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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

NRC STAFF TESTIMONY OF BILL M. MORRIS, JERRY J. SWIFT, JOHN K. LONG
EDMUND T. RUMBLE, III., MOHAN C. THADANI, LEWIS G. HULMAN
ON INTERVENORS' CONTENTION 2 AND ITS SUBPARTS
2c, 2d, 2f, 2g AND 2h
AND CONTENTION 3 AND ITS SUBPARTS
3c AND 3d

Q1. Please state your names and affiliations.

A1. My name is Bill M. Morris, I am employed by the U.S. Nuclear Regulatory Commission as a Section Leader of the Technical Review Section, Clinch River Breeder Reactor Program Office in the Office of Nuclear Reactor Regulation. My involvement in the Clinch River Breeder Reactor (CRBR) review is that I am responsible for direction of the Technical Review Section's review of the fast sodium-cooled-related aspects of the CRBR safety review.

My name is Jerry Swift. I am employed by the U.S. Nuclear Regulatory Commission as a Reactor Engineer, Clinch River Breeder Reactor Program Office in the Office of Nuclear Reactor Regulation. My involvement with this phase of the CRBR licensing review has been to coordinate and review the radioactive source term analysis and analysis of radiological consequences of accidents for the Site Suitability Report and for the Environmental Statement.

My name is John Long. I am employed by the U.S. Nuclear Regulatory Commission as a Reactor Engineer, Clinch River Breeder Reactor Program Office in the Office of Nuclear Reactor Regulation. My involvement with the CRBR review has been with the analysis of core disruptive accidents.

My name is Edmund T. Rumble, III. I am employed as a Corporate Vice President of Science Applications, Inc. (SAI). Presently, I am providing consultant services to the Nuclear Regulatory Commission. My involvement with the CRBR review has been as a member of an SAI team providing technical assistance to the Office of Nuclear Reactor Regulation on safety matters related to the proposed CRBR.

My name is Mohan C. Thadani. I am employed by the U.S. Nuclear Regulatory Commission as a Nuclear Engineer in the Accident Evaluation Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation. My involvement in the Clinch River Breeder Reactor review has been to perform evaluations of environmental and public risks of postulated CRBRP accidents. I have contributed to the preparation of the Draft and Final FES Supplement.

My name is Lewis G. Hulman. I am the Chief of the Accident Evaluation Branch, Division of Systems Integration in the Office of Nuclear Reactor Regulation of NRC. My involvement with the CRBRP review has been in the management and quality control of analyses of accidents, their consequences, probabilities and associated risks with respect to adherence to Commission policy and staff practice.

Q2. Gentlemen, have you prepared statements of professional qualifications?

A2. Yes. A copy of Mr. Thadani's statement of professional qualifications is attached to this testimony. Statements of professional qualifications for the other witnesses have been previously received in evidence on Tr. 2478 for Dr. B. M. Morris, on Tr. 2479 for Dr. Jerry J. Swift, on Tr. 2482 for Dr. Edmund T. Rumble, III, on Tr. 2529 for Mr. Lewis G. Hulman, and on Tr. 2533 for Dr. John K. Long.

Q3. What subject matter does this testimony address?

A3. This testimony addresses the adequacy of the Staff's analysis of accidents for environmental review of the Clinch River Breeder Reactor (CRBR). This issue is defined in Natural Resources Defense Council, Inc. (NRDC) and the Sierra Club's Contentions 2c, 2d, 2f, 2g, 2h, 3c and 3d as follows:

Contention 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), fn. 1.

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.

f) Applicants have not established that the computer models (including computer codes) referenced in Applicants' CDA safety analysis reports, including the PSAR, and referenced in the Staff CDA safety analyses are valid. The models and computer codes used in the PSAR and the Staff safety analyses of CDAs and their consequences have not been adequately documented, verified or validated by comparison with applicable experimental data. Applicants' and Staff's safety analyses do not establish that the models accurately represent the physical phenomena and principles which control the response of CRBR to CDAs.

g) Neither Applicants nor Staff have established that the input data and assumptions for the computer models and codes are adequately documented or verified.

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

Contention 3

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

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.

Q4. Drs. Morris, Swift, Long, Rumble, Mr. Hulman and Mr. Thadani, what analyses of core disruptive accidents (CDAs) and their consequences have been performed by the Staff for purposes of performing the NEPA cost/benefit analyses?

A4. The FES and its Supplement describe CDAs and the general classes of events potentially leading to CDAs. A comparison of selected CRBRP accident sequences was made with those in the Reactor Safety Study (WASH-1400) to gain perspective on risks of very severe accidents in CRBRP. Our discussion of accidents in the FES and its Supplement is in keeping with the guidance of the Commission's Statement of Interim Policy on Nuclear Plant Accident Considerations Under the National Environmental Policy Act of 1969 (45 F.R. 40101, June 13, 1980).

In Appendix J of the Supplement to the FES, accident sequences are discussed further, including estimates of their frequency and consequences. These sequences form a broad characterization of CDAs initiated by:

(1) failure to adequately cool the fuel as may result from a loss of heat sink (LOHS), loss of coolant accident (LOCA), or massive flow blockage; (2) failure to terminate the fission chain reaction when necessary, as may result from a failure to scram during a (unprotected) loss of flow event (ULOF), or an unprotected transient overpower event (UTOP); and (3) core-wide fuel failures as may result from propagation of local fuel faults (FFP). ULOF and UTOP events are specific events within a more general category often designated as Anticipated Transient Without Scram (ATWS).

Q5. Gentlemen, have you considered CDAs under 10 CFR Part 100.11(a), fn. 1 as part of the Staff's environmental review?

A5. No.

Q6. Why not?

A6. As discussed extensively in our testimony for the Site Suitability hearing, in our judgement CDAs are not considered credible events in the context and practice associated with 10 CFR Part 100.11(a). In keeping with NEPA, however, we have given due consideration to the consequences, probabilities and risks of such events (and the full range of possible accidents) in the FES and its Supplement.

Q7. Drs. Morris, Swift, Rumble and Long, are the CDA initiation frequencies employed in the NEPA analysis based on any detailed reliability analyses of CRBRP systems design features?

A7. No. While numerous detailed reliability analyses have been conducted on proposed CRBRP systems and features, they do not form the basis for the estimated CDA initiation frequencies. Instead they form a portion of the knowledge base from which judgements regarding these frequencies were drawn.

Q8. How have these CDA initiation frequencies been determined?

A8. CDA initiation frequencies have been determined by judging the feasibility of achieving a specific level of performance. This judgement was based on three points. First, we considered general characteristics of the CRBRP system design as proposed including its inherent redundancy, diversity, and independence and its perceived interfaces with support systems such as electrical power, operators and maintenance personnel. Secondly, we considered the potential for achieving high reliability in the design through implementation of an effective reliability program. Finally, quantitative bounding CDA initiation frequencies for the CRBR design were estimated based on the above and on relevant LWR operating experience including the pertinent information available from reliability oriented studies of LWRs and LMFBRs.

Q9. How did you arrive at the specific CDA initiation frequency estimates attributed to ATWS events (i.e. ULOF and UTOP)?

A9. In NUREG-460, "Anticipated Transients Without Scram for Light Water Reactors," Vol. I, Section 4.3, an estimate of the frequency of ATWS for typical LWRs was given as 2×10^{-4} per year. Estimates in this same range were subsequently quoted by the Commission in its statement regarding ATWS rulemaking. These ATWS frequency estimates were based on operating LWR experience including a variety of designs and plant ages. Specifically taken into account were the number of years of operating experience, the frequency of anticipated transients, and the occurrence of failures of shutdown system components (ATWS precursors) which if coupled with additional failures could have led to shutdown system failure upon demand.

