, No. 5, September-October 1980) and concluded that the overall availability of containment integrity was about 0.85 (i.e., an unavailability of 15 in 100 per demand). This experience base would dramatically affect the Staff's risk analysis of CRBR. Using LWR experience would appear to increase the estimate for contaiment failure by a factor of 15. Even if the value for PWRs alone is used, the result is only 0.96 (i.e., 4 in 100 per demand unavailability factor). Obviously, if a Category IV CDA (as discussed by Staff) occurs with a breach in containment integrity, a very large release to the environment will occur. Use of actual experience is certainly to be preferred as contrasted with the very soft results obtained from the Staff's "analysis." It has not been shown that there are substantial differences between CRBR and the LWRs that form the present experience base.

In addition, it should be noted that the assumption of the failure of the mitigating systems discussed above (the containment annulus cooling and vent/purge systems) will also dramatically affect source term assumptions for the CRBR plant. Such failures will also increase the failure probability of the primary containment since lack of annulus cooling will cause a more rapid pressure rise and an earlier failure of the primary containment. This allows less time for natural processes to operate to reduce the airborne source term in the containment, and the postulated failure of the vent/purge system will also increase the source term for containment release substantially, especially for particulates and aerosols.

Staff's analysis is inadequate in its failure to address the points noted above and the concomitant large uncertainties inherent in the Staff's assumptions.

Q.26: Turning now to the estimates of the consequences in death and injury of CRBR accidents greater than the design basis, are the Staff's estimates presented in Appendix J likely to be accurate? Explain your answer.

A.26: No, and there are several reasons. First, Staff's assumed radioactivity source terms are not supported by analysis or documentation. When asked the basis for Staff's estimate of the head release fractions selected in Table

J.3 at p. J-10, including all analytical calculations and documentation, Staff stated:

> The head release fractions (Table J.3) were selected on the basis of judgement from consideration of general LMFBR research of energetic CDAs involving a bubble of vaporized fuel material rising against the reactor vessel head, giving consideration also to the relative volatilities of different types of fission products and other materials. The selections were therefore not based on a set of analytical calculations or on any specific documents.

(Staff Response to Interrogatory 53, 27th Set, Oct. 1, 1982, p. 77.)

The release fractions associated with CDAs are highly design dependent. The Staff "judgements," based on no analysis or documentation, represent speculations, and the uncertainties in some of the estimates, e.g., Pu release under Category IV, could be at least a factor of 3.

Second, the CRAC model utilized by Staff assumes the $LD_{50/60}$ (lethal dose to 50% of the exposed population within 60 days) is 510 rads. In my opinion, this assumption is unrealistic. This dose-response level is associated with a dose-response curve depicted graphically at page 9-4 of Appendix VI of WASH-1400. This dose-response curve, however, assumes that the victims receive "supportive treatment," which includes barrier nursing, copious use of antibiotics, massive transfusions, reverse isolation, and other special sterile procedures. WASH-

1400 estimated that the entire medical capability of the United States could provide such treatment to no more than 2,500-5,000 persons. WASH-1400 failed to address, however, how the victims of the highest exposures would be identified when there will be many others who will be suffering symptoms of radiation sickness (such as prodromal vomiting) from lesser exposures.

There is considerable controversy over the use of the 510 rads $LD_{50/60}$. The Risk Assessment Review Group (NUREG/CR-0040, "Risk Assessment Review group Report to the U.S. Nuclear Regulatory Commission," Harold W. Lewis, Chairman, September 1978) concluded that scientific opinion supports a range from 400-600 rads. This range could cause a factor of two change either way in the number of early fatalities. Moreover, the Risk Assessment Review Group concluded with regard to supportive treatment that "the ability to carry out such intervention has not only not been demonstrated, but isn't even well planned at this time" (NUREG/CR-0040, p. 19). Changing the $LD_{50/60}$ from 510 rads for "supportive treatment" to the level of "minimal treatment," i.e., 340 rads, could increase the number of fatalities by a factor of two to four (WASH-1400, Appendix VI, p. 13-50; NUREG-0340, pp. 26-28).

Other groups have used more realistic dose-response relationships which are closer to the "minimal treatment" curve used in WASH-1400. The California underground siting study used an $LD_{50/60}$ for minimal treatment of 286 rads and for supportive treatment of 429 rads (Subcommittee on Energy and the Environment, House Committee on Interior and Insular Affairs, "Reactor Safety Study Review," Serial No. 96-3, 1979, p. 366, attachment to letter dated 21 February 1979, from Bryce W. Johnson, Peter R. Davis, and Long Lee to Hon. Morris Udall, p. D-7). In addition, the "Accident Evaluation Code" (AEC) used to calculate health effects in CRBRP-1 utilizes an $LD_{50/60}$ of 350 rems (SAI-078-78-PA, Z.T. Mendoza and R.L. Ritzman, "Final Report on Comparative Calculations for the AEC and CRAC Risk Assessment Codes," Science Applications, Inc., December 1978, p. 3-6 and 3-8).

Third, the CRAC code contains several "hidden" assumptions regarding the cancer risk estimator for latent cancers, including an assumption that the cancer risk at low dose is a function of dose rate. The net effect of these assumptions appears to be 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 135×10^{-6} potential cancer deaths per person-rem, which Staff claims to use for estimating offsite health effects (FSFES, p. 5-13). Furthermore, a number of experts, including Radford, Morgan, Gofman, Stewart, Mancuso, Kneale, and Tamplin, believe the Staff cancer risk estimator, $135/10^6$ person-rem, is low, or probably low. Their own estimates of the cancer risk vary, but range from a factor of 3 (Radford, Edward, Science 213, 602 (7 August 1981), to a factor of 7 (Morgan) to a factor of 28 (Gofman, John W., Radiation and Human Health (Sierra Club Books, San Francisco, 1981), p. 305) times greater than the Staff's estimate of $135/10^6$ person-rem for fatal cancers due to whold body low-LET exposure.

Fourth, the source terms used by the NRC Staff in the CRBR accident consequence calculations appear to ignore any possible common cause failure of the containment annulus cooling and/or filtered venting systems. Certainly both of these systems are dependent upon offsite and onsite power supplies, and both will fail if all power is lost. On this basis, as noted previously, it makes little sense to largely ignore common cause failures involving these systems, as Staff has done. If the containment annulus cooling system fails, this will shorten the time between initiation of a CDA and failure of the primary containment. This affects decay of radionuclides that make up the source term and reduces the time available for natural processes such as gravitational settling and aerosol agglomeration to reduce the source term. Failure of the filtered venting system shortens the time between primary containment failure and secondary containment failure and also increases the source term when the containment fails. In particular, the source term for particulates and radioiodines will be greater if these systems fail. This scenario will result in a larger source term for release to the environment and will result in more serious consequences than predicted by the NRC Staff analysis.

Another consequence of assumption of the containment annulus cooling and filtered venting systems is a greater release of Lanthanide group radionuclides, including Pu-239. These long-lived radionuclides will certainly have an impact on cancer fatalities and on land contamination (and related interdiction criteria).

Q.27: What is Staff's position regarding the potential for a nuclear explosion in the CRBR?

A.27: In comments on the DSFES, Ohio Citizens for Responsible Energy (OCRE) asserted that "LMFBRs can suffer criticality accidents that can cause <u>nuclear</u> explosions as shown by <u>The Accident Hazards of Nuclear Power Plants</u> by Dr. Richard E. Webb" (FSFES, p. N-10).

Q.28: Do you agree with Staff's position? Explain your answer.

A.28: No. Staff is incorrect in this regard as evidence by Staff's and Applicants' own characterizations of CDAs as explosions. In testimony before the Senate Subcommittee on Nuclear Regulation of the Committee on Environment and Public Works, (attached as Exhibit 3), DOE and NRC Staff witnesses discussed environmental and safety matters related to the CRBR, including "hypothetical core disruptive accidents (HCDAs)," "core meltdowns and energetic disassembly," and design basis accidents. During the course of this testimony the following exchange took place between Senator Bumpers and Edson G. Case, then Acting Director, Office of Nuclear Regulation at the NRC:

> Senator Bumpers: May I ask one question? What is an energetic disassembly? Is that an explosion?
> Mr. Case: In layman's terms, it would be called an explosion. Yes sir. (Exhibit 1, p. 19)

Later in the same hearings the following exchange took place between Senator Bumpers and Eric S. Beckjord, Director of the Division of Reactor Development and Demonstration at ERDA.

> Senator Bumpers: Mr. Beckjord, what are the probabilities by ERDA's estimates of an explosion occurring in a breeder reactor plant?
>
> Mr. Beckjord: That would be the same order, 10^{-8} per reactor year. I might add that one of the margins that is to be included in this plant design is the capability to withstand a very sharp explosion. The words "energetic disassembly" came up earlier. Maybe that is overly technical, but we hva been in discussions with the Nuclear Regulatory Commission on the amount of energy, the amount of explosive force that must be

> accomodated within the structure. That matter is not settled yet. (Exhibit 3, p. 29).

These are not isolated references. The energetic disassembly of a fast breeder reactor is commonly referred to as an "explosive disassembly "[see, e.g., Lee J.C. and Pigford, Thomas," Explosive Disassembly of Fast Reactors, "Nuclear Science and Engineering 48, 28-44 (1972)] or "a small nuclear explosion "Hicks, E.P. and Menzies, D.C., Proceedings of the Conference on Safety, Fuels, and Core Design in Large Fast Power Reactors," Oct. 11-14, 1965, ANL-7120, pp. 654-670], a "low-efficiency nuclear explosion" [Stratton, W.R., and Engle, L.B., "Reactor Power Excursion Studies," "Engineering of Fast Reactors for Safe and Reliable Operation" (1973 Karlsruhe Conference), pp. 1331-1551].

There is no universally accepted definition of the word "explosion." The Webster's Seventh New Collegiate Dictionary defines "explosion" as "a large-scale, rapid and spectacular expansion, outbreak, or other upheaval." Cook defines an "explosive" as "any substance or device which will produce, upon release of its potential energy, a sudden outburst of gas, thereby exerting high pressures on its surrounding" [Melvin A. Cook, The Science of High Explosives (Robert E. Krieger Publ. Co., Huntington, N.Y.) 1971, p.1] Cook groups explosives under three fundamental types, mechanical, chemical and atomic (or nuclear). Johansson C.H. and P.A. Persson in Detonics of High Explosives (Academic Press, London, 1970) state (at p.6):

> Explosion is basically a rapid expansion of matter into a volume much greater than its original one. The word explosion thus includes the effects following or including rapid combustion or detonation, as well as purely physical processes as to bursting of a cylinder of compressed gas. We have chosen not to limit this rather useful wide definition of the word.

By these definitions an energetic disassembly of an LMFBR Core would constitute an explosion. It would not constitute a detonation which is a specific type of exothermic reaction that is always associated with a shock wave. If, as some authors prefer, an explosion is given a more limited definition such as to require the production of a shock wave, then most energetic disassemblies of LMFBR cores would not fit that definition.

