11/1/82

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE 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

JOINT TESTIMONY OF CHARLES FERRELL, HOMER LOWENBERG
LEONARD SOFFER AND IRWIN SPICKLER
ON CONTENTIONS 5(a) AND 7(c)

Q.1. Mr. Ferrell, please state your position, your employer, and the nature of your work?

A.1. I am employed by the U.S. Nuclear Regulatory Commission ("NRC") as a Site Analyst in the Siting Analysis Branch, Division of Engineering. My duties include the evaluation of the reactor site, exclusion area contol, population and nearby industrial, transportation and military facilities. A statement of my professional qualifications is attached to this testimony.

Q.2. What is the nature of your responsibilities regarding the Clinch River Breeder Reactor?

A.2. I was the Site Analyst assigned to the Clinch River Breeder Reactor Project. I was responsible for, or contributed to, the review of the exclusion area, demography, off-site transportation, and industrial and military facilities for CRBR and alternate sites for CRBR. These reviews and contributions are in Sections III A, III B and III C of the Clinch River Breeder Reactor Site Suitability Report ("SSR"), Sections 2.1 and 2.2 of the Applicant's Environmental Report ("ER"), and Section 9 and Appendix L of the 1982 Supplement to the Final Environmental Statement ("FES") for CRBR ("FES Supplement").

Q.3. Mr. Lowenberg, by whom are you employed, and what is your position; and what is the nature of your work?

A.3. My name is Homer Lowenberg, Chief Engineer for the Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission ("NRC").

I am a graduate of Stevens Institute of Technology with degrees in mechanical and chemical engineering and am a professional engineer in the states of Pennsylvania and New York. I have over 25 years experience in the commercial design, construction and operation fields related to a wide variety of nuclear facilities for both the government and industry. Particularly relevant commercial experience includes major responsibilities with regard to the design and construction of a number of reprocessing and fuel fabrication facilities: for the U.S. government at Richland, Washington and Oak Ridge, Tennessee; for the Italian, Swedish and Indian governments; and for a division of the Atlantic Richfield Co.

For the past ten years I have been employed by the Atomic Energy Commission and the NRC. Relevant government experience includes my assignments as assistant director and chief engineer in licensing of commercial nuclear fuel material activities. I was the program manager for NRC's generic analysis of mixed oxide fuel use in light water reactors (GESMO); a member of the U.S. delegation to the International Fuel Cycle Evaluation for the area of fuel reprocessing and recycle; and am involved in the TMI-2 Waste Management Task Force. Further details of my background are contained in my statement of professional qualifications.

Q.4. What is the nature of the responsibilities you have regarding the Clinch River Breeder Reactor ("CRBR")?

A.4. I am the Office of Nuclear Materials Safety and Safeguards ("NMSS") Project Manager responsible for the preparation of the Fuel Cycle portion of the supplement to the Final Environmental Statement ("FES") for the CRBR Plant. I directed and participated in the review of the applicant's updated environmental report related to the various steps in the CRBR fuel cycle including: 1) fuel fabrication, 2) reprocessing, 3) waste management, 4) transportation, and 5) safeguards. In particular, I directed the updating of Appendix D, "Environmental Effects of the CRBR Fuel Cycle and Transportation of Radioactive Materials"; Appendix E, "Safeguards Related to the CRBR Fuel Cycle and Transportation of Radioactive Materials"; section 7.2, "Transportation Accidents Involving Radioactive Material"; section 7.3, "Safeguards Considerations"; as well as section 5.7.2.6, "Transportation of Radioactive Materials";

and section 5.7.2.7, "Fuel Cycle Impacts" of the FES. In addition, I was responsible for the review of section 11.9.5 of the 1977 FES.

Q.5. Mr. Soffer, please state your position, your employer, and the nature of your work?

A.5. I am Section Leader of the Site Analysis Section, Siting Analysis Branch, Division of Engineering, Office of Nuclear Reactor Regulation ("NRR"), of NRC. I am responsible for the review of the population characteristics of nuclear power reactor sites, including the exclusion area, as well as the review of nearby industrial, transportation and military facilities. A statement of my professional qualifications is attached to this testimony.

Q.6. What is the nature of your responsibilities regarding the CRBR?

A.6. I am Mr. Ferrell's immediate supervisor. In this capacity I supervised the review of the exclusion area, population characteristics and nearby industrial, transportation and military facilities of the Clinch River site as well as for each of the alternative sites analyzed for the CRBR. These reviews and contributions are in Sections IIIA, IIIB and IIIC of the Clinch River Breeder Reactor Site Suitability Report ("SSR"), NUREG-0786, Sections 2.1 and 2.2 of the Applicant's Environmental Report ("ER"), and Section 9 and Appendix L of the 1982 Supplement to the Final Environmental Statement ("FES") for the CRBR ("FES Supplement"). I am also responsible for evaluating underground siting of the CRBR as an alternative and my contribution in this regard appears in Section 11 of the FES Supplement.