Against this background we evaluated the CRBR shutdown system design criteria. The most important factor considered was the extra redundancy, independence and diversity of the proposed CRBR shutdown systems. The currently proposed design of the CRBR shutdown system includes two independent and diverse systems, each of which is comparable to an LWR shutdown system. Each of these systems will meet the single failure criterion, the criteria for independence between redundant channels and will include measures for diversity such as diverse logics, circuitry, actuating mechanisms, and sensors. We also took into consideration the nature of the ATWS precursors from LWR experience to determine if there were any special lessons related to the CRBR design. Some LWR ATWS precursors seem relevant to CRBR but others do not. We also took into consideration the potential frequency of occurrence of transients at CRBR, the potential for achieving high reliability through implementation of a formal reliability program, and the possibility of common mode failures of the two shutdown systems.

Without common mode failures, an estimate of the CRBR ATWS fequency could be arrived at by direct multiplication of the failure frequencies of the two shutdown systems as though they were totally independent. However, because of the potential for common mode failure it is not appropriate to attribute ATWS frequencies to CRBR as low (about 10^{-7} per year) as might be obtained by multiplication of the unreliabilities possible for the primary and secondary shutdown systems. Instead, to be conservative, a range of 10^{-5} to 10^{-4} per year was selected as a preliminary estimate for CRBR. Although we believe the most likely CRBR ATWS frequency to be on the low end of this spectrum, we have used 10^{-4} per year as the bounding value for the purpose of risk estimates in Appendix J.

Q10. How did you arrive at the specific CDA initiation frequency estimates attributed to loss of heat sink (LOHS) events?

A10. The frequency of LOHS is based in part on the redundancy and diversity of the CRBR decay heat removal systems and in part on the reliability of PWRs, which have redundancy and diversity in their auxiliary feedwater system (AFWS) similar to the CRBR Steam Generator Auxiliary Heat Removal System (SGAHRS). Evaluations of PWR AFWS reliabilities including that in WASH-1400 and more recent studies, suggest that failure frequencies in the range of 10^{-5} to 10^{-4} per demand may be achieved. The general trend of these studies is the basis for the conclusion that the CRBR SGAHRS can achieve similar reliability. Because CRBR also has a Direct Heat Removal Service (DHRS) to back up the SGAHRS, we believe the LOHS failure frequency will be below 10^{-4} per year. A formal reliability program at CRBR will add further assurance that this will be the case.

Q11. How did you conclude that the CDA initiation frequency from fuel failure propagation would be bounded by the ATWS and LOHS frequencies?

A11. The sodium coolant used to cool the CRBR core will operate far below its saturation temperature, and has a high thermal conductivity. Furthermore the coolant will move with a relatively high velocity through the assemblies. This means that local perturbations such as gas bubbles or debris particles will most likely be swept through the assembly instead of collecting and manifesting themselves as initiators for fuel pin cladding failures. Also if there are such perturbations, even including a release of fission gas from a pin with breached cladding, the efficient heat transfer and high subcooling provide protection against local fault propagation.

The predominant failure mode of stainless steel cladding under extreme hot spot conditions is the development of creep cavities and intergranular, through-wall cracks which lead to a gradual release of fission gases - as opposed to ballooning and bursting rupture of the cladding.

The spiral wire wrap fuel pin support system employed in the CRBR fuel assembly design provides a design which is less sensitive to debris collection and blockage formation than a grid-spaced pin support system.

As a result of the FERMI-I Reactor incident, particular attention has been directed at preventing by design, flow blockages due to debris choking off the inlet portions of the fuel assembly.

There are no stainless steel - sodium reactions (such as the exothermic zirconium oxidation by steam) that would provide a driving mechanism for propagation of failures.

To assure early warning of fuel cladding failures there will be a "tag gas" system. This is a system which can detect gas releases from failed fuel pins and quantify the presence of preloaded selected isotopes of xenon and krypton. For each assembly the pins are loaded with a unique blend of xenon and krypton isotopes which are released to the primary system if cladding failure occurs, and can then be detected.

As an additional means for fuel pin cladding failure detection in CRBR, a delayed neutron detection system will be used. This system monitors the sodium coolant for the presence of fission products which decay with the emission of neutrons. Delayed neutrons are detected when failed fuel pins with fuel in contact with the sodium permit fuel and fission products to be leached into the coolant stream. The reactor will be shutdown when significant delayed neutron levels are detected.

Quality assurance and quality control programs are to be employed for the manufacture of the CRBR fuel pins and assemblies, to assure that fuel with manufacturing defects will not be loaded into the reactor.

All these factors have been considered in arriving at the conclusion that fuel failure propagation at CRBR will be very unlikely, but if it does occur, the failures will be detected early enough to prevent propagation into a CDA. We believe that the probability of a CDA from such events is low because the design features of the fuel and coolant are inherent, passive measures, and because only a simple and inherently reliable detection system is employed.

Q12. How did you conclude that the CDA initiation frequency from loss of coolant accidents would be bounded by the LOHS frequency?

A12. CDA initiation resulting from uncovering the reactor core can be made highly improbable by requiring high integrity of the heat transport system. The principal measures to achieve this are to perform pre-service and in-service inspection of the primary coolant boundary to verify continuing piping integrity, and to install a detection system to detect small leaks, should they occur, before they grow to unacceptable size. Because LMFBR primary coolant systems operate at low pressure and below the saturation temperature of sodium, an Emergency Core Cooling System (ECCS) to rapidly inject additional coolant when a pipe break occurs is not necessary. Instead, it is sufficient to provide (a) guard vessels to catch coolant leakage from portions of the system below the top of the core to ensure sufficient core coverage and (b) piping elevated above the top of the core for other portions of the coolant system to preclude draining the reactor vessel. As the review progresses the Staff will make an evaluation regarding the details and adequacy of implementation of such design features. These conclusions will be reported in the SER. However, based on successful implementation of such features at LWRs or domestic and foreign LMFBRs, the Staff believes it will be possible to implement them acceptably at CRBR, and thereby assure that CDAs related to loss of coolant inventory will be very unlikely.

Because the design features such as guard vessels and leak detection systems required to assure that unacceptable loss of coolant will not occur are passive and/or do not require complex active components and systems, we believe their failure to be very unlikely, in comparsion to the estimated failure frequency for the shutdown system or decay heat transport system. Furthermore, the likelihood of a leak in the CRBR piping is also low. Therefore, we have concluded that the contribution of loss of coolant events to the frequency of CDAs is small compared to the contribution due to LOHS.

Q13. How did you conclude that the CDA initiation frequency from flow blockage would be bounded by the LOHS frequency?

A13. It is necessary to assure that a clear path for coolant flow to the fuel assemblies will be maintained. This will avoid a sudden flow blockage and damage to sub-assemblies such as occurred at the FERMI-1 reactor. It is possible to achieve this by including multiple coolant inlet ports at different planes and by interposing strainers in the flow path. Although high quality of fabrication will be required for CRBR, non-mechanistic deposits of debris or loose parts may be postulated. Flow blockage from such sources can be avoided by employment of core outlet thermocouples or loose parts monitoring systems to aid operators in diagnosing and correcting such conditions.

Because the design features such as multiple inlet ports and strainers required to assure that flow blockage does not occur, are passive and do not require complex active components and systems, and for reasons discussed in A11 of this testimony we believe that CDAs resulting from flow blockage are very unlikely to occur. We further conclude that the frequency of such CDAs is small in comparison to CDAs initiated by LOHS.