A nuclear explosion is an explosion in which most or all of the explosive energy is derived from nuclear processes, either fission or fusion, or a combination of both.* [See generally, Samuel Glasstone, The Effects of Nuclear Weapons, 1962 Ed. ¶ 1.10]. Thus, an explosion in an LMFBR, that is an energetic disassembly following a prompt critical excursion, would constitute a nuclear

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* Fusion does not apply to the LMFBR for reasons that are obvious.

explosion as opposed to a chemical or mechanical explosion.

In response to a series of questions by Judge Linenberger in earlier testimony, I characterized a nuclear explosion as requiring a sufficient rate of energy deposition to result in the generation of a shock wave. Upon reflection, I do not believe this is the preferred definition. In any case, my previous testimony at Tr. 2777, 2779, 2785 and 2789 contains an error in inferring that the energetic disassembly of a fast reactor would result in the production of shock waves.

For the disassembly to be sufficiently energetic for the mechanical loading to challenge the containment, the nuclear excursion in a large Fast Reactor such as CRBR would have to be characterized by a rapid reactivity insertion and the reactivity exceed prompt critical. This will result in a rapid introduction of energy from the nuclear process, a rapid increase in rector power, elevated fuel temperature and vapor pressure formation. In such an event the core will begin to expand.*

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* Core expansion and fuel motion which reduces the material density will produce a negative reactivity feedback. Only a small expansion of the core is required to produce a larage disassembly reactivity. The reactor rapidly becomes sufficiently subcritical that any continued external reactivity insertion mechanism has no appreciable bering on the ultimate consequences. This marks the conclusion of the neutronic excursion and the disassembly of the accident [Waltar, Alan E. (cont. next page)

An energetic disassembly, or nuclear explosion, in an LMFBR differs from a chemical explosion following detonation of a high-explosive in terms of the pressure-time characteristics of the two. Generally mechanical damage from an explosion or pressure transient can be caused by either a shock wave, which is transmitted rapidly to a structure, or the more slowly expanding bubble of reaction products or vaporized material or both. Pressures in a chemical high explosive detonation build up on a microsecond time scale. As a consequence, much of the damage potential of a chemical high explosive to immediate surrounding structures is likely to come from blast or shock wave effects. In an explosion in an LMFBR the build up is over a millisecond time scale and shock waves are generally not produced. Long-term bubble expansion (at least in the absence of a vapor explosion driven by a molten fuel-coolant interaction) would be the predominant damage mode for the slower time scale pressure build up associated with an LMFBR nuclear excursion. (See, generally, Walters and Reynolds, ibid., p. 664.)

Q.29: What is your overall conclusion regarding the Staff analysis in Appendix J?

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and Albert B. Reynolds, Fast Breeder Reactors (Pergamon Press, N.Y.) 1981, p. 619].

A.29: According to Staff Witness Rumble, Appendix J was done hurriedly because of the severe time constraints (Edward Rumble, private telephone conversation, July 27, 1982, as summarized in T.B. Cochran Memo to Files dated July 27, 1982). This is apparent from the depth of the analysis presented.

Staff can correctly point to several conservative assumptions made in Staff's analysis. Nevertheless, Staff's analysis of the CRBR accident probabilities and consequences is inadequate and unreliable. Staff claims "the uncertainty bounds 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" (FSFES, p. J-24) As noted previously, the uncertainties in the probability estimates are larger than those of WASH-1400 and the Commission's previous conclusion -- that the numerical estimates of accident probabilities in WASH-1400 are unreliable -- applies equally to the Staff Appendix J analysis. Furthermore, the consequences (i.e., health risks) of "Class 9" accidents at CRBR as estimated by the Staff are based on a series of assumptions with large associated uncertainties. One can find uncertainties of at least two orders of magnitude and consequences. When these uncertainties are considered together (compounded), I believe they result in an uncertainty of at least two or more orders of magnitude in Staff's estimate of the acute and delayed health effects. With these large uncertainties in the probabilities and consequences, Staff's analysis in Appendix J does not support Staff's conclusions in the FSFES, Section J.1.3, at J-25.

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

In the Matter of	)
	)
UNITED STATES DEPARTMENT OF ENERGY	)
PROJECT MANAGMENT CORPORATION	)
TENNESSEE VALLEY AUTHORITY	)
	)
(Clinch River Breeder Reactor Plant)	)

AFFIDAVIT OF DR. THOMAS B. COCHRAN

City of Washington)
) ss:
District of Columbia)

DR. THOMAS B. COCHRAN hereby deposes and says:

The foregoing testimony prepared by me and dated November 12, 1982, is true and correct to the best of my knowledge and belief.

Dr. Thomas B. Cochran

Signed and sworn to before me
this 12th day of November 1982.

Donna Marie Shelcer
Notary Public

My Commission Expires July 31, 1987

CLINCH RIVER BREEDER REACTOR

HEARING

BEFORE THE

SUBCOMMITTEE ON NUCLEAR REGULATION

OF THE

COMMITTEE ON

ENVIRONMENT AND PUBLIC WORKS

UNITED STATES SENATE

NINETY-FIFTH CONGRESS

FIRST SESSION

JULY 11, 1977

SERIAL NO. 95-H27

Printed for the use of the Committee on Environment and Public Works

Exhibit 3 to Cochran
Testimony, Part IV
Docket No. 50-537

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In conclusion, the NRC staff found the proposed Clinch River site acceptable from an environmental and safety standpoint for the CRBR, assuming that the ERDA programmatic environmental statement was dispositive of the need for a demonstration-scale facility, including its timing and objectives. Naturally, if the findings in the programmatic statement in these critical areas were to change, we would be obligated to again review the environmental acceptability of the Clinch River site based on considerations relevant at that time.

Thank you, Mr. Chairman.

Senator HART. Thank you, Mr. Gossick. I would like to welcome Senator Bumpers, who is a member of the Energy Committee. He has indicated an interest in this subject and has been deeply involved in the general question of the project for the last number of weeks.

We are pleased to have him with us. Unless there is objection, I would like to go to the ERDA testimony and then have questions for both sets of panelists, if that is agreeable to everyone.

I would like to ask one procedural question, Mr. Gossick. That is, in your testimony you say that the NCR staff obtained from ERDA a copy of the Burns & Roe memorandum dated July 6, 1973. That is almost 4 years after the date of the memo. Can you explain to us, according to your own procedures for licensing, why a memorandum calling into serious question a project of serious counts as you described in your statement was not made available to the NRC in 4 years?

Mr. GOSSICK. Sir, as I understand the status of the document, it was an internal memorandum. It was not a part of the material filed by ERDA in the proceeding at that time.

Senator HART. Was it in ERDA's possession?

Mr. GOSSICK. I don't know, sir.

Mr. BECKJORD. No, sir. We received it about 2 weeks ago, after the document was released to the newspapers.

Senator HART. So, it was not in your possession during that period of time?

Mr. BECKJORD. No; it was not.

Senator MCCLURE. What is your customary procedure with respect to the internal memorandum of contracting agencies? Would you normally see this kind of a memorandum? What is the normal flow of that internal information with the contractor, with a regulatory agency or ERDA?

Mr. GOSSICK. Sir, I would point out in this case, of course, that ERDA constituted the applicant to NRC. I think the question is properly one that ERDA should address. We are not involved with contractors in a regulatory sense.

Senator MCCLURE. In the regulatory sense, then you would not ordinarily see the internal document of the applicant?

Mr. GOSSICK. No, sir, unless it would become a part of the——

Senator MCCLURE. Except those portions of their internal documents which they choose to present to the regulatory agency?

Mr. GOSSICK. Yes, sir.

Senator MCCLURE. In support of their application?

Mr. GOSSICK. Yes, sir.

Senator MCCLURE. If there are questions in the minds of what is now NRC, would it be customary for you to ask for internal documents?

Mr. GOSSICK. I know of no reason why we could not ask, depending upon the nature of our concern.

Senator MCCLURE. Have you done so?

Mr. GOSSICK. I am not aware of any incident in this particular case, Senator.

Senator MCCLURE. I wonder if the ERDA witness might also, Mr. Beckjord, perhaps you could indicate whether ERDA looks at applicants' internal memorandums.

Mr. BECKJORD. Senator, we see correspondence that is directed between the various participants in the project, as well as the correspondence that we receive directly. We participate in design review meetings. We ask questions to which there are responses. We feel that we see the important information through those methods of receiving it. As regards internal memorandums, if they are sent to us, we are aware of them. If they are not sent to us, we are not aware of them.

Senator BUMPERS. Mr. Chairman, I don't want to interrupt the proceedings, but I would like to ask a question. Due to the magnitude of the question raised in the internal memo about the suitability of the site which states that it is indeed the worst site ever selected for a nuclear powerplant, I am curious as to whether at that time or subsequent to that time, Burns & Roe called it to your attention that they thought it was the worst site ever selected.

Mr. BECKJORD. I am not aware of the use of these words, Senator. We certainly were aware that there were matters which had to be investigated with respect to the technical suitability of the site. I am going to cover that in my testimony.

Senator BUMPERS. But you don't have any direct correspondence indicating their concern about this site?

Mr. BECKJORD. There is considerable correspondence regarding the technical suitability of the site, Senator, work that was done at that time, particularly site borings, that type of information we were aware of. A complete evaluation of all that information was done before the final placement of the plant was decided upon, and considerable analysis was performed to support it.

Senator BUMPERS. I won't pursue this any further at this time.

Senator HART. I think it is obvious that the concern of the committee and many of us is whether the architect-engineer was saying one thing internally and another thing to the appropriate Government agencies. That is what we want to pursue. We would like to go forward with the ERDA testimony, Mr. Beckjord, accompanied by Mr. Lochlin Caffey, Director of the Clinch River project.

STATEMENT OF ERIC S. BECKJORD, ACCOMPANIED BY LOCHLIN CAFFEY

Mr. BECKJORD. Mr. Chairman and members of the committee, I appreciate this opportunity to discuss the environmental and safety matters related to the Clinch River Breeder Reactor Plant project which were raised in the July 6, 1973, internal Burns & Roe memorandum recently cited in the press.

With your permission, Mr. Chairman, I will submit my written testimony for the record and reduce the part that I give to you.

Senator HART. Without objection, that would be very agreeable to us.

Senator DOMENICI. Might I ask one clarifying question before he testifies? In the practical field of contracting, what does an internal memorandum mean? Who were they writing this to? What was the purpose of it? How does this occur in the day-to-day business of evaluating that kind of site?

Mr. BECKJORD. Senator, my understanding from the information available to us, which is the memorandum and the statement which Burns & Roe made to the press when this was released, is that this was an internal memorandum, the purpose of which was to advise the directors of Burns & Roe of the situation of the project with some recommendations regarding their subsequent business actions toward the project.

The was the purpose of the memo. As indicated, it was a private and internal memo. Evidently they did not intend to make that particular document available to the project.

Senator DOMENICI. It was their own assessment, directed at their people, as they proceeded to evaluate their job?

Mr. BECKJORD. At the time, there were evidently a number of important business decisions that the company intended to make. I think that is covered in that clarifying statement. The purpose of the memo was to address those decisions.

Senator DOMENICI. Thank you, Mr. Chairman.

Mr. BECKJORD. I reviewed the Burns & Roe memorandum in detail. My statements on it are based on information available to me.