Q.7. Mr. Spickler, please state your position, your employer, and the nature of your work?

A.7. My name is Irwin Spickler. I am the Leader of the Meteorological Section, Accident Evaluation Branch, Assistant Directorate for Radiation Protection, Division of Systems Integration, NRR, of NRC. I supervise the review of the meteorological aspects of nuclear reactor licensing actions. A statement of my professional qualifications was received into evidence during the hearing session commencing August 23, 1982 (Tr. 2541).

Q.8. What is the nature of your responsibilities regarding the CRBR?

A.8. I was responsible for the meteorological review for CRBR, as presented in Sections 2.6, 5.3, 5.7, 6.1.3, 6.2.3, 9.2, 11.2, and Appendix L of the FES Supplement for CRBR.

Q.9. What is the subject matter of your testimony?

A.9. Our testimony addresses Joint Intervenors' Contentions 5(a) and 7(c). Contention 5(a) states:

> Neither Applicants nor Staff have established that the site selected for the CRBR provides adequate protection for public health and safety, the environment, national security, and national energy supplies; and an alternative site would be preferable for the following reasons:

(a) The site meteorology and population density are less favorable than most sites used for LWRs.

(1) The wind speed and inversion conditions at the Clinch River site are less favorable than most sites used for light-water reactors.

(2) The population density of the CRBR site is less favorable than that of several alternative sites.

(3) Alternative sites with more favorable meteorology and population characteristics have not been adequately identified and analyzed by Applicants and Staff. The analysis of alternative sites in the ER and the Staff Site Suitability Report gave insufficient weight to the meteorological and population disadvantages of the Clinch River site and did not attempt to identify a site or sites with more favorable characteristics.

Contention 7(c) states:

c) Alternative sites with more favorable environmental and safety features were not analyzed adequately and insufficient weight was given to environmental and safety values in site selection.

(1) Alternatives which were inadequately analyzed include Hanfard Reservation, Idaho Reservation (INEL), Nevada Test Site, the TVA Hartsville and Yellow Creek sites, co-location with an LMFBR fuel reprocessing plant (e.g., the Development Reprocessing Plant), an LMFBR fuel fabricating plant, and underground sites.

In particular, our testimony will discuss the applicable NRC criteria for meteorology and demography and will show that the CRBR site meteorology and population density meet these criteria. Our testimony will compare these characteristics with those of other sites used for LWR's, and will present the bases for the Staff conclusion that the site selected for the CRBR provides adequate protection for the public health and safety as well as the environment, and that there are no alternative sites that are environmentally preferable to the Clinch River site with regard to site meteorology and population density. Our testimony will also address the co-location and underground siting concepts.

Q.10. Mr. Spickler, is meteorological data specific to the CRBR site available to the NRC Staff ("Staff")?

A.10. Yes.

Q.11. Describe how this meteorological data was collected.

A.11. Since April 1973 a temporary 200-ft instrumented tower has been in operation southward of the reactor site. In February 1977, two permanent instrumented towers were installed: a 10 meter tower south of the site and a 110 meter tower southeast of the site. Simultaneous measurements were taken on the temporary and permanent towers during the period of February 16, 1977 to March 2, 1978. The 110 meter tower was put back into service during April of 1982 and will operate during construction of CRBR. The 10 meter tower instrumentation consisted of wind speed and wind direction sensors located at the 10 meter level. The 110 meter tower instrumentation consists of wind speed and direction sensors located at the 10, 60, and 110 meter levels; temperature sensors at the 10-, 60-, amd 110-m levels; dew point sensors at the 10 meter level; and solar radiation atmospheric pressure and precipitation sensors at the 1 meter level. Additional information on the Applicants' meteorological monitoring program is provided in Section 6.1.3 of the FES Supplement.

The Staff analyzed the data collected on site on the permanent towers for the period February 17, 1977 through February 16, 1978. For that one year period, the joint data recovery rate of 10 meter wind speed and wind direction, and the temperature difference between the 10 meter and 60 meter levels, was 97 percent.

Q.12. Does the Applicants' onsite meteorological monitoring program, in terms of sensor accuracy, calibration intervals, and recovery rate meet the standards recommended in Regulatory Guide 1.23?

A.12. Yes.

Q.13. Please present the meteorological data for the CRBR site.

A.13. The CRBR site is characterized by a high frequency of stable atmospheric diffusion conditions, westerly winds, and low wind speeds which are typical of the northern Appalachian area of the Southeastern United States.

The joint frequency of wind speed direction and atmospheric stability during the period February 17, 1977 through February 16, 1978 are presented in Chapter 2.3 of the PSAR (Amendment 65, February 1982) and Chapter 2.6 of the ER (Amendment XI, January 1982). Stable atmospheric diffusion conditions (E, F & G) occurred

56 percent of the year. Neutral stability (D) and unstable (A, B & C) conditions occurred 36% and 8% of the year, respectively. The prevailing wind sectors are from the west, the WNW, W, WSW winds occurring 35%, 29%, and 26% of the year, respectively.