Q14. In the FES Supplement, Appendix J, Section J.1.2(1), in the paragraph summarizing the subsection titled "Initiators of Core Disruptive Accidents," the Staff has summed the frequencies of core disruption events and estimates a combined or net frequency of 10^{-4} per reactor year or less. Since 10^{-4} per reactor year or less was the estimated frequency of each of the classes of initiators, how did the Staff arrive at the conclusion that the sum of these is no larger than each of the individual contributions?

A14. The initiator class frequencies represent, in each case, a judgement that each frequency is no greater than 1×10^{-4} per reactor year and is expected to be appreciably smaller. Further, the scoping nature of this analysis is consistent with order of magnitude estimates of individual contributors. In each case, frequencies are rounded off to the next largest order of magnitude to obtain bounding estimates. Thus it is from the viewpoint that each class frequency is expected to be appreciably smaller than 1×10^{-4} per reactor year that the judgement is made that the sum of these frequencies is no greater than 1×10^{-4} per year.

Q15. Are the conditional frequencies of containment isolation failure and containment annulus cooling and vent-purge system failure based on detailed reliability analyses of the CRBR design?

A15. No.

Q16. How have these conditional frequencies been determined?

A16. They are based on the feasibility of the general CRBRP design achieving a specific level of reliability considering environmental factors, common mode failure and an appropriate level of reliability of required supporting systems and functions. In the case of the containment isolation system, LWR containments incorporate systems of similar function and design; thus bounding frequency estimates for CRBRP including environmental, support, and other interacting factors, can be made with sufficient confidence. In the case of the annulus cooling and vent-purge system, an equivalent level of LWR experience is not available. Thus confidence in the bounding frequency estimate is based upon the systems' inherent redundancy, diversity and independence as well as the feasibility of improving system performance, should this be deemed necessary, coupled with a reliability program and a testing and inspection program of sufficient frequency to provide the required reliability.

Q17. What releases of fission product and core materials, including halogens, iodine, and plutonium, from CDAs have been evaluated?

A17. The releases of fission product and core-materials from CDAs which have been evaluated are presented in Table J.2, Appendix J of the FES Supplement.

Q18. What accident processes are the basis for selection of the release fractions in Table J.2 of the FES Supplement?

A18. Release fractions are specified for CDA Classes 1 through 4 as indicated in Table J.2 of the FES Supplement. Each of these four sets of release fractions is based upon a specific accident scenario with regard to containment response and phenomenological events which occur after initiation of a CDA; however, for all CDAs it is assumed that the total noble gas inventory would be released from the containment building.

Estimates of the fraction of the core radionuclides released to the outside environment are made for each nuclide group identified in Table J.2 of the FES Supplement. These release fractions depend upon the fraction of each nuclide group released from the fuel, the primary system via the reactor vessel head, the sodium pool and subsequently the dry reactor cavity.

Initiation of a CDA may be followed by release of core materials, other radionuclides and sodium from the primary system through two modes. An immediate release through the upper reactor vessel head may occur if there is an energetic CDA resulting in mechanical damage to the upper reactor vessel head. Core materials and sodium not released into the Reactor Containment Building (RCB) through the upper head are released into the reactor cavity. Even if head release does not occur, it is assumed that the core debris will eventually melt through the bottom of the vessel resulting in deposition of core debris and sodium into the reactor cavity from which vents lead to the containment atmosphere. In the reactor cavity, the effects of sodium pool heatup, boiling and dryout as well as the potential for sparging of the remaining core debris via attack of the underlying concrete are also considered.

Release fractions of the fission products from the fuel after a CDA were conservatively selected considering core disruption phenomena and analysis of radionuclide releases in WASH-1400, Appendix 7, pp. 1-15, and the data provided in the document "Nuclear Aerosols in Reactor Safety, the State of the Art Report," Nuclear Energy Agency, OECD, CSNI/SOAR, No.1, June 1979, p. 228. The reactor vessel head release fractions were conservatively selected on the basis of judgement from consideration of general LMFBR research on energetic CDAs taking into account the relative volatilities of the different radionuclide species and other materials. Although the selection of head release fractions was not directly based upon a set of analytical calculations, review of CRBRP-3 helped in forming the judgement regarding the head release fraction values selected.

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The fission products, core materials and sodium not escaping through the reactor vessel head are assumed to drain into the reactor cavity shortly after the occurrence of a head release. Release fractions escaping from the sodium pool were conservatively estimated using experimental data and analyses reported in, for example, "LMFBR Safety - Fission Product Behavior in Sodium," Nuclear Safety, Vol. II, Sep.- Oct. 1970, pp. 379-390, by A. W. Castleman, and experimental data such as reported by W. Schutz and H. Sauter in, "Fuel and Fission Product Release and Transport From Hot Sodium Pools," Proceedings of the International Meeting on Fast Reactor Safety, Seattle, Aug. 19-23, 1979, pp. 1455-1464, and "UO_2- und Spaltproduktfreisetzung Aus Natriumlachen," Kernforschungszentrum Karlsruhe, KFK-3010, Nov. 1980.

Release fractions of fission products remaining in the reactor cavity after the sodium pool has boiled off were conservatively selected considering, for example, information regarding vaporization releases in WASH-1400, Appendix 7, and discussions of the subject in CRBRP-3.

The fractions of the materials which escape to the outside environment then depend upon their fallout rates (except for noble gases), the leakage rates of the containment atmosphere and the filter efficiency. The bases for data with regard to these phenomena are discussed later in this testimony.

Q19. What is the purpose of addressing the accident processes described in A18.

A19. The purpose of addressing the above accident processes in A18 is to provide a mechanistic basis from which to estimate the thermodynamic conditions in the RCB and reactor cavity as a function of time and thereby estimate the mass transfer of the radionuclides into the RCB and the environment as a function of time.

Q20. How are releases of radionuclides to the RCB estimated in the FES Supplement from the accident processes you have discussed?

A20. While the release of radionuclides into the RCB, following a CDA and primary system failure, is a continuously varying process, the estimates in the FES Supplement are based on simplification of this process into a series of discrete constant rate phases to expedite the analysis. The phases considered include:

- reactor vessel head release to initiation of sodium pool boiling
- initiation of sodium pool boiling to sodium pool dryout
- post sodium pool dryout

Release rates of radionuclides into the RCB are considered to vary significantly between these phases while they will remain relatively constant during each phase.

Releases to the RCB during the above three phases are therefore based on contributions from three sources: vessel head releases, pool releases and dry cavity releases. The head releases to the

Reactor Containment Building (RCB) are specified in Table J.3. The head releases in Primary System Failure Category III (moderate head releases) are conservatively used for CDA Class 3 of Table J.2. CDA Classes 1, 2 and 4 are conservatively assigned Primary System Failure Category IV (large) head releases.

Pool releases to the RCB depend on the relative volatility of the specific isotopes compared to that of the sodium as the sodium heats up and boils. All the I and Cs-Rb isotopes remaining in the pool are assumed to be released to the RCB. About 50% of the remaining Te-Sb, and Ba-Sr isotope groups are assumed to be released and none of the solid fission product groups (Ru and La) are assumed to be released to RCB during the pool boiloff process.

After cavity dryout, about 12% of the remaining Te-Sb, and Ba-Sr isotope groups (about 5% of their total inventory) and about 5% of the remaining Ru and La groups (nearly 5% of their total inventory) are estimated to be released to the RCB. These additional releases are assumed to occur as a result of sparging by gaseous products liberated during decomposition of the underlying concrete by the core debris.

Q21. What other information is required to estimate the releases to the environment?

A21. In addition to the timing and amount of sodium and fission products released into the RCB, leakage and venting rates of the atomsphere out of the RCB are also required to enable estimating releases to the environment.