The CRBRP project is a joint government-industry cooperative arrangement for demonstrating a liquid metal fast breeder reactor power plant as authorized by Congress on June 2, 1970—Public Law 91-273. The partners in this project are the Energy Research and Development Administration, Commonwealth Edison, Tennessee Valley Authority, and Project Management Corp.

The objectives of this project are to design, license, construct, test and operate an LMFBR demonstration plant. In May 1976, ERDA assumed full management control of the project with continued utility industry support and participation.

I have had the ERDA responsibility for this project since March 1976. During that time, project accomplishments have been good, with design now over 40 percent complete, all of the longlead equipment on order, and the final environmental statement and site suitability report issued by the Nuclear Regulatory Commission.

I have examined project records, reviewed the numerous reports and hearings concerning the project, and inquired extensively into project procedures and status, particularly in environmental, safety and related licensing matters. Generally, I can say that the project has also made good progress in these licensing areas during the past year, working toward its goal of a limited work authorization as required under the NEPA act of 1970, until the recent suspension of the environmental hearings in April. The environmental hearings suspension was requested by ERDA, pending a final decision on whether the project is to be terminated or continued.

I have reviewed the Burns & Roe memorandum in detail since it became available to me about 2 weeks ago. My statements on it are based on the information available to me as a result of research done in the interim. Some of the issues raised were speculative and others were founded on incomplete or incorrect information. Of the remaining issues, I found either that they have already resolved or that work toward proper resolution is underway in conjunction with licensing activities as required by NRC.

Comments on the specific issues raised by the Burns & Roe memorandum are as follows: I refer now to numbers in the original memorandum, in the summary section, page 2, item 5, and also page 3, item 5. The issue here is the suitability of the site and the associated costs of site development.

The plant site was selected following consideration of several possible alternative sites. In late 1971, the AEC appointed a Senior Utility Steering Committee and Senior Utility Technical Advisory Panel to assist them in selecting a utility partner to design, build and operate the demonstration plant. Proposals were submitted to the Steering Committee and AEC by groups of utilities interested in participating in the demonstration plant program.

There were in fact three sites that were considered. The Steering Committee found that the proposal from Commonwealth Edison and the Tennessee Valley Authority offered increased siting flexibility over the other proposals. This was the proposal that finally was accepted by the Steering Committee, and following that, by the AEC. I will not go over the details of the site comparisons that were made.

The soundness of that original decision was supported by the comprehensive and detailed site investigation program conducted during 1973, subsequent to the Burns & Roe memorandum. In contrast to the Burns & Roe apprehension, the site was actually found to be similar to others utilized for nuclear powerplants in the region and was demonstrated to be fully acceptable from all standpoints.

The Nuclear Regulatory Commission also confirmed the acceptability of this site based on their independent review and assessment as documented in the final environmental statement for the CRBRP issued in February 1977, and the site suitability report issued in March 1977. In the site suitability report, NRC concluded that the foundation conditions were generally good and there were no subsurface conditions expected which would preclude the suitability of the site or the construction of the proposed plant.

As the nuclear powerplant siting criteria have undergone very substantial evolution over the past several years, the continued acceptability of this site further reinforces the soundness of its selection.

With regard to the cost of preparing this site, any additional costs incurred for preparation of this site compared to a hypothetical "optimum" site will be small when considered in the context of the many other factors influencing site selection. For example, the cost of highways that are necessary to transport equipment, can be a major variable in the cost of site preparation and this could vary considerably from site to site.

I refer now to the issue of compliance with licensing requirements. The statement in the Burns & Roe memorandum, page 8, item 5, page 9, paragraph 1 and page 9, paragraph 3, concerning compliance with 10 CFR 50 requirements appear to be in direct conflict with the requirements established by the AEC for this project in material submitted to the Congress prior to authorization.

In the original program justification data arrangement for this project submitted to the JCAE on August 11, 1972, it was clearly stated that "all applicable laws and regulations, including those pertaining to AEC licensing and regulations, will be complied with."

This same requirement, updated to reflect the establishment of the independent NRC, is in the Revised Program Justification Data Arrangement No. 77–106, which covers the project at this time.

As to my comment on it, the minutes of the Project Steering Committee have been reviewed and no record was found to support the statement made by Burns & Roe concerning compliance with 10 CFR 50 requirements.

Senator BUMPERS. Did you talk to the man who wrote the memo?

Mr. BECKJORD. I have not had detailed conversations with Mr. Young concerning the memo. I concluded that that was not proper in view of this hearing to be held.

Senator HART. But you have had some talks with him, or some contact with him?

Mr. BECKJORD. Oh, yes, I have had contacts with Mr. Young because he is responsible for the project for Burns & Roe. I mean with regard to this specific memo, I have not had detailed discussions with him.

Senator HART. But you have had some discussions with him?

Mr. BECKJORD. I have had some discussions with him.

Senator HART. About the memo?

Mr. BECKJORD. The discussion concerned whether we wished to see the testimony which he planned to give. He indicated he would send a copy of the testimony.

Senator HART. But you didn't discuss the substance of the memo?

Mr. BECKJORD. Beyond a few comments, there was no detailed discussion of the substance of his testimony or the memo.

Senator HART. What was the nature of his comments?

Mr. BECKJORD. It concerned this passage regarding compliance with 10 CFR 50 requirements.

Senator HART. What was the nature of that discussion?

Mr. BECKJORD. I asked for clarification as to what was intended. He indicated that the clarification would be in his testimony.

Senator HART. He didn't go into it at that point?

Mr. BECKJORD. No.

Senator BUMPERS. Mr. Chairman, I don't want to interrupt his testimony further than necessary at this point, but I think this is very crucial. You say that—this is one of the most critical parts of the memo as far as I am concerned. You say you have talked to Messrs. Milton Shaw, Thomas Nemzek, Mr. Wagner of TVA, Mr. Wallace Behnke of Commonwealth Edison, Messrs. John Taylor and George Hardigg of Westinghouse, and each of them has assured you there was never either a policy or a practice of avoiding compliance with the AEC Division of Regulation licensing requirements. My question is did you ask him where he got that information and whether he based that information on the memo?

Mr. BECKJORD. Did I talk to Mr. Young?

Senator BUMPERS. You were saying here you have talked to everybody who might have told Burns & Roe that they would not have to comply with some of these basic safety requirements and that all of them say, you say each of them assured you, there never was a policy or a practice of avoiding compliance with the AEC Division of Regulation licensing requirements.

If you talked to the writer of the memo, did you ask him what caused him to put that in the memo?

Mr. BECKJORD. I did not ask him that question, Senator.

Senator BUMPERS. Thank you, Mr. Chairman.

Senator HART. Proceed.

Mr. BECKJORD. I will read this in its entirety.

The minutes of the Project Steering Committee have been reviewed and no record was found to support the statement made by Burns and Roe concerning compliance with 10 CFR 50 requirements. In addition, I have personally called a number of men who were leaders in the early days of the project. These are Messrs. Milton Shaw and Thomas Nemzek, former Directors of ERDA's Reactor Development Division; Mr. Wagner of TVA, Mr. Wallace Behnke of Commonwealth Edison, Messrs. John Taylor and George Hardigg of Westinghouse.

Each of them assured me there was never either a policy or a practice of avoiding compliance with the AEC Division of Regulation licensing requirements. It was, in fact, the policy to go through the entire safety and licensing process as part of the project objectives.

It was understood by the project leaders that modifications to some of the 10 CFR 50 general design criteria would need to be developed, simply because of the technical differences between light water reactors, for which the general design criteria were originally written, and the Clinch River breeder reactor, for which general design criteria were not yet written in 1973.

These modifications were developed within the licensing process and are consistent with the evolution of the licensing process for LMFBR's. It should be noted that much work and discussion was required to resolve the differences of technical opinion prior to the final issuance of CRBRP general design criteria by NRC on January 9, 1976.

The fact that there were significant differences of technical opinion during this effort, however, does not lead to the conclusion that the project was trying to avoid compliance with safety requirements. The safety requirements were properly established when NRC issued, and the project accepted, these criteria.

The objective of the design criteria and the net effect of the CRBRP licensing process is to make the CRBRP at least as safe as a light water reactor located at the same site. To suggest, as the Burns and Roe memorandum does, that there was an intent not to comply with licensing requirements or that the AEC desired to avoid including needed safety features because of cost considerations is simply not supported by the facts.

I can further testify that during my association with the project, the policy has been, is now, and will continue to be, to comply with the Nuclear Regulatory Commission's licensing requirements.

The three level defense-in-depth safety philosophy currently being used for design of LWR's was also adopted for CRBRP. This requires design measures to prevent accidents, to provide protection against either anticipated or unlikely faults that might occur, and beyond this, to provide appropriate engineered safety features in the design to safely accommodate extremely unlikely faults, if they somehow should occur, in order to protect the health and safety of the public.

Furthermore, ERDA and NRC have agreed that, for the CRBRP, it is prudent to include additional measures in design to further limit potential consequences to the health and safety of the public. Accordingly, the project has included margins, beyond the necessary design basis, in order to reduce the postulated consequences of hypothetical accidents involving core meltdown and energetic disassembly.

At the time of the Burns and Roe memorandum, there were ongoing discussions between RRD and DRL concerning whether hypothetical core disruptive accidents HCDA should be included in the design basis for LMFBR's. The resolution with DRL was that, to avoid schedule delay, two CRBRP designs would be submitted for concurrent review, one without and one with HCDA's in the design basis, the reference design and a parallel design.

In a May 1976 letter, the NRC agreed that HCDA's can and should be excluded from the design basis. Subsequently, the project withdrew the parallel design from further consideration by NRC, but it was mutually agreed that margins would be provided in the plant design in order to reduce the postulated consequences of such hypothetical accidents so that the CRBRP would be comparable to current LWR's.

Senator HART. Let me run through that in English so I understand what that means. It seems to me what happened here was in the discussions within ERDA and with the contractors, that it was decided, for purposes of determining the safety of the project, that there would be two hypotheticals, or there would be two critical paths followed, one which included the so-called hypothetical core disruptive accidents which, I assume, are core meltdowns and things of that sort, and one which did not.

Because in an effort to avoid what are called schedule delays—you took the two path method to avoid the delays. Later on in an agreement by May 1976 letters, the NRC agreed, I don't know with whom, that the path including the hypothetical core disruptive accidents, which is an interesting phrase in itself, would be excluded from what is called the design basis; presumably the basis upon which the decision to go forward would be made.

So the project withdrew the so-called parallel design, including the hypothetical core disruptive accident presumptions. Then you proceed, it was mutually agreed that margins would be provided in the design to reduce the postulated consequence of such hypothetical accidents.

What does that m an?

Mr. BECKJORD. It means this, Mr. Chairman, and I will try to explain it in English. A design basis accident is an accident which is assumed to happen and the course of the accident is evaluated. By the definition of design basis accident, what is meant is the particular part of the system or the plant in its entirety has to accommodate the consequences of that accident and control them with no adverse consequences within design limits.