The annual 10 meter wind speed had an occurrence of winds less than 1.5 m/sec 60% of the time, winds less than 2.5 m/sec 80% of the time and winds less than 0.4 m/sec 3% of the time.

Q.14. How did the Applicants utilize this data to analyze the consequences of routine and accidental radiation releases?

A.14. The Applicants used the 10 meter wind speed and direction and the 10 to 60 meter temperature gradient data (atmospheric stability), measured on-site between February 17, 1977 through February 16, 1978, to determine the diffusion factor (X/Q) to be utilized in their analyses of the consequences of routine and accidental releases of radioactivety.

In evaluating the atmospheric transport and diffusion characteristics from routine radioactivity releases, the Applicants used a Straight-Line Trajectory Model, as described in Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors." All releases were assumed to be at ground level. The calculations also included an estimate of the maximum increase in calculated relative concentration and deposition due to recirculation of airflow.

Short-term (up to 30 days) X/Q values were calculated by the Applicants in order to analyze the consequences of accidental releases, in accordance with the methodology described in Regulatory Guide 1.145. A direction dependent atmospheric dispersion model with enhanced lateral dispersion during neutural and stable atmospheric conditions accompanied by low wind speeds was used. X/Q values for each of the 16 cardinal point sectors that is not exceeded 0.5% of the total time were calculated by the Applicants. The highest of each of these 16 sector X/Q values was defined as the maximum section X/Q value, and was compared with the overall site X/Q that is exceeded no more than 5% of the total time. Whichever value was higher was used to determine the consequences of accidental releases at the exclusion zone boundary ("EAB") of 670 meters and outer boundary of the low population zone ("LPZ") of 4023 meters. For the Clinch River site the more conservative X/Q values were those based upon the 0.5% sector values and was thus utilized by the Applicants to evaluate the consequences of design basis accidental releases.

Q.15. What are the Applicants' calculated X/Q values at the EAB and the LPZ for analyzing the consequences of accidental releases?

A.15.

Time Period	Accident X/Q Values Distance (meters)	X/Q (sec/m^3)
0-2 hours	EAB, 670 meters	1.1×10^{-3}
0-8 hours	LPZ, 4023 meters	1.2×10^{-4}
8-24 hours	LPZ, 4023 meters	8.4×10^{-5}
1-4 days	LPZ, 4023 meters	3.7×10^{-5}
4-30 days	LPZ, 4023 meters	1.2×10^{-5}

Q.16. What are the Applicants' calculated X/Q values for estimating the consequences of routine radioactivity releases?

A.16. The most limiting off-site annual average X/Q value calculated by the Applicants was 1.02×10^{-4} sec/m^3 which was associated with winds from the southeast.

Q.17. Did the Staff verify the Applicants' calculated X/Q values?

A.17. Yes. The Staff utilized the same data base as utilized by the Applicants, and performed independent X/Q analyses in accordance with Regulatory Guides 1.111 and 1.145.

Q.18. What are the Staff's calculated X/Q values for CRBR at the EAB and LPZ?

A.18. Accident X/Q Values

Zone	Distances (Meters)	Time Period	X/Q (sec/m^3)
EAB	670	0-2 hours	1.22×10^{-3}
LPZ	4023	0-8 hours	1.2×10^{-4}
LPZ	4023	8-24 hours	8.4×10^{-5}
LPZ	4023	1-4 days	3.9×10^{-5}
LPZ	4023	4-30 days	1.4×10^{-5}

Q.19. What was the Staff's calculated value for the most limiting off-site annual average X/Q value?

A.19. The Staff's calculated X/Q value in the most limiting off-site annual average case was 1.2×10^{-4} sec/m^3.

Q.20. Does the Staff use the same methodology for calculating X/Q values for light water reactors ("LWRs")?

A.20. Yes.

Q.21. How do the X/Q values for the CRBR site compare with X/Q values for licensed LWR sites?

A.21. The diffusion conditions at the CRBRP are better than some of the LWR diffusion conditions that have already been permitted or licensed and are comparable to LWR sites in the general region.

Q.22. How did the Staff evaluate the diffusion characteristics of the potential alternate sites to CRBR?

A.22. The Staff made 2 comparisons to characterize diffusion conditions of each potential alternate site. First, the Staff reviewed the joint occurrences of stable atmospheric diffusion conditions and the average wind speeds for these conditions, because this combination of conditions largely determine the relative diffusivity of an area under the poorest diffusion conditions. A comparison of the frequency of stable atmospheric conditions and the average stable wind speed for each of the alternate sites is presented in the following table:

Site	Frequency Stable Atmospheric Conditions (%)	Average Stable Wind Speed (MPH)	Source of Date	Period
CRBR	56	4.4	CRBR PSAR	2/77-2/78
Hartsville	51	4.2	Hartsville PSAR	2/73-1/74
Murphy Hill	54	4.0	Bellefonte PSAR	11/72-10/73
Phipps Bend	54	3.2	Phipps Bend PSAR	2/74-1/75
Yellow Creek	52	3.6	Yellow Creek PSAR	7/74-6/75
Savannah River	44	5.4	Vogtle PSAR	12/72-12/73
Hanford	58	4.5	WPPS-2 PSAR	4/74-3/76
Idaho	57	5.4	PBF SER	1967 & 1968