Release of the atmosphere out of the RCB occurs through:

- leakage at the containment design basis limits,
- filtered venting,
- overpressure failure, or
- containment isolation failure,

depending on the CDA class under consideration.

For each CDA Class specified in Table J.2 of the FES Supplement and each RCB source term (head, pool, dry cavity releases), the mode of release from the containment (filtered or unfiltered) and rate, as well as the approximate sodium aerosol concentration in the RCB, are estimated.

Release from the RCB, considering CDA Class 1, involves leakage at design basis rates of 10^{-4} to 10^{-5} per hour followed by venting, both through filtration which is at least 97% efficient in removing iodines and 99% efficient in removing particulates. In CDA Class 2, approximately 57% of the RCB atmosphere will be released to the environs soon after the failure by overpressure, as the RCB pressure drops from about 2.3 atmospheres (abs) to 1 atmosphere (abs). Thereafter leakage through the steel containment shell breach is about equal to the release rates of fission products and other gases into the RCB (10^{-1} to 10^{-2} per hour). The leakage rate to the environment considering failure of the containment to isolate a ventilation supply or exhaust line (CDA Class 3 and 4) is estimated to be on the order of 10^{-1} to 10^{-2} per hour, similar to the rates after overpressure failure. Thus, for each release class the release of a volume of gases several times the containment atmospheric volume will occur during the estimated 100-200 hour time period in which the sodium pool boils.

Q22. How are estimates of the RCB source terms (discussed in A20) and estimates of leakage rates out of the RCB (discussed in A21) used to estimate releases to the environment surrounding the CRBRP?

A22. Using the estimates of RCB source terms and leakage rates of the containment atmosphere out of the RCB, the ratio of leakage rates to leakage plus fallout rates, as discussed below and in the FES Supplement, are estimated for each CDA Class and RCB source term. This ratio, when multiplied by the inventory fraction of each isotope in the RCB, results in an estimate of the fraction of each isotope released from the RCB. If filtering is operative, the filtering inefficiency (1 minus filter efficiency) is also multiplied by the release fraction to obtain the environmental release fraction.

Once the release fractions to the environment are calculated for each isotope group of each RCB source term, they are combined to form a total release fraction for each isotope group of each CDA class. Each CDA class environmental release represented by a set of isotope group release fractions is then used as input into the consequence model.

Q23. How are the environmental releases characterized for input into the consequence model, CRAC?

A23. The consequence model, CRAC, requires input data to characterize the environmental releases. Therefore, the environmental releases for CDA Classes 1 through 4 were assigned start and duration times as well as release heights and energy content. The values for each CDA Class are provided below.

CDA Class	Time of Release (Hours Since CDA Initiation)	Duration of Release (Hrs)	Height of Release (m)	Energy Content of Release (Calories)
1	24	10	60	0
2	24	3	0	0
3	0	40	0	0
4	0	30	0	0

These release and duration times are chosen such that the input environmental releases to the consequence model occur earlier and for a shorter duration than best estimate analyses would indicate. This is done to ensure that early and latent fatalities are not underestimated due to the use of these data.

The height and energy content of each release are also assigned conservative values with respect to early fatalities for the CRBR site. Use of these values tends to underestimate latent fatalities; the effect of increasing these input values (height and energy content) over a reasonable range is to increase the latent fatalities by about 50%. These conclusions were based upon CRAC sensitivity analyses and are considered valid within the range of uncertainties identified in A51 of this testimony.

Q24. What is the basis for stating in Appendix J that sodium aerosol deposition rates will be between 0.5 and 1.0 per hour?

A24. Several computer codes exist which model the temporal concentration of aerosols in a single chamber. An example of one of these codes is HAARM. A modification of HAARM, HAARM-3, for example, has been benchmarked against experiments and run parametrically within the ranges of interest for the CRBRP to determine typical sodium aerosol behavior (See NUREG-1989). It is observed from these studies and studies using other codes, that deposition rate constants vary over a fairly narrow range for most of the suspended aerosol mass concentrations and generation rates of interest.

For the purposes of the calculations regarding aerosol deposition in the FES, therefore, values between 0.5 and 1.0 per hour were chosen as representative deposition rate constants for the suspended sodium aerosol concentrations anticipated.

Q25. What is the basis for the Staff's assumption that "because there are more than one million pounds of primary coolant sodium a dense aerosol (10-100 micrograms/cc) could be airborne in the RCB"?

A25. Repeated experiments with sodium fires have failed to produce aerosols of concentration as high as 100 micrograms/cc. At these concentrations the deposition rate becomes so high that concentrations cannot be further increased. The behavior of an aerosol resulting from a large sodium fire is, for example, illustrated in BMI-NUREG-1989, Figure 8

(p. 41). Thus 100 $\mu g/cm^3$ (micrograms/cc) is conservative for the characterization of the upper limit of aerosol concentration resulting from a large sodium fire. During the period in which the pool is boiling, on the order of 5000 kg/hr of sodium will be released into the RCB atmosphere. This source rate is consistent with an airborne concentration of 10 $\mu g/cm^3$ or greater. For example, the steady state airborne concentration is simply $S/V\alpha$, where S = average source rate in $\mu g/hr$, V = containment volume $10^{11} cm^3$, and α = removal rate hr^{-1}. Using a value of S equal to 5×10^{12} $\mu g/hr$ (which is consistent with vaporization of the sodium pool of over one million pounds in 100 hours) and α = 1. per hour, yields a steady state airborne concentration of 50 $\mu g/cm^3$. The Staff does not rely on any specific documents for its judgement of the aerosol concentration. NUREG-1989 mentioned above is one example of a document that supports this range.

Q26. What is the basis for the Staff's estimate that "in CDA Class II, approximately 57% of the RCB atmosphere will be released soon after failure by overpressurization because the RCB pressure drops from about 2.3 atmospheres (abs) to one atmosphere (abs)?"

A26. Considering normal design margin required by code requirements it was estimated that, conservatively, the containment vessel should hold at least twice its design basis pressure of 10 psig; this is about 2.3 atmospheres (abs). If overpressurization failure produced a leak of sufficient size, the containment vessel atmosphere would vent down to atmospheric pressure, releasing about 57% of its gaseous contents. These are rough estimates; a larger fraction could be so vented if the containment held to higher pressure, and a smaller fraction might be so vented if the failure produced a more limited leak.

Q27. What is the basis for the Staff's estimate that "the leakage rate to the environment considering failure of the containment to isolate a ventilation supply or exhaust line (CDA Classes III and IV) is estimated to be on the order of 10^{-1} to 10^{-2} per hour, similar to the rates after overpressurization failure?"

A27. The estimate is based on the condition that since there will be no appreciable obstruction to the flow of gases between the containment and the environment, the RCB will be close to atmospheric pressure upon failure of the containment system to isolate a ventilation supply or exhaust line (or after the overpressurization failure has relieved the RCB pressure). Therefore in this condition, the leakage rate from the RCB to the environment will depend on the volume and thermal energy input into the RCB from the reactor cavity and from the head release (containment isolation failure case). Leakage rates in the range of 10^{-1} per hour adequately bound those anticipated for the head release, and leakage rates around 10^{-2} per hour adequately bound leakage rates anticipated for the (longer term) pool releases. While no documents were specifically relied upon for these estimates, CRBRP-3, Volume 2 (pages 3-103 and 3-173) indicates that these estimates are conservative.

Q28. What is the basis for the Staff's estimate that sodium boiling will occur in a 100-200 hour period and not a longer or shorter period?