What that means, design limit for example, I will attempt to explain in a simplified manner. In the case of a metal bar, let us say, if we were to accommodate an accident within the design limit for a metal bar, after the accident had occurred, there would be no deformation of the bar because design limit requires the design st--sses not be exceeded. So the bar might deform, but when the event was over, the bar would be elastic and it would return to its initial condition.

I can consider accidents which go beyond the design basis. In that case, I will evaluate that accident. If such an accident occurred that did go beyond the design basis, the bar might be deformed so that it would not return to its initial condition. That doesn't mean that anything has happened. It does not mean that there are adverse consequences. It simply means that the system is capable of accommodating one time occurrences far beyond, in many cases, the limit of the design.

So the question here is whether this HCDA should be a design basis accident and the entire system should accommodate it within design margins or whether going beyond design limits should be permitted with the accident contained in other ways. That is a one time event.

I believe that correctly summarizes the differences between a design basis accident and an accident which goes beyond a design basis accident.

Senator HART. A hypothetical core disruptive accident, that means a core meltdown?

Mr. BECKJORD. It means a core meltdown or it could mean a core disassembly through some means of sudden energy release which would cause it to disperse.

Senator HART. Is it safe to say that that phrase includes the worst possible things that could happen?

Mr. BECKJORD. I believe it does.

Senator HART. So in this case, in order to save time, it seems to me two decisions were made, that you would go on two paths in your planning, one which included these most serious accidents for design purposes and one which did not. That went on for awhile, it is a little unclear for how long.

Mr. BECKJORD. I think it was midsummer of 1974 until the May 6 letter, of 1976.

Senator HART. So almost 2 years, you went on a two path basis, one with the accidents, one without.

Mr. BECKJORD. Yes.

Senator HART. Then NRC agreed, with whom and for what reason it is unclear, that the planning path that included the most serious accidents can and should be excluded. Then the project withdrew the path with the most critical accidents included from further consideration by the NRC, but it was mutually agreed that margins would be provided.

What does that mean, margins would be provided?

Mr. BECKJORD. As when I was trying to come up with a simple example, margins would be provided to accommodate the consequences of a hypothetical core disruptive accident, but not as a matter of a design basis—that is, not to say with the bar returning to its initial original position.

Specifically, these margins have to do with the structure of the plant and the containment, the heat capacity in the base of the floor under the reactor, and in the ability for heat removal from the containment so that if this accident still occurred, it would reach an equilibrium point.

It is a very low probability accident, but nonetheless, these design margins would make it possible to control the accident.

Senator HART. There are two questions that come to mind. First of all, who defines the margins, specifically and, second, why not, for purposes of public safety, accept the design course that included the most serious accidents? Why go on the two-path method in the first place? Why not take the worst case basis for a design study?

Mr. CASE. Senator Hart, may I respond to that?

First, one should understand there are two aspects to this hypothetical core disruptive accident. First, you do everything you reasonably can do to prevent the accident. There has been no change with the NRC requirements with regard to that. In other words, we still require all the features necessary to prevent the occurrence of such an accident.

The other side of the coin is to assume, nevertheless, having all the features, the accident occurs anyway for some hypothetical reason. On that score, we took a course of action in between the so-called reference design and parallel design requiring the plant be designed to accommodate some of the effects of this accident, but not all of them. In other words, we continue to require that the containment system maintain its integrity for at least 24 hours following the occurrence of this same hypothetical core-disruptive accident.

The reason we don't require all of the other features of plant beyond that time is simply that we don't think it is necessary from a safety standpoint in view of the very low probability of the occurrence in the first place due to the features required to prevent the accident occurrence.

Senator HART. On the breeder reactor program, are the same standards used for the light water reactors in this regard, in both regards?

Mr. CASE. The standards here are more severe than those for the light water reactors. For the light water reactors, we require all the features to prevent such accidents. We do not require it to accommodate the accident, in the event it should occur. In this plant, we do.

Senator MCCLURE. Might I just ask this question? Some reference has been made to, in order to save time or to avoid schedule delay, that refers, if I understand it correctly, that refers only to the parallel design feature for a period of time and not to the ultimate decision. Am I correct?

Mr. CASE. That is not correct. That factor did not enter into our decision. Our decision was strictly based on our judgment that the risks of this reactor should be comparable to light water reactors.

Senator MCCLURE. As a matter of fact, you required a design beyond that required of the light water reactor?

Mr. CASE. Yes, but I want to make the record clear that light water reactors have some inherent features that this plant does not. In requiring features on this plant, our objective was to make the risks comparable.

Senator MCCLURE. So that there was no element of saving time or meeting a schedule that was motivating in your decision not to require the most severe accident containment?

Mr. CASE. That is correct, Senator.

Senator BUMPERS. May I ask one question? What is an energetic disassembly? Is that an explosion?

Mr. CASE. In layman's terms, it would be called an explosion. Yes, sir.

Senator MCCLURE. It sounded like some OSHA language.

Senator HART. Mr. Beckjord, proceed, please.

Mr. BECKJORD. All of the relevant CRBRP safety issues, including those raised by the Burns & Roe memorandum, are being properly and thoroughly analyzed during the course of the licensing process. Most of the issues have been resolved in a manner mutually acceptable to ERDA and NRC. Work is continuing on the remainder of these issues at this time. No unusually difficult problems in design have been identified.

To date, the project has made design changes estimated to ultimately cost $60 million in order to meet additional licensing requirements which have evolved during the interactions with NRC, and it is possible that other changes may yet be required. You may be assured, however, that we have always been, and are at present, dedicated to meeting all necessary licensing requirements.

Referring again to the Burns & Roe memorandum, on page 14, item 5, and on page 17, item C, there is an issue raised regarding project requests for special licensing variances. The CRBRP project has asked for no special licensing variances.

Consistent with one of the major CRBRP project objectives of demonstrating the licensability of the LMFBR concept, the CRBRP is being subjected to the identical licensing process by the NRC as would any commercial nuclear powerplant.

At the time of the Burns & Roe memorandum, the project was expecting to request an exemption to conduct certain site preparation activities prior to receipt of a construction permit, as was permitted by the AEC regulations under 10 CFR 50.12(b).

However, that procedure was changed, I believe, in anticipation of the establishment of NRC. That attempt was dropped and instead we began to pursue the limited work authorization, which is the end of the process required under the NEPA Act of 1970. The limited work authorization would permit us to begin site preparation activities.

When the CRBRP environmental hearing activity was suspended in April of this year, we were in the process of pursuing the request for a limited work authorization. My point is that, with the exception I have just expressed, there have been no requests for special variances.

Regarding other NRC requirements, the project will meet all of the applicable requirements. However, as already stated, some of the NRC requirements were formulated for LWR's and have either no applicability or only partial applicability to the CRBRP. In these cases, the project will meet the intent of the LWR requirements by developing modified or new requirements in cooperation with NRC; that is, 27 of the 56 general design criteria were modified, plutonium dose guidelines were developed, and new containment criteria were developed.

Referring again to the Burns & Roe memorandum, on page 17, item D, on page 17, item 7, on page 18, items A through D, and on page 18, the first full paragraph, the technical suitability of the plantsite is discussed. I won't read through all of those points.

These apprehensions of Burns & Roe about the site were based on 24 core borings at the proposed site, of which only 4 were in the immediate vicinity of the plant location. After a comprehensive and detailed site investigation program, the final plant location at the Clinch River site was proven to be sound.

This site investigation program included over 100 additional core borings, a test grouting program to confirm the homogeneity of the foundation stratum, detailed geophysical studies, and other extensive analyses and tests. All the points raised by the Burns & Roe memorandum were fully and thoroughly reviewed with NRC prior to their issuance of the final environmental statement and the site suitability report for the CRBRP. The NRC staff concluded that the foundation conditions are good and that the site is suitable for construction of the plant.

Referring to the Burns & Roe memorandum on page 22, item F, the issues presented are safety approaches and plant licensability. This comment on the licensing process was made at an early point in the plant design.

As has already been explained, one of the key objectives of this project has been to license this plant in the same manner as a commercial LWR plant. Many of the specific approaches and features which were ultimately incorporated into the design required extensive study, analysis, and development.

The problems identified in the Burns & Roe memorandum have each been addressed in the licensing process as the design has evolved. Either they have been resolved or appropriate work is underway to resolve them.

In conclusion, I wish to emphasize the following points: the goal of the CRBRP design has been to provide a plant which is at least as safe as an LWR located at the same site. Since the commencement of the project, it has been the policy to go through the entire licensing process and to comply with licensing requirements established by the AEC Division of Regulation and its heir, the Nuclear Regulatory Commission. All NRC licensing requirements are being fulfilled in the project implementation.

The internal Burns & Roe memorandum is over 4 years old. Some of the issues raised in it were speculative, and we have not found a basis for them. The remaining issues have each been properly addressed in our detailed design and site investigations and with the NRC in the licensing process. Each issue has been fully and completely resolved or appropriate work toward resolution is currently proceeding.

NRC has agreed that the comprehensive site investigation program has established that the site meets NRC requirements. Good progress was made by the project in the licensing area during the past year until the suspension of licensing hearings in April.

That concludes my statement, Mr. Chairman. I would be glad to answer any other questions that you have.

Senator HART. Thank you, Mr. Beckjord.

I think we will direct questions to both Mr. Gossick and Mr. Case as well as the earlier witnesses. One thing that concerns me, Mr. Gossick, about your statement is the tone and passive character of some of the sentences where you talk about the site and the project as being not inconsistent with NRC objectives and standards.

At very few points in your statement do you go out of your way to give extraordinary assurances to us for the American public about this project. For example, you say the staff review has been aimed at assuring that these concerns were resolved in a manner consistent with a safe facility design and operation.

That is a very carefully worded statement. You use words like "aimed at" and "matters consistent with." In matters of this sort, what the American people want, at least what I want, is something a little more than that, how safe these facilities are and that the way that the Clinch River project has been going is not inconsistent with other projects and things of that sort.

There is lacking, I think, a kind of positive note in your testimony that I think we would like. Is that a problem for you?

Mr. GOSSICK. Sir, I think I must address the point with regard to the site as we have concluded in my statement. We are convinced that the site is a satisfactory site. We have not finished the safety review, the other part of the review of the CRBR, Mr. Chairman. It is still undergoing staff review. It is a process not yet complete.

Therefore, we must not speculate about the outcome of that until the hearing is finished on the safety aspects and the staff action is completed. All I can say is it is going along as any other application, recognizing it is a first of its kind.

So there is no intent to indicate either pessimism or, for that matter, no particular grounds that I can cite at this point for saying that we are convinced that it will be a safe design. We have just not completed that process.

Senator HART. The tentative nature of your statement is attributable more to the fact that you are still in the process and not that you have lingering hesitancy itself.

Mr. GOSSICK. That is correct.

Senator HART. Specifically, in your testimony in connection with the Burns & Roe statement about it being one of the worst sites ever selected, you have the following language, "Reduction in accident risks achievable with remote location"—talking about the staff balance—"against the resulting costs and inability of the demonstration plant to accomplish its goals on a time frame compatible with the present timing goals of the LMFBR program."