Second, the Staff compared atmospheric dispersion conditions used for accident consequence assessments relative concentration (X/Q) values were obtained from the Staff SER for each alternate site or from an appropriate nearby site. The following table presents a comparison of X/Q calculations at EAB and LPZ of the alternate sites:

Alternate Sites	EAB Distance (meters)	0.2 Hr.* (sec/m^3)	LPZ Distance (meter)	0-8 Hr. (sec/m^3)	8-24 Hr. (sec/m^3)	1-4 Day (sec/m^3)	4-30 Day (sec/m^3)
CRBR	670	1.22 E-3	4023	1.2 E-4	8.4 E-5	3.9 E-5	1.4 E-5
Bellefonte	914	1.8 E-3	3219	1.8 E-4	1.2 E-4	4.8 E-5	1.3 E-5
Hartsville	1220	4.9 E-4	4828	5.9 E-5	4.1 E-5	1.9 E-5	6.2 E-6
Phipps Bend	760	1.8 E-3	4827	1.2 E-4	8.0 E-5	3.5 E-5	1.1 E-5
Yellow Creek	695	1.5 E-3	4828	6.4 E-5	3.5 E-5	1.2 E-5	2.4 E-6
Vogtle	1098	1.8 E-4	3220	3.3 E-5	2.2 E-5	9.2 E-6	2.8 E-6
WPPS-2	1950	1.7 E-4	4829	3.8 E-5	2.8 E-5	1.4 E-5	5.3 E-6

*Table Values are expressed as follows: 2.3 E-3 = 2.3×10^{-3}

Data from the Bellefonte nuclear power plant site, which is across the lake from Murphy Hill, was utilized in the two previous tables to represent the Murphy Hill site. Data from the Vogtle nuclear power plant site, which is in the same general area as the Savannah River site, has been utilized to represent the Savannah River site. Data from WPPS-2 has been utilized to represent both Hanford and Idaho because the occurrence of stable diffusion diffusion conditions and the average wind speeds were the approximately the same, and because both sites are in areas which are characterized by desert diffusion parameters.

From the above tables it can be seen that the five TVA area sites (CRBR, Murphy Hill, Hartsville, Phipps Bend and Yellow Creek) all have comparable accident X/Q values. All have comparable stable atmospheric diffusion occurrence frequencies and average stable wind speeds. The Savannah River site has significantly less frequent stable conditions with higher wind speeds and shows significantly better diffusion conditions. Hanford and Idaho have high stable atmospheric diffusion frequency with a higher average wind speed. Based upon extensive diffusion studies at both Idaho and Hanford, it has been found that desert diffusion is better than non-desert locations and a different set of diffusion parameters (sigma y and sigma z) have been developed for desert areas. Thus, the accident diffusion conditions at both Hanford and Idaho are better than the TVA area sites.

Q.23. Mr. Soffer, what criteria are utilized by the Staff for evaluating the siting of nuclear power reactors?

A.23. The Staff utilized the Commission's criteria for determining the suitability of proposed sites for nuclear power plants contained in 10 C.F.R. Part 100. Proposed sites are required to meet certain tests related to the surrounding population.

A site is required to have an exclusion area surrounding the reactor where resident individuals are excluded. The Applicants must also define a low population zone ("LPZ") immediately beyond the exclusion area. In addition, the distance from the reactor to the nearest population center must be at least one and one-third times the low population zone outer radius, and the radiological consequences of an assumed hypothetical fission product release must meet certain dose guidelines to an individual located at the boundaries of the exclusion area and the low population zone.

Q.24. What is the exclusion area for CRBR, as defined by the Applicants?

A.24. The Applicants have specified the exclusion area as a 1364 acre tract of land in Roane County, Tennessee, as described in section 2.1 of the ER and the PSAR, and described in section II.A of the Staff Site Suitability Report (SSR), NUREG-0786.

Q.25. Does the Staff agree with the Applicants' definition of the CRBR exclusion area?

A.25. Yes.

Q.26. What is the low population zone ("LPZ"), as defined by the Applicants?

A.26. The Applicants have specified the LPZ as a circular area with a radius of 2.5 miles centered on the proposed reactor.

Q.27. Does the Staff agree with the Applicants' definition of the LPZ?

A.27. Yes.

Q.28. What is the population center for the CRBR and the population center distance for CRBR as calculated by the Applicants?

A.28. The nearest population center has been designated to be Oak Ridge, Tennessee. The population center distance designated by the Applicants is 7 miles in the north-northeast direction.

Q.29. Does the Staff agree with the Applicants' identification of the population center for CRBR, and the Applicants' calculated population center distance?

A.29. Yes.