A28. This comes from consideration of a simple heat balance taking account of the decay heat and other input energy sources. While no specific documents were relied upon to make this estimate, it can be compared to CRBRP-3 (pages 3-21 and 3-25) for example, which indicates a boiling period of about 120 hours.

Q29. How did the Staff perform the calculations that form the basis for the statement that "considering leakage rates between 10^{-2} and 10^{-1} per hour, therefore indicate that between 1% and 20% of the particulate airborne fission products may eventually be released to the environment?"

A29. Consideration of differential equations for the removal rates, involving terms of the e^{-Rt} type, leads to the (leaked) release fractions being the result of ratios $R_1/(R_1 + R_2)$ where R_1 is "leakage rates between 10^{-2} and 10^{-1} per hour" and R_2 is "deposition rates in a single chamber of between 0.5 and 1.0 per hour"; the results range from 1% to 20%.

Q30. What is the analytical bases for the use of design basis leakage rates of 10^{-4} to 10^{-5} per hour corresponding to 10^{-3} to 10^{-5} long-term release fractions and for assuming filtered venting is 97% to 99% efficient?

A30. The containment design basis leakage rate of 0.1% by volume per day at 10 psig is about 4×10^{-5} per hour. The analysis for the long-term release fractions is parallel to that given in the response to Question 29, the resulting range is 2×10^{-4} to 1×10^{-5}. The Staff has estimated filter efficiencies in both design basis leakage filtering and filtered venting at 97% for iodines and 99% for particulates. For the final CRBRP design, filters will be required that can withstand the environmental conditions and achieve such efficiencies. The Applicants have shown the Staff results of scaled tests based on a system like the proposed filtered venting system; the results provide the required efficiencies.

Q31. What is the basis for the Staff's assumption that a release to the environment would not occur until about 24 hours after the head release and about 14 hours after pool boiling begins?

A31. Heat balance estimates indicate that boiling begins at about 9 hours. Pressurized hydrogen would increase in the containment building at rates dependent on the rate of sodium boiloff and sodium concrete reactions. The Applicants' analysis indicates that filtered venting and cooling should begin at about 36 hours (CRBRP-3). Based on the Staff's knowledge of the possibility of sodium concrete reaction rates greater than that assumed by the Applicants, we have selected 24 hours as a reasonable estimate of the time at which venting and cooling would be necessary. We have assumed that one of the active systems would fail to function, causing immediate containment failure at 24 hours.

Q32. Mr. Thadani and Dr. Rumble, what are the sodium isotopes and their inventories in the primary system?

A32. During operation, the activation product isotopes Na^{24} and Na^{22} of the alkali metal, sodium, used in the primary system, are anticipated to reach total activity levels on the order of 1.7×10^7 and 10^3 curies respectively. The half-lives of these isotopes are 0.63 and 956 days respectively. (Other radioisotopes of sodium have half-lives of seconds and are not considered significant).

Q33. Mr. Thadani, was radioactive sodium included in the calculation of risk?

A33. Radioactive sodium was not included in the evaluation of accidents presented in the Draft FES Supplement. Sodium-24 has, however, been included in the results of the evaluation of postulated accident risks presented in the Final FES Supplement, (in response to comments made on the Draft Supplement).

If sodium is to be released to the environment in the course of a core disruptive accident, there also must occur release of other fission products, especially the noble gases and the chemical species of equal or greater volatility. Thus, if sodium were released, one would anticipate, for example, accompanying releases of xenon, krypton, iodines, rubidium, and cesium.

Comparison of the approximately 10^3 curies of sodium-22 which could potentially be released if all the primary system sodium escaped to the environment, with the available inventories of other accompanying radionuclides (see Table J.4 of Appendix J to the FES Supplement), shows that sodium-22 is an insignificant radiological contributor.

Q34. Mr. Thadani and Dr. Rumble, what were the effects of including sodium-24 in the consequence analysis?

A34. The consideration of sodium-24 which we have included in the consequence analysis leads to a small increase in the early fatality risk estimate and no increase in the latent fatality risk estimate, over similar estimates made without sodium-24. This increase is considered not significant since it was estimated using conservative sodium-24 release fractions (larger than those for cesium) and using cesium-136 (which is more radiotoxic and has a longer half life than sodium-24 for the whole body and all organs except the thyroid; the thyroid dose is, however dominated by iodine doses) as a surrogate for sodium-24. No additional effort was expended to reduce these conservatisms regarding the treatment of sodium-24 since its contribution to the risk is demonstrated to be not significant. In the consideration of sodium-24 in our analysis, we did not account for the agglomeration of the sodium aerosols after release from the containment with consequent rapid settling of sodium and fission product aerosols near the site that would cause a reduction in the offsite health effects. This effect could reduce the estimated risks to the public under the low wind speed conditions common at the CRBR site.

Q35. Drs. Morris, Swift, Long and Rumble, how has the capability of the containment design for reducing calculated offsite doses been factored into the environmental analysis?

A35. Table J.2 of the FES Supplement provides a list of CDA sequence classes which are used as input to the consequence calculation. The calculated health effects for each of the four CDA classes are added together to determine the integrated effects from all the CDA classes.

In the Class 1 CDA accidents, we postulate generic core disruption and successful operation of the containment. The core inventory released to the environment for this CDA class as shown in Table J.2 is relatively small since the containment functions as designed, to hold up the accident effluent release for 24 hours, and when release occurs, it occurs through an efficient filtering system.

Q36. Has the containment been assumed to function perfectly in all scenarios?

A36. No. Table J.2 of the FES Supplement indicates that in the three less frequent CDA classes of the four considered, the containment is considered to fail due either to overpressure or to failure of containment isolation. The conditional probabilities of each of these two modes of containment failure is assumed to be 10^{-2} per demand.

Q37. Drs. Morris, Swift, Long, and Rumble, and Mr. Thadani: have any computer codes or other calculational tools been employed to evaluate the frequency of CDA energetics for the purpose of performing the NEPA evaluation?

A37. No. Computer codes and calculational tools have not been directly used to evaluate CDA energetics for the NEPA evaluations of CRBRP. There has been for many years, however, a large effort directed toward analyzing CDA energetics in general in the United States and in foreign countries. General results for a spectrum of sizes and types of fast reactor designs have formed a collection of background information which is part of the basis for evaluating CDA energetic frequencies.

Q38. What, if any, computer codes have been used for the environmental analyses relating to CDA consequences?

A38. The consequence model used in the RSS (NUREG-0340) called CRAC was adapted for the CRBRP site and used to calculate the atmospheric pathway risks (consequences times probabilities). This code is discussed in greater detail under the section of this testimony discussing the risks to the public.

Q39. Drs. Morris and Rumble, if computer analyses of CDA energetics have not been employed, what confidence do you have that the risks from CDAs have been properly evaluated in the FES?

A39. The present body of information regarding the energetics resulting from physically reasonable core rearrangements of sodium, cladding, or fuel indicates that the magnitude of such energetics is well within the containability range of the primary system. If after completion of the Staff review of the potential for core associated energetics, a conclusion is reached that energy releases beyond the primary system capability to maintain sufficient integrity cannot be precluded, the Staff will require design modifications to prevent early containment failures from such effects as missiles or spray fires. It is the Staff's judgement that such modifications are clearly feasible and not so costly as to significantly affect the overall cost-benefit balance. Thus the releases from CDAs as indicated in Table J.2 of the FES Supplement do not include early containment failures from extremely energetic CDAs since they will be of sufficiently low likelihood that their contribution to the risk to the public will not be significant.