What that says to me is there is a balancing of risk against cost and time. You resolve it slightly, at least in the direction of cost and time. If I am wrong, correct me. I want to quote in that connection the context from which that assurance came or that statement came, the final environmental statement dated February 1977 has the following sentence in it, or paragraph, that I will extract:

> Another measure of the relative differences among the sites was obtained by estimating the relative consequences in terms of overall population exposure out

to 50 miles. The radiological dosage at the alternative sites would be roughly a factor of 10 less than the Clinch River site by this measure.

I think the question is in this balancing: How much does the risk go up in order to keep cost and time down?

Mr. GOSSICK. Sir, I would like to ask Mr. Case to address the details of this. I would say at the outset, however, as I have already indicated, that the objectives of the CRBR program within the context of the overall LMFBR program and the ability to meet those objectives on a certain time scale have been stated by ERDA as required, and have been discussed with the Congress and the administration, and were taken into account in that balancing process.

Specifically how that was treated, I would like to ask Mr. Case to address.

Mr. CASE. First, risk is a product of probability times consequences. Your question really was what is——

Senator HART. Say that again.

Mr. CASE. Risk is a product of probability times consequences—the probability of an accident times consequences of an accident. The factor of 10 which you mentioned, which comes from our environmental statement, deals only with the consequence side.

It is indeed true, taking into account the population distribution at the alternative sites considered, that the consequences, should a serious accident occur, would be 10 times higher at the Clinch River site as compared to these alternative sites.

Senator HART. Because of population density?

Mr. CASE. Primarily because of population density.

Senator HART. I think there is a quarter of a million people living within 50 miles of the Clinch River site.

Mr. CASE. Yes; the element we must also consider is the probability of the accident. In both cases, due to the design requirements, the probability will be very low.

However, the consequences would be 10 times higher, although this is a 10 times change in a small risk. That is the point which had to be balanced against, in accordance with the Commission's decision, the effect of moving from the Clinch River site to these alternative sites, the effect on the timing goals set forth by the ERDA Administrator in his programmatic statement, since there would be some delay involved going from one site to the other.

Taking that timing into account, it was our view that you could not meet the programmatic goals as set forth by the ERDA Administrator.

Senator HART. What about the degrees of probability among the various sites?

Mr. CASE. Essentially, no difference at all.

Senator HART. Probability remains constant, consequences increase by virtue of staying at Clinch River?

Mr. CASE. Yes.

Senator HART. You mentioned, Mr. Gossick, the need for finding where "solution cavities exist" at the site. Can you assure the committee that this will take place or has taken place, talking about the site questions?

Mr. GOSSICK. With regard to the possible cavities?

Senator HART. Yes.

Mr. GOSSICK. Yes, sir. Certainly, that will be under continuing review and scrutiny by the NRC. It would continue if the project continues and, certainly, I assure the committee that will be looked at very carefully.

Senator BUMPERS. Would the chairman yield at this point?

What is pressure grouting?

Mr. GOSSICK. Sir, it is an injection, as I understand it, of cement or concrete into the subsurface, into the areas where it is suspected or known that there are cavities that have been formed by erosion.

Senator BUMPERS. I believe it was in your testimony that you said that would be a possible suitable solution to solving the cavities problems?

Mr. GOSSICK. Yes, sir. It is a common technique. As I understand it, many of the dams in the Tennessee area, one in particular I am familiar with, have used that technique.

Senator BUMPERS. The one I am familiar with is the Teton Dam. They used that technique there.

Mr. GOSSICK. I am not familiar with that, but that is putting it into the rocks. I think that was dealing with an earth dam. We are talking here about rock.

Senator BUMPERS. Are you not aware of the fact that that is precisely what caused the Teton Dam to——

Senator MCCLURE. I would say to the Senator that is not what caused the Teton Dam failing. The pressure grouting worked. They didn't do some other things that should have been done.

Senator HART. Rather than debate the Teton Dam, Mr. Case, I think you referred to the atmosphere in connection with consequences. Do I understand that among the alternative sites that the atmospheric conditions at Clinch River are such that any escaping radioactivity would remain in the area longer than the alternative sites?

Mr. CASE. The diffusion conditions are worse at the Clinch River site as compared to the alternative sites, so, the answer is yes.

Senator HART. On the question of containment and the consequences of core meltdown, since that has come up, we will quantify that, if we may. If you could, describe very briefly how such an incident would occur, or accident. It is my understanding what happens is the core eats its way down through the containment, possibly, and would potentially release large amounts of radioactive materials.

Second, in view of the seriousness of those consequences, what is the justification for excluding the so-called CDA from the required design criteria?

Mr. CASE. Yes, to your first question, a possible way of violating containment integrity following an extensive core meltdown would be for the core to melt down through the concrete and then violate integrity by moving into the ground.

An important consideration before that sequence of events is another possible method of losing containment integrity. That would be to literally blow the containment up due to overpressurization during a much shorter period of time. That is our principal concern with regard to the Clinch River reactor.

Our requirements are to avoid loss of containment integrity during the first 24 hours due to overpressurization, admitting the possibility, as is true in light water reactors, that you might lose containment integrity after that time through this meltdown process which you have described. The advantage of maintaining the integrity through the 24-hour period is to reduce the potential consequences of accidents due to radioactive decay during the 24-hour period.

The basis for accepting the small risk of the loss of containment integrity due to the meltdown phenomenon is the low probability that we believe of such an accident due to other design provisions.

Senator HART. Does the Clinch River design include a so-called core catcher?

Mr. CASE. The specific method by which they would assure this requirement of 24-hour containment integrity, I don't believe the project has figured it out yet, nor submitted it for our review.

Senator HART. It hasn't included or excluded it?

Mr. CASE. Right.

Senator HART. The French and British do include that feature?

Mr. CASE. Yes.

Senator DOMENICI. Did you say it had to be a core catcher?

Mr. CASE. The method used to satisfy this requirement has not been proposed by the applicant.

Senator DOMENICI. Thank you.

Senator HART. Senator McClure?

Senator MCCLURE. Thank you, Mr. Chairman.

Can you gentlemen tell us how long we have had liquid metal fast breeder reactors in operation?

Mr. BECKJORD. Since 1951.

Senator MCCLURE. EBR-1 went operational in 1951 and EBR-2 in 1963. There are others in the world besides those two experimental breeder reactors in the United States, is that correct?

Mr. BECKJORD. There are, I believe, eight that have been placed into operation, Senator, in the world.

Senator MCCLURE. Some of the design criteria in Clinch River are not necessarily just dreamed up out of engineers' dreams? They are based upon some experience with a breeder reactor of this kind?

Mr. CASE. Yes, sir.

Senator MCCLURE. The difference between this and those experimental breeder reactors is that of scale and the problems on scaling up to a demonstration plant and applying new techniques learned during the experimental breeder reactor operation. Is that correct?

Mr. CASE. Yes, sir.

Senator MCCLURE. Mr. Gossick, in your statement, you say, "Informational deficiencies were identified by the staff in a letter of November 1, 1974."

Have you compared those informational deficiencies with the allegations of the Burns & Roe memorandum?

Mr. GOSSICK. Let me check with the staff.

Senator MCCLURE. I see a number of heads shaking behind you.

Mr. GOSSICK. I am advised that some of the questions were involved and relate to the matters we have discussed here this morning that are in the memorandum.

I will ask Mr. Case to elaborate, but this is a normal part of our licensing process where the application is received and there is needed

information missing or information that needs to be clarified for the purposes of our staff review.

Mr. CASE. This is the usual case for us to find information deficiencies in a tendered application and to require that the deficiencies be remedied in the application to be docketed for review. There is nothing unusual in this case.

Senator MCCLURE. What I am interested in is whether or not the information which was in the Burns & Roe memorandum was by one means or another made known to or made a concern of the NRC.

Mr. CASE. The concerns with regard to grouting, solution cavities, were made known to the NRC, and were followed up in our review. The concerns relating to the physical characteristics of the site were made known to us, yes.

Senator MCCLURE. Even though the memorandum was not furnished to you and you didn't know of it until 2 weeks ago?

Mr. CASE. That is correct.

Senator MCCLURE. Nevertheless, the design criteria or the site selection problems that were outlined in that memorandum were either known to you or discussed by you over the period of the last 4 years?

Mr. CASE. Yes, sir.

Senator MCCLURE. I guess the bottom line would be, is there anything in the Burns & Roe memorandum which would change the NRC position on the site?

Mr. CASE. No, sir.

Senator MCCLURE. Mr. Beckjord, you were asked the question if you had discussed with Mr. Young the background of his assertions.

You said you had not discussed that with Mr. Young. In spite of the fact that you have not discussed it with him directly since the memorandum was called to your attention, do you have any knowledge of the background for his assertion?

Mr. BECKJORD. No, sir, I do not.

Senator MCCLURE. I suppose one thing that would concern me is the complexity of management of a plant of this kind, particularly with the way in which it was originally conceived.

As I understand it, and correct me if I am wrong, Consolidated Edison and TVA were copartners with ERDA in the development of this plant originally.

Mr. BECKJORD. Commonwealth Edison.

Senator MCCLURE. Excuse me, Commonwealth Edison and TVA. They were the essential prime participants in the Project Management Corp., that, as the cost overruns began to mount and the cost of the project and the delay of the project increased, in May of 1976, ERDA took over the management of the project, is that correct?

Mr. BECKJORD. That is correct, Senator.

Senator MCCLURE. ERDA has primary responsibility now, although Commonwealth Edison and TVA are still involved in supervision of the project?

Mr. BECKJORD. Yes, sir.

Senator MCCLURE. Do you see any difference in the difficulty of overseeing the project from ERDA's standpoint? Was there greater difficulty prior to May of 1976 than there is at the present time?

Mr. BECKJORD. There was greater difficulty prior to May 1976.

Senator McCLURE. Simply because there were more cooks stirring the broth?

Mr. BECKJORD. Yes, sir, at this point ERDA is solely responsible for the project and ERDA can act. There were possible situations before the change in May of last year where the activity could have become deadlocked because of disagreement.

If a disagreement had occurred among the principals, activity could have been brought to a stop. But that can't happen now.

Senator McCLURE. Mr. Gossick, could you comment on the same question, from the NRC standpoint?

Mr. GOSSICK. Senator McClure, from the information that we have, our staff does not consider that the Project Management constitutes a safety issue as far as the difficulty in managing the program is concerned. We consider that purely ERDA's concern.

Senator McCLURE. Again, the bottom line, I assume, from the standpoint of the hearing today is that there is nothing in the Burns & Roe memorandum of 1973 which you have not dealt with or are not dealing with currently?

Mr. GOSSICK. That is correct, sir.

Mr. CASE. Restricting it to those things that affect safety. There are a number of aspects that don't affect safety that we didn't even follow.

Senator McCLURE. They would not be your responsibility?

Mr. CASE. That is correct.

Senator McCLURE. Might I address the same question to ERDA. There are some aspects that are not simply from the standpoint of safety, that NRC would not be involved with, that ERDA might be concerned with.

ERDA has dealt with or is dealing with all of the items that are listed in the Burns & Roe memorandum of 1973?

Mr. BECKJORD. I would ask Mr. Caffey to comment on that.

Mr. CAFFEY. I would say, Senator McClure, that all aspects and apprehensions and concerns listed in the Burns & Roe memorandum which affect the project, this is aside from business matters of Burns & Roe, have been adequately dealt with except for those individual items of safety issues which we are still interfacing with NRC about.