Q.30. Does the exclusion area, LPZ, and population center distance comply with NRC regulations?

A.30. Yes. The exclusion area and LPZ meet the definitions given in 10 C.F.R. Part 100. In addition, the population center distance of 7 miles is at least one and one-third times the LPZ outer radius of 2.5 miles. Even if future population growth results in a population center distance of 5 miles, this value will also meet the requirement of 10 C.F.R. Part 100.

Q.31. Has the Staff compared the exclusion area, LPZ, and population center distance for the CRBR site with other LWR sites?

A.31. Yes.

Q.32. How does the size of the CRBR exclusion area compare with those of other LWR sites?

A.32. The minimum distance from the CRBR reactor to the exclusion area boundary is about 2200 feet, or 0.41 mile. The exclusion area distance distribution for other LWR sites is shown in the accompanying table.

Exclusion Area Size (miles)	Percentage of LWR Sites
less than 0.4	40%
0.4-0.6	31%
greater than 0.6	29%

Based on this data, we conclude that the exclusion area size for the CRBR site is about average when compared to other LWR sites.

Q.33. How does the size of the CRBR LPZ compare with that of other LWR sites?

A.33. The LPZ for the CRBR site is 2.5 miles. The LPZ size distribution for other LWR sites is shown below:

LPZ Size (miles)	Percentage of LWR Sites
less than 2	20%
2 - 3	40%
greater than 3	40%

Based on this data, we conclude that the LPZ for the CRBR site is about average when compared to other LWR sites.

Q.34. How does the distance to the nearest population center for the CRBR site compare with that of other LWR sites?

A.34. The distance to the nearest population center for the CRBR site is 7 miles. The population center distance distribution for other LWR sites is shown below:

Pop. Center Dist (miles)	Percentage of LWR Sites
less than 5	12%
5 - 10	27%
greater than 10	61%

Based on this data, we conclude that the population center distance for the CRBR is slightly less than average when compared to other LWR sites.

Q.35. What is the population distribution around the CRBR site?

A.35. The resident population out to 30 miles for the year 1980, and projections for 1990 and 2030, are shown in Table III of the SSR.

Q.36. Has the Staff made any efforts to verify the accuracy or reasonableness of this data.

A.36. Yes. As described in Section II.B of the SSR the Staff obtained an independent estimate of the 1980 population within 50 miles and compared this with the Applicants' value. In addition the Staff examined population growth rates presented by the Applicants with those from independent sources. The Staff also examined population data for 1970 at distances of 5, 10, 20 and 30 miles and using known growth rates from 1970 to 1980, examined the Applicants' 1980 population data. On the bases of these verifications the Staff concludes that the Applicants' population data and projections are reasonable.

Q.37. Mr. Soffer, are there any Commission regulations regarding population density which the Staff utilizes for evaluating nuclear power reactor siting?

A.37. No. 10 C.F.R. Part 100 contains no requirements regarding population density.

Q.38. In the absence of specific Commission requirements on population density, has the Staff established any population density criteria to act as guidance to applicants?

A.38. Yes. Criteria on population density have been published in Regulatory Guide 4.7, Revision 1, "General Site Suitability Criteria for Nuclear Power Stations" (November 1975). As set forth in Section C.3. of Regulatory Guide 4.7, if the population density, including weighted transient population, projected at the time of initial operation of a nuclear power station, exceeds 500 persons per square mile averaged over any radial distance out to 30 miles (cumulative population at a distance divided by the area at that distance), or if the projected population density over the lifetime of the facility exceeds 1000 persons per square mile averaged over any radial distance out to 30 miles, applicants must give special attention and consideration to alternative sites with lower population densities. The population density levels set forth in the Regulatory Guide do not represent upper bound limits of acceptability, but are merely "trip" levels. If the population density "trip" levels are exceeded at the site, the site must be determined to have significant offsetting advantages as compared with available alternate sites of lower density.

Q.39. Mr. Ferrell, has the Staff calculated population density for CRBR?

A.39. Yes. The 0-30 mile population density for the year 1990, as reported in Appendix L of the FES Supplement, is 197 persons per square mile.

Q.40. How do the population density values for the Clinch River site compare with the "trip" levels of Reg. Guide 4.7?

A.40. As noted on page III-2 of the SSR, the Staff concludes that the population density (including weighted transients) for the Clinch River site at projected time of plant startup (year 1990) is well below 500 persons per square mile out to 30 miles. Similarly, the population density at end-of-plant life (year 2030) is well below 1000 persons per square mile out of 30 miles.

Q.41. Does the CRBR population density meet the density criterion ("trip" levels) of Regulatory Guide 4.7?

A.41. Yes.

Q.42. Mr. Soffer, has the Staff compared the population density around the CRBR site with those of other LWR sites?

A.42. Yes. The Staff performed an analysis which lists a first-order prioritization of all power reactor sites with regard to power level and population density. This analysis (attached to this testimony), was presented as part of a Staff paper (SECY 81-25) to the Commission, and divides all LWR reactor sites into 5 groups on the basis of reactor power level and weighted population density. We have also examined the CRBR site in regards to reactor power level and weighted population density using the same methodology as given in the above-referenced SECY paper, and find that the CRBR site falls into the category labeled Group II - Average.