Head releases for those CDA Classes analyzed in the FES Supplement are presented in Table J.3. These releases are selected to approximate potential bounding head releases for two different levels of energetics, given the design of the primary containment system and potential variations thereof. While these releases are not derived from specific analyses of the CRBR, they have been selected on the basis of the ranges of such releases that have been estimated for CRBR and other plants. Further, the releases of different isotope groups were set relative to each other to account for the spectrum of volatile species present in the core inventory. Thus, for example, more cesium and rubidium would be expected to be released from the head for a given energetic CDA than tellurium or antimony.

Q40. How reasonable are your estimates of CDA radionuclide releases to the environment?

A40. From the background information available regarding energetics and from a design feasibility standpoint, it is the Staff judgement that these release values are appropriate and probably conservative (based upon NUREG-0772). The sensitivity of these values was tested by considering variations in these head release fractions, using the CDA classes as defined in Table J.2. As described later in this testimony, this sensitivity test did not significantly affect the risk with regard to its impact in the NEPA cost/benefit analysis.

Q41. If the head release fractions (other than Xe-Kr) in Table J.3 were higher, how would this affect the releases to the environment as shown in Table J.2?

A41. In each of the four CDA Classes presented in Table J.2, failure of the vessel bottom by melt-through is assumed to occur shortly after the head release. Subsequently, as the sodium boils in the reactor cavity the remaining volatile and semi-volatile fission products are released to the Reactor Containment Building (RCB). Following boiloff of the sodium, release of solid fission products and fuel is then possible due to reactions of the core materials with the underlying concrete. This process is assumed to also release significant fractions of radioactive material to the RCB. Thus the total fraction of the core radionuclide inventory released to the RCB is not very sensitive to the head release fractions in Table J.3. Therefore, differences in environmental releases are due mainly to the time in which the released core inventory enters the RCB either from the head or from the reactor cavity.

Since the containment systems function as designed in CDA Class 1, the radioactive nuclides are held within the RCB and only partially (with the exception of noble gases) released through efficient filters. Thus CDA Class 1 releases to the environment are insensitive to the magnitude of the head releases.

CDA Class 2 includes a containment overpressure failure, in which 57% of the RCB atmosphere is released to the outside environment at about 24 hours after the accident. Since deposition inside the containment of the radionuclides released from the head will occur during the 24 hour period prior to containment failure, CDA Class 2 releases are insensitive to the magnitude of the head releases. Environmental release fractions of the Te-Sb and Ba-Sr groups would increase by 40% if the entire inventory entered the RCB during the head release, since 60% of these isotopes are released to the RCB in the FES analysis. Environmental release fractions of the Ru and La groups would increase no greater than linearly with increases in their head release fractions, because of the deposition of these isotopes during the 24 hour period before containment failure.

CDA Classes 3 and 4 represent CDAs followed by failure to isolate containment. The former class incorporates the head release fractions associated with Primary System Failure Category III (moderate head release) and the latter with Primary System Failure Category IV (large head release). In these classes, the energy (associated with the head release) deposited into the RCB causes a surge in the leakage rate to the environment. This leakage subsides to leakage rates comparable to long term leakages which occur in CDA Class 2 after the overpressure failure. The sensitivity of CDA releases to head releases is evident by comparing the environmental releases of CDA classes 3 and 4 in Table J.2 and the head releases of Table J.3. A factor of 10 increase in the head release values for the I, Cs-Rb, Te-Sb and Ba-Sr groups in Table J.3 results in a factor of 2 to 3 increase in their environmental releases shown in Table J.2. A factor of 20 increase in the head release of the Ru and La groups in Table J.3 results in a factor of about 6 increase in the environmental releases.

Q42. What are the Staff's conclusions regarding the likelihood of occurrence of those accidents capable of causing head releases?

A42. The Staff has determined that less than one in ten CDAs are energetic enough to cause primary coolant system seal failure.

Q43. More specifically, how has it been determined that less than one in ten CDAs are energetic enough to cause primary coolant system seal failure?

A43. The judgement that not more than one in ten CDAs could be energetic enough to cause primary coolant system seal failure is based on the present body of knowledge and the capacity of the primary system to withstand mechanical damage. The Staff is reviewing the potential for energetic recriticalities to determine the magnitude of energy release anticipated. The frequency of one in ten, therefore, is set conservatively as a reflection that some uncertainty remains with regard to energetic recriticalities.

More specifically, the Staff's estimate that the conditional probability of primary system failure Category IV is 0.1 was based on two points. First, for simplicity a single general CDA initiation frequency of approximately 10^{-4}/year which included the combined frequencies of various specific CDA initiators was used. However, the specific CDA initiators do not have equal potential for resulting in an energetic CDA. The fraction, 0.1, was therefore in part employed to compensate for this simplification. Second, the Staff's general knowledge of and experience with the extensive research on the phenomena that may occur in a core disruptive accident has led to the conclusion that energetics large enough to cause a Category IV (large head release) type failure are relatively unlikely to occur even if a CDA is initiated. Factors which have been considered in this conclusion are that (a) incoherent fuel failures and material rearrangement are more likely than the coherent behavior associated with high energetics, (b) small criticalities which disperse fissionable material without significant energetics are more likely than large energetic criticalities, (c) the heterogeneous core design slows down power escalations due to voiding and minimizes the potential for rapid reactivity insertion due to fuel motion, and (d) the upper internals structures have an effect in mitigating CDA generated forces.

Q44. How have accidents involving core meltthrough and sodium-concrete reactions been included in the environmental analysis?

A44. Table J.2 of the FES Supplement provides a description of the CDA Classes analyzed. In each Class, thermal failure of the lower reactor vessel head, subsequent release of the sodium and core materials into the reactor cavity, and cavity liner failure as well as heatup, boiling and dryout of the sodium pool in the reactor cavity are assumed. During this process sodium-concrete reactions take place adding heat and reaction products to the sodium pool.

These phenomena influence the amount and timing of radionuclide releases to the Reactor Containment Building (RCB) and thus the releases to the environment. Further, when the containment annulus cooling and vent/purge systems are assumed to fail (CDA Class 2), an overpressure failure of the containment occurs due in part, to the energy and gases generated from the sodium-concrete reaction.

Q45. How have the ways that human error could initiate, exacerbate, or interfere with the mitigation of CRBR accidents been factored into the NEPA cost/benefit analysis, and specifically into the determination of CDA initiation frequencies and conditional probabilities of containment isolation failure or containment cooling and venting systems failure?

A45. In each case where a system or function frequency is estimated, consideration has been given to the manner in which operators or maintenance personnel may interact with them. Relative assessments are then made comparing the complexity, stress level, time available, potential for confusion and other related factors with situations in LWRs, where appropriate. For example, the reactor protection and containment isolation systems have comparable operator and maintenance personnel interfaces and thus the judgement can be made that the contribution of human error to their unavailability is similar. In other instances, for example, loss of heat sink accidents, more time may be available in specific scenarios for the operator to respond at CRBRP than in the case of LWRs. Typically, however, no additional credit was given in the quantification of the associated frequency for such perceived beneficial factors. Instead, the approach was to use for the CRBRP situations, which had equal or better conditions versus comparable LWR situations, the same level of human error induced unavailability estimates as is generally used for LWRs.

Q46. Mr. Thadani, what does this part of the testimony regarding computer codes address? What is the purpose of your testimony?

A46. The purpose of my testimony is to briefly describe the model and the code used to evaluate the consequences of atmospheric releases resulting from severe accidents postulated in the Draft and Final FES Supplement, and to show that the model used in the analysis is adequately documented, and reasonably verified. My testimony will further show that the data used as input to the code are also adequately documented and verified. I will also discuss some of the results obtained from my analysis, and the uncertainties associated with them.

Q47. What models and/or computer codes were used by the Staff to calculate the risks to the public due to CRBRP core disruptive accidents (CDAs)?