All of the management aspects have been adequately dealt with.

Senator HART. Senator Domenici?

Senator DOMENICI. Thank you, Mr. Chairman. Just a few questions. Mr. Case, with reference to your statement defining risk as probability times consequences. Could you enlighten me with some specifics?

What kind of probability are you talking about in the two areas that have been discussed here today?

Mr. CASE. The probability that we are talking about, in our judgment, for a core disruptive accident is about 1 in 1 million or less per reactor year. In other words, the probability of such an accident, we believe, is less than one in a million per reactor per year.

Senator DOMENICI. You would be multiplying that times consequences of various alternative sites to arrive at your risk?

Mr. CASE. Yes, sir.

Senator DOMENICI. You made the conclusion then that because the probability is so small, when it is multiplied times a higher consequence, the risk is not increased that much in terms of other considerations. Is that correct?

Mr. CASE. That is correct. The risk is acceptable in either location. There is less risk at these alternative sites. But taking that smaller risk must be balanced against meeting the program objectives.

Senator DOMENICI. Just one last summary question for myself. I have been through the Clinch River project in the Energy Committee as a new member for a couple of months. In the process, I find we have been on this project for years with all kinds of differing scientific positions.

There have been scientists on both sides of this issue from its inception. There have been energy people on each side of this issue.

Is there anything about the internal memorandum which you now have in your possession which in any way changes your decisions to this point in time about its value?

Mr. GOSSICK. There is not, sir.

Senator DOMENICI. How about ERDA?

Mr. BECKJORD. None, sir.

Senator DOMENICI. If you had known about the memorandum 6 months after it was written, can you tell us that nothing would have changed with reference to the way you have proceeded with this project?

Mr. BECKJORD. There might have been a lot of activity when we discovered it as there has been over the past 2 weeks, Senator. I think that——

Senator DOMENICI. Would we be where we are today with this project, with the same requirements imposed at this point and the same licensing procedure?

Mr. BECKJORD. That is a fair statement, Senator. I believe so.

Senator DOMENICI. How about the NRC?

Mr. GOSSICK. I would concur in that. The matters in the memorandum that deal with the site have been brought out. So, there is nothing that would change matters as far as I can see.

Senator DOMENICI. Has there been a recent comparison of the three sites from the point of view of the allegations in the internal memorandum? Do we have that kind of evaluation somewhere in the record of the Federal Government?

Mr. BECKJORD. I guess I would refer to the report, the final environmental statement in which the alternate sites were evaluated. The general considerations were looked at at the alternative sites as well as Clinch River, the difference is that I don't believe extensive new borings were taken at those alternative sites.

If serious consideration were to be given at a future time to a different site, then that is the kind of work that you would do to establish that it is in fact suitable.

General considerations were looked at, at alternative sites, but not the specific structural mechanics of the sites.

Senator DOMENICI. Is it true that when you did do the specifics on this site, it proved out satisfactory with reference to meeting the necessary safety requirements?

Mr. BECKJORD. To the best of my knowledge, that site is wholly acceptable.

Senator DOMENICI. Thank you, Mr. Chairman.

Senator HART. Senator Bumpers?

Senator BUMPERS. Mr. Chairman, I just have one item I want to pursue at the expense of going over territory we have already covered.

I would like to ask Mr. Beckjord this: the thing that has caused me more concern, I think, over the Burns & Roe memo than anything else is the statement here, for example, where Mr. Young says:

> The overall approach to reactor safety matters has to date been based upon the Fast Flux Test Facility approaches, the policies established by Mr. Shaw in RRD, which are in many ways contrary to those of the AEC Nuclear Commission.

For example, Westinghouse and Burns & Roe have been told orally by RRD and PMC that we should not comply with the requirements of 10 CFR 50. They cite the DRL safety considerations and would not necessarily provide a simple reliable plan. Then he goes ahead to say this is part of the power struggle between the AEC and so on.

In your testimony, Mr. Beckjord, you say you started developing parallel systems; then you say, to cover hypothetical core disruptions, and then you drop that.

> In a May 1976 letter, the NRC agreed that these hypothetical core disruptions can and should be excluded from the design basis. Subsequently, the project withdrew the parallel design from further consideration by NRC, but it was mutually agreed that margins would be provided in the plant in order to reduce the postulated consequences of such hypothetical accidents.

It really seems to me, and I admit that I may be in error and I may be inferring something here that is in error, but it occurs to me that what Mr. Young has been told orally is precisely what happened, that we have cut corners on the safety specifications.

Mr. BECKJORD. Senator, I don't believe that is the case. Let me take your second question first, relating to the HCDA. The question that relates to the HCDA is whether the HCDA is to be accommodated within design basis.

That comes back to a discussion which I was trying to clarify earlier this morning, how a design is accomplished; as to whether the accident is fully contained and controlled within the design limits.

In the case of the HCDA, what has been decided is that the HCDA is not accommodated within design limits; it is accommodated in another way with margins built into the plant design so as to mitigate the consequences of that accident.

Mr. Case was explaining what the rationale for this is, namely, the probability of an HCDA is very low. My figures are somewhat lower than his. I would say that the probability of an HCDA is reckoned to be of the order of 10 to the minus 8 per year or less. So it has a very low probability of occurrence. The question is, what do you do about it.

Senator BUMPERS. You are not suggesting that you are entirely accurate on the probability, are you?

Mr. BECKJORD. No, sir, 10^{-8} or less.

Senator BUMPERS. OK.

Mr. BECKJORD. This accident has been studied extensively. For it to occur—let me just say a little more about it. I know of no mechanistic way that it can happen. It is called hypothetical because for the purposes of analysis and discussion, we assume it can happen, but nobody has come up with a mechanism by which it could logically occur.

As an example, the core of the Clinch River reactor consists of fuel material and it is encased in cladding material and structural material. In order for the worst HCDA to occur, they would have to develop some way in which the cladding material and the structural material would fall away. It might melt, but the fuel would stay in place. I don't know of a way that this can happen, so it is considerations like this.

I am trying to describe it in a very simple fashion which has been studied extensively. It is by reasoning such as this that the probability that it could happen is reduced; and 10^{-8} is a very small number.

What do you do about it? Do you conceive of a design which will accommodate this very unlikely event within design limits or variables or do you find some other way to handle it? The path that has been chosen is to build other margins into the plant.

Senator BUMPERS. Mr. Beckjord, what are the probabilities by ERDA's estimates of an explosion occurring in a breeder reactor plant?

Mr. BECKJORD. That would be the same order, 10^{-8} per reactor year. I might add that one of the margins that is to be included in this plant design is the capability to withstand a very sharp explosion.

The words "energetic disassembly" came up earlier. Maybe that is overly technical, but we have been in discussions with the Nuclear Regulatory Commission on the amount of energy, the amount of explosive force that must be accommodated within the structure. That matter is not settled yet.

Senator BUMPERS. Incidentally, the one that Senator McClure referred to that was put in operation in 1951 did explode, didn't it?

Mr. BECKJORD. No; it did not. That was a meltdown.

Senator HART. I think that was the original question. You say you are using figures of 1 out of 10^{-8}, when in fact six breeders have been developed where two of them have had meltdowns which I understand to be contained in the definition of a core disruptive accident.

When you use the term hypothetical because you can't conceive of it ever happening, it has happened twice, at the Idaho Falls plant and the plant in Detroit. Am I missing something here?

Mr. BECKJORD. Yes. The hypothetical accident we are talking about here is a lot more severe.

Senator HART. Let's talk about one that is not so severe because I understand the definition of hypothetical care disruptive accidents includes core meltdown and it has happened two times out of six.

Mr. BECKJORD. We are talking about a total core meltdown.

Senator HART. Well, let's talk about a little core meltdown.

Mr. BECKJORD. One occurred at the plant in Detroit. Part of the subassembly did melt.

Senator HART. Does HCDA include a little core meltdown?

Mr. BECKJORD. No; that is a big one.

Senator HART. What do you call a little one?

Mr. BECKJORD. A little one is a core melt.

Senator HART. A hypothetical core disruptive accident——

Mr. BECKJORD. That is the big accident.

Senator HART. What is the dividing line between big and little?

Mr. BECKJORD. A little one, I would define that as the accident that occurred at the Fermi plant. Part of the assembly melted. The reactor was shut down. It was safely shut down without activity released to the environment or injuries to the public.

Senator BUMPERS. It is still shut down, isn't it.

Mr. BECKJORD. After that accident, the vessel was opened, the cause of the accident was determined, the deficiencies were corrected, and that plant was placed back into operation. It operated, I don't know, for 2 or 3 years. It was finally shut down based on economic considerations; but the plant did operate again after that accident.

Senator HART. It seems to me there is little circular reasoning here, it is little if nothing bad happens. If something bad happens, it is big; but a big one can't happen.

Mr. BECKJORD. That is certainly not the impression I am trying to convey, Mr. Chairman.

Senator HART. The Fermi meltdown, little because nothing got away from it, the operator?

Mr. BECKJORD. Yes.

Senator MCCLURE. I thought he said the Fermi could be characterized as big.

Mr. BECKJORD. No.

Senator HART. It can't be big because a big one can't happen.

Mr. BECKJORD. The Fermi accident occurred. There was a flaw in design. It happened one day that the coolant flow channel was blocked. That is what happened at the Fermi reactor. With no flow permissible in that channel, there was melting. When the assembly was melting, the plant was shut down right away. It was detected.

Senator HART. What we are trying to get at is what a hypothetical core disruptive accident is.

Mr. BECKJORD. A hypothetical core disruptive accident is the worst accident that can be conceived of for this reactor.

Senator HART. But it can't happen, but it can be conceived of?

Mr. BECKJORD. No; it can be conceived of; but what I am saying is that I can't give you a mechanism by which it could happen. In other words, we assume that something like that could happen and we look at the consequences; but I am telling you I don't know how it could happen. I can't come up and give you a sequence of events that will lead to that accident.

It is typical in the accident analysis of nuclear reactors that we don't always go into the mechanism. We assume that the worst possible thing can happen. We try to figure out a way in which it might happen. If we can figure out a way, then we do something about it.

Senator HART. The key point here is you structure your design studies and analyses by a standard called a hypothetical core disruptive accident, but by your own definition, that is a set of circumstances which cannot occur or which you cannot conceive of occurring?

Mr. BECKJORD. No, sir. I don't know of a way it could happen. The studies have shown that the probabilities of it happening are very small. That is what we are saying. However, nonetheless, even though they are very small, there are margins in the design to accommodate such an event and to mitigate its consequences. Those have been required by the Nuclear Regulatory Commission.

Senator HART. I apologize for interrupting, Senator Bumpers.

Senator BUMPERS. I am about finished anyway. The term meltdown could not have occurred if we had used the so-called core catcher technology—I am sorry, the pool technology which the French and British are using?

Mr. BECKJORD. Yes, sir. Could you repeat that?

Senator BUMPERS. Could the Fermi meltdown have occurred if we were using the so-called liquid sodium pool technology?