Q.43. How does the 0-30 mile population density for CRBR compare with those of other LWR sites?

A.43. The CRBR site, on the basis of reactor power level and weighted population density, is average when compared to other LWR sites.

Q.44. Mr. Ferrell, has the Staff calculated population densities for each of the alternate sites which were evaluated in Appendix L of the FES Supplement?

A.44. Yes. The 0-30 mile population densities for the year 1990, as reported in Appendix L of the FES Supplement, is presented below:

Reactor	Population Density (people/mile^2)
Hanford	66
Hartsville	66
Idaho	36
Murphy Hill	103
Phipps Bend	166
Savannah River	93
Yellow Creek	48

The population densities are lower at each of the alternative sites, compared to the Clinch River site.

Q.45. Mr. Soffer, are any of the alternate sites environmentally preferable to the Clinch River site, on the basis of population density?

A.45. No, since the Staff does not attach any significance to the differences in population density between Clinch River, and each of the alternative sites.

Q.46. Why does the Staff find no significance to the numerical differences in population density between the Clinch River site and each of the alternative sites?

A.46. The Staff uses population density as a relatively crude surrogate for the residual risk associated with accidental releases of radioactivity. The Staff performed an assessment of the residual risk of severe accidents at the Clinch River site in Appendix J of the CRBR FES Supplement. In Appendix J the Staff concluded that the risks to the public were very low for the Clinch River site. Accordingly, any reduction in the already very low residual risk associated with accidental radiation releases which are attributable to population density reductions are not significant.

In addition, as stated in Answer 40, the 0 to 30 population density of the Clinch River site is well below the trip level set forth in Regulatory Guide 4.7. Regulatory Guide 4.7 states that areas with low population densities are to be preferred for the siting of nuclear power reactors. However, the Regulatory Guide does not make any distinction with regard to sites with differing population densities which are below the "trip" levels, and defines "low population densities" to be those which are below the trip levels. Consequently, the Staff concludes that any differences in population density between Clinch River and the alternative sites is insignificant, and that no alternative site is preferable to Clinch River with regard to population density.

Q.47. Mr. Soffer, describe the underground siting concept for nuclear power reactors.

A.47. Underground siting of a nuclear power plant would involve locating the nuclear reactor and possibly other plant equipment beneath the surface of the earth either in a mined rock cavity or by covering the plant with fill earth after construction in an excavated cut.

Q.48. Has the AEC and the NRC evaluated the underground siting concept for nuclear power reactors?

A.48. Yes. Underground siting has been studied in the U.S. for almost 20 years. In July 1973, the AEC issued a report entitled "The Safety of Nuclear Power Reactors and Related Facilities," WASH-1250, which discussed, among other things, underground siting. The report cited the attractiveness of the possibility of "absolute" containment of fission products in the event of an accident, but found that "the AEC has found little technical basis for encouraging the general use of underground siting." The report concluded that:

> "the weight of evidence currently suggests that undergound siting: a) has necessary features (e.g., penetrations) which tend to offset the presumed containment advantages, b) would add significantly to the costs of nuclear power plants, c) requires extensive and costly R&D for unresolved engineering problems, and d) does not offer a general solution to siting problem in the U.S."

The report also stated a general AEC position that: "although the AEC does not reject the concept of underground siting, it finds little basis for favoring it over surface siting."

In 1975 a study was initiated by the NRC to obtain authoritative answers to generic questions associated with the underground siting concept. This research was carried out by Sandia Laboratories and resulted in the publication in August 1977 of a report entitled "Underground Siting of Nuclear Power Plants: Potential Benefits and Penalties" NUREG-0255. The report concluded that while underground plants had certain inherent safety advantages over surface plants, there were also inherent disadvantanges with regard to safety and that overall "the expected benefits of underground siting in terms of improved safety do not appear to offset the penalties."

Studies have also been carried out independently of the AEC and NRC. Probably the most extensive of these is one carried out for the State of California Energy Commission, entitled "Underground Siting of Nuclear Power Reactors: An option for California," which was published in June 1978. The study found that underground siting offered a potential for reducing consequences from core-melt accidents to very low levels, but that other alternatives such as remote siting and controlled release of excessive pressure through simple, engineered filter systems captured some of the benefits of underground siting at less cost. The study recommended that:

> "underground siting not be mandated due to a) the uncertainty remaining over costs, construction time and possible licensing concerns; b) the existence of what appear to be moderately effective and less expensive technical alternatives; and c) the opportunity to implement remote siting within California."

Q.49. Would the underground siting concept be applicable to a Liquid Metal Fast Breeder Reactor ("LMFBR") such as CRBR?