A47. The Staff used the consequence model described in the Reactor Safety Study, RSS, (WASH-1400, NUREG-75/014) and the associated computer code "CRAC", adapted and modified to treat the CRBRP reactor core characteristics and the CRBRP site features.

Q48. Is the RSS consequence model documented? If so, where?

A48. Yes. The consequences model is documented in Appendix VI of the Reactor Safety Study (NUREG-75/014), and in "Overview of the RSS Consequence Model" (NUREG-0340).

Q49. Is the "CRAC" computer code documented? If so, where?

A49. Yes. The code is not documented in a listed form. However, it exists in the form of computer tapes which are maintained by Sandia National Laboratory.

Q50. How was the CRAC code validated?

A50. The computer models are generally validated by (1) performing programming and modeling error checks, (2) verification of the parts of the code for which answers are either known or can be generated by hand calculations, (3) empirical verification of the phenomena modeled by the code.

The CRAC code has been thoroughly checked for modeling and programming errors by many users world-wide. An International Benchmark Comparison of Reactor Accident Consequence Models program was devoted to performing verification of the parts of the code and to comparison with the other similar international codes. The benchmarking program was carried out under the aegis of the Nuclear Energy Agency's Committee for the Safety of Nuclear Installations (1981). The empirical verification of the code as a complete unit cannot be done because of the nature of the problems modeled by the CRAC. (The results would have to be compared with the consequences of an actual severe accident none of which have ever occurred.)

The International Benchmark Comparison study observed the following with respect to certain models which are employed in the CRAC code:

1. The Gaussian representation of the distribution of the radioactive material in the plume is generally acceptable and has been used in all international models.
2. The ICRP methodology of dosimetric calculations have been generally accepted in all international models, yielding good agreement in the dose calculations.

3. Linear relationship between dose and risk of latent cancer incidence has been generally accepted.

4. The bone marrow dose relationship to mortality used by the participants differ significantly. The differences appear to arise from imprecision of the statistics of the available human data, as well as from the varying assumptions of medical treatment.

Q51. Mr. Thadani, what are the uncertainties associated with the "CRAC" code, and what effect do the uncertainties have on the consequence outcome calculated by "CRAC" code?

A51. The uncertainties in the calculations performed by the CRAC code are discussed in the Final FES Supplement in Section J.1.2(6). The state-of-the-art for quantitative evaluation of the uncertainties in the probabilistic risk analysis is not well developed. Therefore, although the Staff has made a reasonable analysis of the CRBRP risks resulting from CDAs, there are large uncertainties associated with the results presented in the FES Supplement. It is my judgement that the overall uncertainties (including the uncertainties in the probabilities and the release fractions) in the results of "CRAC" code calculations of CRBRP accident consequences would probably be larger than 10, and may even be as large as 100, but would probably not exceed 100. These estimates are based upon judgements stemming from CRAC sensitivity studies for both CRBR and LWRs.

Q52. Describe the input data that were used to evaluate the consequences of CRBRP accidents in Appendix J of the FES Supplement.

A52. The inputs to the "CRAC" code consisted of (1) inventories of radionuclides in the CRBRP Core, (2) fraction of radionuclides estimated to be released from CRBRP to the atmosphere, (3) duration of the release of radionuclides to the atmosphere, (4) amount of energy released to the atmosphere, (5) probabilities of the estimated releases, (6) the site meteorology, (7) the site population distribution, and (8) the site emergency planning parameters.

Q53. Mr. Thadani, were the input data for the CRBRP CRAC run validated or their uncertainties accounted for in the analysis?

A53. Yes. Input data concerning items (1) through (5) in A52 were discussed previously in this testimony by Drs. Morris , Swift and Rumble, and as indicated there, we have performed a sensitivity analysis of the risks to public by increasing the source terms (other than noble gases) by a factor of 3, and obtained a measure of the sensitivity of the consequences to the uncertainties in the release fractions. The meteorological data and the site population data were obtained by the Staff specifically for the CRBRP site, while the evacuation parameters were conservatively selected to be more adverse than those used by the Staff at other sites.

The meteorological data selected for use in the CRAC analysis for CRBRP were based on measurements from a 110 meter permanent meteorological tower located at the proposed CRBR site. One year of data was used in the CRAC analysis, covering a period from February 17, 1977 to February 16, 1978, and represents the latest available information on CRBRP meteorology. A detailed description of the data is discussed in the FES Supplement, and in testimony by Mr. Spickler regarding Contentions 5a and 7c.

The site population distribution data were based on the Applicant's information and 1970 census data. The population distribution information up to 50 miles from the plant were based on Staff verification of the Applicant's data by checking against 1980 census data. Beyond 50 miles, the Staff used the 1980 census data. All population data were updated and modified by using the Bureau of Economic Analysis methodology for projecting the population estimates to the year 2010. A detailed discussion of the population data is provided in the Staff testimony of C. Ferrell regarding Contentions 5a and 7c.

The Staff has conservatively assumed a 12 hour delay in the start of the evacuation at the CRBRP site. This is a very conservative assumption when compared to a 1 hour delay assumed by the Staff at LWR sites. The Staff's conservatism is provided to account for the special facilities near the proposed CRBRP site. An evacuation speed of 1 meter per second (2.2 miles per hour) used in my analysis is within the range of values used by the Staff in its LWR risk analyses using the CRAC code. The Staff's analysis is based on an evacuation radius of 10 miles from the proposed plant location. Although the evacuation parameters selected for the

CRAC analysis are reasonable ones, they do not replace the Staff's review of the Applicant's emergency preparedness plans required by the provisions of 10 CFR 50.47(b). Such a review will be performed by the Staff and the acceptability of the site emergency preparedness plans will be reported in the Staff's Safety Evaluation Report.

The CRAC consequence model used by the Staff does not, at present, account for the consequences of sodium-24. In lieu of code modifications, I have therefore, used a surrogate from the list of nuclides incorporated in CRAC model to represent sodium-24. A comparison of the dose conversion factors of sodium-24 with those of the other alkali metals in the CRAC model showed a good comparison between the radiotoxicity of sodium-24 and cesium-136. Accordingly, the consequence analysis was performed by increasing the cesium-136 source inventory by the amount equivalent to sodium-24 in the CRBRP coolant.

Q54. What are the results of your analysis of the public risks from the postulated CRBRP accidents?

A54. The results of my analysis indicate that the risks to the public from the postulated CRBRP accidents would be comparable to the risks calculated by the Staff for light water reactors. The bases for this conclusion include sensitivity studies involving CRAC calculations for a PWR or BWR at the CRBR site, and CRAC generated risk estimates incorporated in Environmental Statements for contemporary LWRs at other sites.

Q55. Mr. Thadani, how important is the consideration of sodium-24 in the analysis of CRBRP accident risks to the public?

A55. The results of my analysis indicate that the radioactive sodium release does not significantly increase the calculated CRBRP accident risks to the public. The aerosol agglomeration effects of sodium, however, are expected to further reduce the quantity of radionuclides released to the offsite environs in an accident involving sodium release, over what I have estimated. Because there is limited information on the behavior of sodium aerosols in the outdoors atmosphere, I have conservatively not included the reduction of risk that could result from the agglomeration characteristics of sodium.

Q56. What would be the effect on estimated risk to the public as a result of including the sodum-24 consideration and of varying some of the input parameters in the CRAC analysis?