Mr. BECKJORD. The pool or the loop would make no difference. That it would not have an effect on meltdown—it could happen. If there was the same design defect in the pool system, it could have happened there.

Senator BUMPERS. Do you personally feel as far as you know, anybody in the agency feel that the loop method which we are going to use is preferable to the pool techniques?

Mr. BECKJORD. Let me give you a short answer on that, Senator. I believe that a safe system can be built using either approach. Each one has advantages and disadvantages. I think that from a safety point of view, they can and will be equivalent.

What we don't really know, what nobody knows is which one is going to be more economical in the end. The French cite important advantages for their system. There are important advantages for ours. One which we think is important and which the Germans also think is important is the ability to inspect the entire system during periods of shutdown. That is not totally possible with the pool system. That is an advantage for the loop type of system.

Senator BUMPERS. Have you seen this memo dated June 20, 1976, submitted to ERDA and the Electric Power Research Institute? It has Burns & Roe and Rockwell International at the bottom of that. Have you seen that? It is NRB 76-1. I assume that this is something that came to ERDA from Rockwell and Burns & Roe. Their conclusion is that the pool concept is favored over both the hybrid and the loop designs and they set out numerous reasons why.

Mr. BECKJORD. Yes; I am aware. I recall now that report. I think that I will stand on my statement. I think that most of the people in the business in this country will agree that either system can be made, that the two systems can be made equally safe, Senator; but as I say, there is this controversy over which one will ultimately be more economical.

Senator BUMPERS. They go ahead to say that the total probability of the core disruptive accident occurring by the pool concept is calculated to be approximately one-fifth and two-fifths that of the loop and hybrid concepts, respectively. That is contrary to what you said a minute ago. These are the people that are building it.

Mr. BECKJORD. Can I provide an answer for the record on that point, Senator? I will stand on my statement.

Senator BUMPERS. Yes. Of course, we are going to debate this thing this afternoon. If you don't have it to me before 2 o'clock, I will take dramatic liberties with this memo and debate on the floor.

Mr. BECKJORD. All right, sir, 2 o'clock.

Senator MCCLURE. Mr. Chairman, I think it might be helpful if we would put in the record a listing of the liquid metal fast breeder reactor plants that have been either operated or under design and include a prototype which has been talked about for operation in 1988.

It starts with Clementine, an experimental reactor in 1946 which has been decommissioned; EBR-1 which was operational in 1951 and has been decommissioned. Incidentally, EBR-1 is the plant that first produced commercial electricity. It lighted a small city in Idaho near the test station.

The Fermi plant was decommissioned, but became operational in 1963; EBR-2 in 1963; SEFOR, which was located in Arkansas, started in 1969, has been decommissioned; and the FFTF, Clinch River breeder reactor; if that list might be made part of the record.

Senator HART. Without objection.

[The list follows:]

U.S. LMFBR PLANTS

Name	Type	Location	Power level	Initial Operation	Current status
Clementine	Experimental	Los Alamos	0.25 MWt	1946	Decommissioned.
Experimental Breeder Reactor I (EBR I).	Experimental and demonstration.	National Reactor Testing Station (NRTS), Idaho.	1.2 MWt	1951	Do.
Fermi Atomic Powerplant	Power	Michigan	66 MWe	1963	Do.
Experimental Breeder Reactor II (EBR II).	Testing	NRTS, Idaho	62 MWt (20 MWe)	1963	Operational.
SEFOR	do	Arkansas	20 MWt	1969	Decommissioned.
Fast Flux Test Facility (FFTF)	do	Hanford, Wash	400 MWt	1979	Under construction.
Clinch River Breeder Reactor (CRBR).	Power (demonstration).	Tennessee	350 MWe	1983	Under design.
Prototype Large Breeder Reactor (PLBR).	Power (near commercial).	Undetermined	1000 MWe	1988	Under conceptual design.

Senator HART. Gentlemen, thank you, very much.

Senator DOMENICI. If you will supply that answer that you were going to provide for Senator Bumpers.

Mr. BECKJORD. Yes, sir, before 2 o'clock.

[The information requested by Senator Bumpers and Mr. Beckjord's prepared statement follow:]

HYPOTHETICAL CORE DISRUPTIVE ACCIDENTS FOR LMFBR'S POOL VERSUS LOOP

The risk associated with a postulated HCDA is the product of the consequences (magnitude) and probability of occurrence. The magnitude of such a postulated event is dependent on the core composition and geometry, and therefore consequences are not affected by whether a pool or loop design is assumed. A report dated June 25, 1976, FBR-76-1, by a single contractor team (AI/BRI) concludes that the probability of occurrence of an HCDA may be a factor of five less for a pool type LMFBR than for a comparable loop plant similar to the CRBRP.

Both the loop and pool concepts are safe and either would meet all appropriate safety requirements. The comparisons made by AI/BRI were to consider overall design advantages and not to compare absolute safety.

The AI/BRI conclusion that the pool design has a factor of five lower probability for occurrence of an HCDA than a loop plant is based solely on the larger sodium inventory immediately surrounding the core. For postulated events such as loss of offsite power or large earthquakes coupled with a simultaneous loss of core cooling, they compute that about 14 hours are available for corrective action for the pool plant as opposed to four hours for the loop plant. Either time is sufficient to take corrective action but this difference does affect the probabilities somewhat. However, since they are comparing very small numbers like 1 chance in a million to 1 chance in a billion, uncertainties in the input data are too great to claim factor of five difference between pool and loop designs.

STATEMENT OF ERIC S. BECKJORD, DIRECTOR, DIVISION OF REACTOR DEVELOPMENT AND DEMONSTRATION

Mr. Chairman and Members of the Committee, I appreciate this opportunity to discuss the environmental and safety matters related to the Clinch River Breeder Reactor Plant (CRBRP) Project which were raised in the July 6, 1973, internal Burns and Roe memorandum recently cited in the press.

The CRBRP Project is a joint government-industry cooperative arrangement for demonstrating a Liquid Metal Fast Breeder Reactor power plant as authorized by Congress on June 2, 1970 (Public Law 91-273). The partners in this project are the Energy Research and Development Administration (ERDA), Commonwealth Edison (CE), Tennessee Valley Authority (TVA) and Project Management Corporation (PMC). The objectives of this Project are to design, license, construct, test and operate an LMFBR demonstration plant. In May 1976, ERDA assumed full management control of the Project with continued utility industry support and participation.

I have had the ERDA responsibility for this Project since March 1976. During that time, Project accomplishments have been good, with design now over 40 percent complete, all of the long lead equipment on order, and the Final Environmental Statement and Site Suitability Report issued by the Nuclear Regulatory Commission (NRC). I have examined Project records, reviewed the numerous reports and hearings concerning the Project, and inquired extensively into Project procedures and status, particularly in environmental, safety and related licensing matters. Generally, I can say that the Project has also made good progress in these licensing areas during the past year, working toward its goal of a Limited Work Authorization (LWA) as required under the NEPA Act of 1970, until the recent suspension of the environmental hearings in April. The environmental hearings suspension was requested by ERDA pending a final decision on whether the Project is to be terminated or continued.

I have reviewed the Burns and Roe memorandum in detail since it became available to me about two weeks ago. My statements on it are based on the information available to me as a result of research done in the interim. Some of the issues raised were speculative and others were founded on incomplete or incorrect information. Of the remaining issues, I found either that they have already been resolved or that work toward proper resolution is underway in conjunction with licensing activities as required by NRC.

Comments on the specific issues raised by the Burns and Roe memorandum are as follows:

In the "Summary" section, pages 2 and 3, Burns and Roe stated:

"The site selected is likely to be very costly to prepare and could even be unsuitable.* * *"

The cost of preparing the Clinch River site will have been proven to be substantially more than estimated. The site costs and problems could be such as to indicate a change of site."

The plant site was selected following consideration of several possible alternative sites. In late 1971, the AEC appointed a Senior Utility Steering Committee and Senior Utility Technical Advisory Panel to assist them in selecting a utility partner to design, build and operate the demonstration plant. Proposals were submitted to the Steering Committee and AEC by groups of utilities interested in participating in the demonstration plant program. Each of the principal sites advanced in the proposals received appeared to meet the general requirement that the proposed site should require no unusual design features or special consideration in licensing. The Steering Committee found, however, that the proposal from CE/TVA offered increased siting flexibility over the other proposals. This CE/TVA proposal for building and operating the LMFBR demonstration plant was ultimately accepted by the Steering Committee and the AEC.

Three candidate sites within the TVA area were considered: Widow's Creek, John Sevier and Clinch River. Analysis of the relevant siting, environmental and direct cost factors for the three sites disclosed no clear-cut or overriding advantages for any single site. Such differences as existed were considered amenable to treatment in the design within the limits of existing technology. As a practical matter, the three candidate sites were found to be equivalent from site characteristic and environmental standpoints.

Although comparisons of direct site cost slightly favored the Widow's Creek and John Sevier sites because of the availability of some site services, the differences were within the range of uncertainty inherent in such cost estimates.

An overall analysis of the three sites, including considerations of meeting project and program objectives, showed Clinch River to have a decisive advantage, because the new site services to be provided at Clinch River would be more compatible with the nuclear steam supply system.

The soundness of that original decision was supported by the comprehensive and detailed site investigation program conducted during 1973, subsequent to the Burns and Roe memorandum. In contrast to the Burns and Roe apprehension, the site was actually found to be similar to others utilized for nuclear power plants in the region and was demonstrated to be fully acceptable from all standpoints. The Nuclear Regulatory Commission also confirmed the acceptability of this site based on their independent review and assessment as documented in the Final Environmental Statement for the CRBRP issued in February 1977 and the Site Suitability Report issued in March 1977. In the Site Suitability Report, NRC concluded that the foundation conditions were generally good and there was no subsurface conditions expected which would preclude the suitability of the site or the construction of the proposed plant. As the nuclear power plant siting criteria have undergone very substantial evolution over the past several years, the continued acceptability of this site further reinforces the soundness of its selection.

With regard to the cost of preparing this site, any additional costs incurred for preparation of this site compared to a hypothetical "optimum" site will be small when considered in the context of the many other factors influencing site selection.

In the "Background" section, pages 8 and 9, the Burns and Roe memorandum states:

"The overall approach to LMFBR reactor safety matters has to date been based on FFTF [Fast Flux Test Facility] approaches and policies established by Mr. Shaw and RRD [Division of Reactor Research and Development] which are in many ways contrary to those of the AEC Division of Regulation (DRL). For example, Westinghouse and Burns and Roe have been told orally by RRD and PMC that we should not comply with the requirements of 10CFR50 Appendix A (General Design Requirements) for LMFBR where such requirements arise from theoretical DRL safety considerations and would not necessarily provide a simple, reliable plant. * * * "This approach is being fostered in full knowledge that it may not result in meeting DRL's licensing requirements and that many issues would have to be taken to the AEC Commissioners for resolution. It is part of a power struggle between parts of the AEC. The LMFBR Demonstration Plant is viewed as a test case in which RRD and PMC can knock out many theoretical safety-oriented design features which complicate commercial plants and make them more expensive, and in which a new approach to safety and licensing can be established. In addition, the Demonstration Plant is viewed as having to be consistent with FFTF in order to justify the approaches on that project. Unfortunately, some safety approaches on FFTF were apparently decided on because of the severe cost bind that project is in. * * *

"A number of existing approaches based on FFTF practices are already known as potential problem areas. These include the lack of specific safety criteria for the project; present emergency core cooling provisions and natural circulation assumptions: the current assumption that a double-ended pipe break is not a credible accident; the assumptions as to the extent of the Hypothetical Core Disruptive Accident (HCDA) and features needed to contain it; the effects of sodium spills and fires; radioactivity release above the operating floor; plutonium leakage and levels at the site boundaries; and the ability to design an effective system to contain a core and reactor vessel meltdown. * * *"

This statement concerning compliance with 10CFR50 requirements appears to be in direct conflict with the requirements established by the AEC for this Project in material submitted to the Congress prior to authorization. In the original Program Justification Data Arrangement for this Project submitted to the JCAE on August 11, 1972, it was clearly stated that:

"All applicable laws and regulations, including those pertaining to AEC licensing and regulations, will be complied with."