A.49. Yes. Underground siting of an LMFBR breeder reactor was suggested in studies as early as 1972 (see, for example, Smernoff, B.J., "Underground Siting of the LMFBR Demonstration Plant: A Serious Alternative," HI-1618/2-P, Hudson Institute, September 12, 1972). The Applicant considered underground siting for the CRBR in Section 2.3.2 of the "Supplemental Alternative Siting Analysis for the LMFBR Demonstration Plant." There appears to be no technical reasons why underground siting would be precluded for an LMFBR such as the CRBR.

Q.50. What are the advantages and disadvantages of underground siting of CRBR?

A.50. The Staff evaluation of underground siting of the CRBR has been discussed in Section 11.9.6 of the FES and updated in the same section of the FES Supplement. Based on the studies of WASH-1250 and NUREG-0255, underground plants have safety advantages over surface plants with regard to:

1) protection against aircraft crashes or warfare munitions which could conceivably initiate a reactor accident;

2) improved retention of radioactive releases to the atmosphere following a core meltdown, provided that the numerous penetrations to the surface from an underground plant were promptly isolated and maintained in an isolated condition;

3) a modest reduction in seismic vulnerability for underground plants.

Underground plants have the following safety disadvantages as compared to surface plants:

1) greater operational problems associated with inservice inspection and maintenance which in turn, could lead to decreased equipment reliability and an increased probability of an accident;

2) greater potential for flooding;

3) greater potential for groundwater contamination following an accident.

Q.51. Is underground siting of the CRBR technologically feasible?

A.51. The above studies have concluded that underground siting of nuclear power plants appears to be technically feasible, although no engineering design presently exists. Certain engineering and occupational problems have been identified. For example, the success of the underground siting concept depends on the prompt isolation of the penetrations to the surface. Maintenance of seals which isolate the penetrations has been identified as a critical design problem for underground plans. Moreover, prompt isolation of such penetrations could reduce the movement of any operating or maintenance personnel located below ground at the time of the accident, which may present an occupational hazards problem.

The few research reactors that have been located underground are in mined rock caverns having diameters up to about 20 meters. The CRBR would require a cavity of about 75 meters in diameter and hence would require cavities or excavations significantly larger than presently existing ones. Although an excavation of this size is considered feasible, the effort is unprecedented and could lead to unforeseen difficulties.

Based on the NUREG-0255 study, an underground plant is estimated to cost about 20 to 40 percent more than a surface plant.

Q.52. What is the Staff's conclusion regarding underground siting as a siting alternative for the CRBR?

A.52. As presented in Section 11.9.6 of the 1982 FES Supplement, the Staff concludes that underground siting has been sufficiently evaluated and while feasible, the expected benefits in terms of improved safety do not appear to offset the penalties of construction difficulties, operational problems leading to degraded safety, and additional costs.

Q.53. Mr. Lowenberg, what does the term, "co-location", refer to with regard to nuclear facilities?

A.53. Co-location of nuclear facilities has been considered or postulated for several general applications:

1. Centralized location of large scale fuel cycle facilities such as commercial fuel reprocessing plants and fuel fabrication plants.

2. Centralized location of a number of nuclear power reactors for potential improvements in economy, licensing, socio-economic and emergency response aspects.

3. Centralized location of large scale fuel cycle facilities with nuclear power reactors.

The primary potential benefits from co-location of nuclear facilities are generally ascribed to the co-location of large scale fuel cycle facilities (application 1.), which may have safeguard merits.

Co-location of such facilities would minimize the handling and transportation of large amounts of strategic nuclear materials and possibly improve waste management activities.

Q.54. How could co-location be applied to the CRBR and its related fuel cycle?

A.54. Since the CRBR project involves only one reactor and the related fuel cycle facilities, only application 3, centralized location of large scale fuel cycle facilities with a nuclear power reactor, is relevant for consideration.

Q.55. What would be the advantages and disadvantages of co-location of nuclear power reactors with related fuel cycle facilities?

A.55. The co-location of power reactors with large scale fuel cycle facilities has been considered and found to have essentially as many disadvantages as advantages. The most significant potential advantage of co-location of nuclear facilities comes from the possibility of decreasing the transportation of separated strategic nuclear materials. This may be accomplished in a realistic manner by co-location of large scale fuel reprocessing and fuel fabrication plants. Co-location of a nuclear power reactor with fuel cycle facilities would only decrease the shipment distances of a small amount of fresh and spent fuels. This has never been considered as a very significant factor that should be considered in the cost/benefit evaluation process for a single reactor.

The primary disadvantage of co-location of nuclear power reactors with fuel cycle facilities is the need to constrain the size of the fuel cycle facilities to match the fuel capacity of the reactors. These advantages may be realized only when the fuel requirements of the reactors approximately matches the fuel cycle facility capabilities.

Q.56. In view of the above considerations do you believe that there is potential merit to co-location of the CRBR with other LMFBR fuel cycle facilities?