A56. I examined the influence on estimated risks of radioactive sodium and the variation of the elevation of the release from the containment, the time required to evacuate the site region after a severe accident, evacuation speed, and the percentage of core fractions released to the atmosphere. The results of my analysis indicate that the inclusion of sodium-24 to the postulated accident radionuclide releases does not increase the risks to the public significantly. The computed early fatality risks increased only slightly, and the latent fatalities did not change. The small increase in early fatalities is not considered significant in view of the conservative assumptions regarding release fractions and the radiotoxicity of sodium used in the analysis. A reduction of the evacuation delay time from 12 hours to 1 hour results in no early fatalities. Reduction of the evacuation speed from 1.0 meter/sec to 0.5 meter/sec for a 1 hour evacuation delay results in early fatalities which are a factor of 100 below the case of 12 hour delay in the evacuation and evacuation speed of 1.0 meter/sec. A threefold increase in all radionuclide release fractions except noble gases (100% of the noble gases were used in all cases) results in about a thirteen fold increase in early fatalities; a reduction of the source term to a third of all postulated release fractions except noble gases results in a factor of 3 reduction of the early fatalities and a factor of 2 reduction in the latent fatalities.

Q57. In addition to the estimated average annual risks to the public, what would be the distribution of the probability of the impacts of early fatalities, latent fatalities, and economic costs assocated with offsite mitigation actions?

A57. The Final FES Supplement presents in Appendix J, in Figures J.2, J.3 and J.4, the probability distribution of the early fatalities, the latent fatalities, and the economic costs associated with the offsite mitigation action. These results indicate that if one early fatality were to occur as a result of a severe accident, there is an approximately equal likelihood that about 10 early fatalities would occur. The probability of substantially more fatalities, however, drops by orders of magnitude and there would probably be 1 chance in 10 billion per year that 30 or more early fatalities might occur. Similarly, there is about one chance in a billion per year that there would be about 1000 latent fatalities as a result of a severe accident. The results also show that at the extreme end of the offsite mitigation costs spectrum the costs could be as high as several hundred million dollars.

Q58. Gentlemen, what are your conclusions with regard to Contention 2, in particular 2.c?

A58. We have concluded that the analyses of CDAs and their consequences as described in the Supplement to the Final Environmental Statement (NUREG-0139) meet all the requirements for environmental impact considerations under NRC regulations and policy, and under the National Environmental Policy Act, for the description of such impacts and performing the NEPA cost/benefit analysis, and are totally adequate for such purposes. The radiological source term analysis has adequately considered the possible releases of fission products and core materials, and also the potential environmental conditions in the reactor containment building created by the possible release of substantial quantities of sodium. Staff has adequately considered the potential release of sodium following a CDA, including the possible range of quantities released, and has considered the environmental conditions caused by such a release in the analysis of radiological consequences.

Q59. Gentlemen, what are your conclusions with regard to Contention 2.d?

A59. From our evaluation of the proposed containment design for CRBRP, we have concluded that the proposed containment system, or suitable feasible modifications thereof, can adequately reduce calculated offsite doses to an acceptable level, and that it can serve adequately toward keeping the risks from the CRBRP comparable to, or better than, the risks from current LWRs.

Q60. Gentlemen, what are your conclusions regarding Contention 2.f?

A60. We have concluded from the Staff's evaluation of the physical phenomena and principles which control the response of CRBR to CDAs, and of the models of them used for the environmental review, that the Staff's analyses have established that the models do represent the physical phenomena and principles controlling the response of CRBR to CDAs with accuracy adequate for all purposes involved in the NEPA review. We have concluded that the models and computer codes used in the Staff safety analyses of CDAs and their consequences for the NEPA review have indeed been, for such purposes, adequately documented, and verified, and validated by comparison with applicable theory and experimental data. Finally, in this regard, we have concluded that the computer models (including computer codes) referenced in the Staff accident analyses for the NEPA review are valid.

Q61. Gentlemen, what are your conclusions regarding Contention 2.g.?

A61. We have concluded that the Staff has established sufficiently for all purposes of the NEPA review that the input data and assumptions for the computer models and codes involved are adequately documented and verified.

Q62. Gentlemen, what are your conclusions regarding Contention 2.h?

A62. We have concluded that, in the context of the analyses for adequately describing the environmental aspects of CRBRP for the NEPA review, this contention has no validity. The Staff has established that the proposed primary system and containment system designs provide sufficient containment function capability, taking into consideration the feasibility of modification if further enhancement of the containment is necessary, to assure that the analyses of radiological consequences of accidents as presented in the NEPA review are valid and provide the descriptions and analyses needed to meet NEPA and other Federal regulatory requirements for such purposes.

Q63. Gentlemen, what are your conclusions with regard to Contention 3, in particular 3.c?

A63. We have concluded that the Staff has given sufficient attention to CRBR accidents other than the DBAs, i.e., that the Staff has evaluated, adequately for the NEPA review, possible CRBR accidents other than DBAs, as evidenced in the FES and its Supplement. Furthermore, as part of that effort, the Staff has given considerable attention to accidents associated with core melt-through following loss of core geometry and sodium-concrete interactions, and we have concluded that, for all NEPA review, the Staff has adequately analyzed such accidents.

Q64. Gentlemen, what are your conclusions with regard to Contention 3.d?

A64. We have concluded that, in the Staff's evaluation of the full range of accidents possible at CRBR, including the initiation, control and mitigation of accidents, the Staff has, for the purposes of environmental review, adequately identified and analyzed and given due consideration to the ways in which human error can initate, exacerbate, or interfere with the mitigation of CRBR accidents.

PROFESSIONAL QUALIFICATIONS

OF

MOHAN C. THADANI

I am employed as a Nuclear Engineer in the Accident Evaluation Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation. My responsibilities include the reviews and the analyses of designs and operations of nuclear power plant systems to determine the acceptability of the plant safety and the environmental impacts.

I graduated from the University of Bombay in 1955, with a Bachelor of Science (Honors) degree in Chemistry and Physics. I received a post-graduate diploma in Chemical Engineering from the University of London. Subsequently, in 1964 I received a Master of Science degree in Chemical Engineering from the University of Tennessee. In 1957, I joined the Nuclear Power Division of Head Wrightson and Company in Stockton-On-Tees, England. I was assigned to the thermal and hydraulic design and analysis of the Bradwell Nuclear Power Station in England.

In 1959, I joined the Foster Wheeler Limited of London, England. I was assigned to the research department on the design and testing of heat exchange components of the Pressurized Water Reactors for the British submarines.

From 1964 to 1970, I worked for the aerospace companies, Northrup Space Laboratories, Grumman Aerospace Corporation, and Fairchild Industries. I performed thermodynamics and reliability analyses for the Apollo Saturn Launch Vehicles, NERVA nuclear rocket systems, Lunar Module, Earth Orbital Shuttle Systems, and several satellite systems.

In 1971, I joined NUS Corporation as a senior engineer responsible for preparation of safety and environmental evaluations for nuclear power plant systems. While at NUS, I attained progressively increasing responsibilities, being promoted to the positions of section leader, and senior staff consultant. I was assigned as a project manager for the preparation of Safety Analysis Reports and Environmental Reports for Construction Permit and Operating License Applications for Nuclear Power Plants.

In 1978, I joined Teknekron, Incorporated, as a Senior Scientist and served as a Principal Investigator for analyses and evaluations to guide and support the development of Nuclear Regulatory Commission's proposed rule 10 CFR 60 concerning the safety of the geologic isolation of high level nuclear wastes.

In April 1980, I joined the Nuclear Regulatory Commission as a Nuclear Engineer in the Environmental Evaluation Branch, Division of Operating Reactors, Office of Nuclear Reactor Regulation. Following a reorganization of the Office of Nuclear Reactor Regulation, I was assigned to my present position as a Nuclear Engineer in the Accident Evaluation Branch, Division of Systems Integration.