This same requirement, updated to reflect the establishment of the independent NRC, is in the Revised Program Justification Data Arrangement No. 77-106 which covers the Project at this time.

The minutes of the Project Steering Committee have been reviewed and no record was found to support the statement made by Burns and Roe concerning compliance with 10CFR50 requirements. In addition, I have personally called a number of men who were leaders in the early days of the Project. These are

Messrs. Milton Shaw and Thomas Nemzek, former directors of ERDA's reactor development division, Mr. Wagner of TVA, Mr. Wallace Behnke of Commonwealth Edison, Messrs. John Taylor and George Hardigg of Westinghouse. Each of them has assured me there was never either a policy or a practice of avoiding compliance with the AEC Division of Regulation licensing requirements. It was in fact the policy to go through the entire safety and licensing process as part of the project objectives. It was understood by the project leaders that modifications to some of the 10CFR50 General Design Criteria would need to be developed, simply because of the technical differences between Light Water Reactors (LWRs), for which the General Design Criteria were originally written, and the Clinch River Breeder Reactor, for which general design criteria were not yet written in 1973. These modifications were developed within the licensing process and are consistent with the evolution of the licensing process for LMFBRs. It should be noted that much work and discussion was required to resolve the differences of technical opinion prior to the final issuance of CRBRP general design criteria by NRC on January 9, 1976. The fact that there were significant differences of technical opinion during this effort, however, does not lead to the conclusion that the Project was trying to avoid compliance with safety requirements. The safety requirements were properly established when NRC issued, and the Project accepted, these criteria.

The objective of the design criteria and the net effect of the CRBRP licensing process is to make the CRBRP at least as safe as a light water reactor located at the same site. To suggest, as the Burns and Roe memorandum does, that there was an intent not to comply with licensing requirements or that the AEC desired to avoid including needed safety features because of cost considerations, is simply not supported by the facts.

I can further testify that during my association with the Project, the policy has been, is now, and will continue to be, to comply with the Nuclear Regulatory Commission's licensing requirements.

The three level defense-in-depth safety philosophy currently being used for design of LWRs was also adopted for CRBRP. This requires design measures to prevent accidents, to provide protection against either anticipated or unlikely faults that might occur, and beyond this to provide appropriate engineered safety features in the design to safely accommodate extremely unlikely faults, if they somehow should occur, in order to protect the health and safety of the public. Furthermore, ERDA and NRC have agreed that, for the CRBRP, it is prudent to include additional measures in design to further limit potential consequences to the health and safety of the public. Accordingly, the Project has included margins beyond the necessary design basis in order to reduce the postulated consequences of hypothetical accidents involving core meltdown and energetic disassembly. At the time of the Burns and Roe memorandum, there were on-going discussions between RRD and DRL concerning whether hypothetical core disruptive accidents (HCDAs) should be included in the design basis (level three) for LMFBRs. The resolution with DRL was that, to avoid schedule delay, two CRBRP designs would be submitted for concurrent review, one without and one with HCDAs in the design basis (the reference design and a parallel design).

In a May 1976 letter, the NRC agreed that HCDAs can and should be excluded from the design basis. Subsequently, the Project withdrew the parallel design from further consideration by NRC, but it was mutually agreed that margins would be provided in the plant design in order to reduce the postulated consequences of such hypothetical accidents so that the CRBRP would be comparable to current LWRs.

All of the relevant CRBRP safety issues, including those raised by the Burns and Roe memorandum, are being properly and thoroughly analyzed during the course of the licensing process. Most of the issues have been resolved in a manner mutually acceptable to ERDA and NRC. Work is continuing on the remainder of these issues at this time. No unusually difficult problems in design have been identified. To date, the Project has made design changes estimated to ultimately cost $60 million in order to meet additional licensing requirements which have evolved during the interactions with NRC, and it is possible that other changes may yet be required. You may be assured, however, that we have always been, and are at present, dedicated to meeting all necessary licensing requirements.

In the "Background" section, pages 14 and 17, the Burns and Roe memorandum states:

"The licensing approach involves numerous variance requests and submittals not originally included. * * *

"It appears likely that the Regulatory group of the AEC will be made independent of the development part of AEC soon. This would mean far less chance of early and unique licensing approvals. * * *"

The CRBRP Project has asked for no special licensing variances. Consistent with one of the major CRBRP Project objectives of demonstrating the licensability of the LMFBR concept, the CRBPR is being subjected to the identical licensing process by the NRC as would any commercial nuclear power plant. At the time of the Burn and Roe memorandum, the Project was expecting to request an exemption to conduct certain site preparation activities prior to receipt of a Construction Permit, as was permitted by the AEC Regulations under 10 CFR 50.12(b). At that time, the AEC was granting exemptions for commercial nuclear power plants under this regulation since this was prior to institution of the use of LWAs. When the regulations were changed to incorporate the LWA procedure, the Project abandoned consideration of an exemption request and oriented licensing activities toward obtaining an LWA.

Regarding other NRC requirements, the Project will meet all of the applicable requirements. However, as already stated, some of the NRC requirements were formulated for LWRs and have either no applicability or only partial applicability to the CRBRP. In these cases, the Project will meet the intent of the LWR requirements by developing modified or new requirements in cooperation with NRC (e.g., 27 of the 56 General Design Criteria were modified, plutonium dose guidelines were developed, and new containment criteria were developed).

In the "Background" section, pages 17 and 18, of the Burns and Roe memorandum, additional statements concerning the site appear:

"The site conditions described below may delay establishment of the suitability of the site.* * *

"The Clinch River site selected for the LMFBR Demonstration Plant is one of the worst sites ever selected for a nuclear power plant based on its topography and rock conditions. The suitability of the site will not be confirmed until after an extensive soil boring program. There is a possibility that the site may not be acceptable. As a minimum, site development costs will be high. The reasons for the above conclusions are as follows:

"(a) The site has varying rock conditions. The rock on which we are attempting to place the plant is known to be somewhat nonhomogeneous and to be subject to possible solution activity problems and perhaps voids and cavities. These conditions may require some rock treatment such as grouting, and verification of the results by an added soil boring program. Previous sites with similar problems have been difficult to license and have been difficult and costly to prepare.

"(b) The areas surrounding the present estimated plant location are known to have an as yet undetermined degree of voids and cavities. Because of this condition and the large amount of excavation required by the design depth of containment at the present time, an extensive rock treatment (grouting) effort appears to be required, followed by a detailed soil boring program to verify that the results are satisfactory. This effort is anticipated to be required to avoid possible severe subsidence problems, which could be the equivalent of a seismic event. The AEC has insisted on such actions for previous sites with less extents of voids and cavities; considerable costs and delays have been involved.

"(c) Slope stability will be a problem during construction due to the nature of the site material.

"(d) Extensive excavation, including much into bedrock, and backfill is presently estimated to be required because of the hilly terrain and subsurface conditions at the site.

"The results of the above could mean a minimum of more than six months' delay and millions of dollars in cost increases. In addition, final location and orientation of the plant will be delayed pending results of the soil boring program. * * *"

These apprehensions of Burns and Roe about the site were based on twenty-four core borings at the proposed site, of which only four were in the immediate vicinity of the plant location. After a comprehensive and detailed site investigation program, the final plant location at the Clinch River site was proven to be sound. This site investigation program included over one hundred additional core borings, a test grouting program to confirm the homogeneity of the foundation stratum, detailed geophysical studies, and other extensive analyses and tests. All these points raised by the Burns and Roe memorandum were fully and thoroughly reviewed with NRC prior to their issuance of the Final Environmental Statement and the Site Suitability Report for the CRBRP. The NRC staff concluded that the foundation conditions are good and that the site was suitable for construction of the plant.

In the "Background" section, page 22, the Burns and Roe memorandum states:

"Many safety approaches incorporated in FFTF and planned for the LMFBR Demonstration Plant may not be commercially licensable. These plant features could be addressed and resolved during the Demonstration Plant licensing process."

This comment was made at an early point in the plant design. As has already been explained, one of the key objectives of this Project has been to license this plant in the same manner as a commercial LWR plant. Many of the specific approaches and features which were ultimately incorporated into the design required extensive study, analysis and development. The problems identified in the Burns and Roe memorandum have each been addressed in the licensing process as the design has evolved. Either they have been resolved or appropriate work is underway to resolve them.

In conclusion, I wish to emphasize the following points:

The goal of the CRBRP design has been to provide a plant which is at least as safe as an LWR located at the same site.

Since the commencement of the project, it has been the policy to go through the entire licensing process and to comply with licensing requirements established by the AEC Division of Regulation and its heir, the Nuclear Regulatory Commission. All NRC licensing requirements are being fulfilled in the project implementation.

The internal Burns and Roe memorandum is over four years old. Some of the issues raised in it were speculative and we have not found a basis for them. The remaining issues have each been properly addressed in our detailed design and site investigations and with the NRC in the licensing procedures. Each issue has been fully and completely resolved or appropriate work toward resolution is currently proceeding.

NRC has agreed that the comprehensive site investigation program has established that the site meets NRC requirements.

Good progress was made by the Project in the licensing area during the past year until the suspension of licensing hearings in April.

That concludes my statement, Mr. Chairman. I will be glad to answer any additional questions the Committee may have on this subject.

Senator HART. The next witness will be Mr. William Young, vice president of the Breeder Reactor Division of Burns & Roe.

Would you identify for the record, those who are accompanying you?

STATEMENT OF WILLIAM H. YOUNG, VICE PRESIDENT, BREEDER REACTOR DIVISION, BURNS & ROE, INC., ACCOMPANIED BY DR. SEYMOUR BARON, SENIOR CORPORATE VICE PRESIDENT FOR ENGINEERING AND TECHNOLOGY

Mr. YOUNG. Yes, sir. I am William H. Young. This is Dr. Seymour Baron, who is senior corporate vice president for engineering and technology.

I would like to read through my prepared statement which has been submitted along with a number of detailed attachments which I will not read.

On the point that Senator Bumpers just brought up, after my prepared statement, I certainly would be willing to answer questions on that document that he held up. I think it might be quite important.

Senator HART. At an appropriate time. I would encourage you to condense where possible your prepared statement.