A.56. As discussed in Section 11.9.5. of the 1977 FES and Answer 53 above, co-location of nuclear power reactors with large scale fuel cycle facilities is feasible only where the fuel cycle facility capabilities approximately match the fuel requirements of the reactors. The capabilities of the fuel cycle facilities that are proposed for the CRBR are significantly larger than the CRBR fuel needs. There is little apparent merit to co-location of the CRBR with the proposed pilot or developmental LMFBR fuel cycle facilities. Accordingly, the co-location of the CRBR with any of its related fuel cycle facilities would not have a significant effect on site selection considerations.

Prioritization of Sites with Regard to Population Density

1. Introduction

In comparing and evaluating the population around nuclear power reactor sites, the staff has long recognized that the population characteristics of a site, that is, its density and distribution, are a relatively crude measure of the consequences associated with the accidental release of radioactivity. The residual risk from an accident would depend not only upon the population density of the site, but also upon many other factors, such as reactor design, onsite and offsite management and technical support resources, external hazards, liquid pathway considerations, meteorological conditions at the time of the accident, and effectiveness and nature of public protective actions taken. In addition, the risk is not uniform for all members of the population regardless of distance from the site, but would be higher for those persons relatively close to the site, and would generally decrease with distance away from the site.

An analysis has been carried out to obtain a first-order prioritization of sites based upon population density and distribution. The discussion that follows outlines the rationale and methodology used and gives the results of this analysis.

2. Methodology

In carrying out this analysis, the following assumptions and methodology were used:

(a) All sites where a reactor was either in operation, under construction, or where a construction permit was presently under active review were evaluated. This involved a total of 93 sites.

(b) The population data used were taken from NUREG-0348, based on the 1970 census. The population data for the Fermi site as reported in NUREG-0348 are in error and were corrected for this analysis by a special computer run of the 1970 census tape.

(c) Although it is well-known that individuals closer to the reactor are at a higher level of risk, given an accident, than those more remotely located, the precise quantification of the variation of risk with distance is still somewhat uncertain. For the purpose of this analysis, the distance weighting given by the Site Population Factors (SPF), as given in WASH-1235, were used. Further, population beyond 30 miles was neglected, because the consequences at distances within 30 miles were considered to dominate any considerations of overall societal impact, and beyond 30 miles the potential population exposure differences from site to site become less sharp. Preliminary analyses carried out by the staff have indicated that somewhat differing weighting schemes, or the factoring in of population out to 50 miles, does not change the resulting prioritization of sites to a significant degree.

(d) The power level of the largest reactor at the site was multiplied by the SPF value to account, in a first-order way, for the variation of reactor fission product inventory from site to site. Only one reactor at a site was considered, even where multiple reactors exist or are contemplated, because the probability of an accident involving more than one reactor simultaneously was considered negligible. Although it can be argued that the population around a 4 reactor site is at a higher level of risk than those around a single reactor site, the prioritization of sites is intended to give a measure of the relative consequences, given that an accident has occurred. The number of reactors at a site presumably effects only the probability of an accident. Also, it could be argued that a multi-reactor site would have some attributes that would reduce risk, compared to a single-reactor site, because of greater management and technical resources that can be applied to reducing either the likelihood or consequences of an accident. Using the above methodology, the reactor power level times the SPF value was calculated and tabulated for each of the 93 sites considered. The results are discussed below.

3. Results

The reactor power level times SPF (P x SPF) was calculated for each of the 93 sites. The resulting values ranged from a high value of 2980 to a low value of 6. The median value is 206; and the median site has a population of less than 100 persons per square mile, which is almost a factor of two less than the population of the average site. The sites are not listed in numerical order, since this would imply a greater degree of precision than is warranted by the uncertainties in the analysis. Also, as pointed out previously, the residual risk at a particular site cannot be measured in terms of consequences alone, since plant design and other factors are important contributors to risk. Therefore, we decided to place each site into one of five groups or categories. The variation within a given group was selected to be sufficiently small so that each site within that group is considered to have about the same ranking. In selecting the groups we decided to use the median value and factor of two variation about the median to demarcate the "average" group boundaries. The other groups were chosen as indicated below.

Group No.	Title	Range
I	Below Average	PXSPF less than one-half the median value (PXSPF < 100)
II	Average	PXSPF between one-half and twice the median value (PXSPF from 100 to 400)
III	Slightly Above Average	PXSPF between twice and four times the median value (PXSPF from 400 to 800)
IV	Above Average	PXSPF between four and eight times the median (PXSPF from 800 to 1600)
V	Substantially Above Average	PXSPF greater than eight times the median (PXSPF > 1600)

Within each group the sites have been listed in alphabetical order, as shown in the following tables.

Group V - Substantially Above Average

1. Indian Point
2. Limerick
3. Zion

Group IV - Above Average[1]

1. Bailly
2. Beaver Valley
3. Fermi
4. Millstone
5. Seabrook
6. Shoreham
7. Three Mile Island
8. Waterford

Group III - Slightly Above Average

1. Byron
2. Catawba
3. Cook
4. Cherokee
5. Erie
6. Forked River
7. Haddam Neck
8. Hope Creek
9. McGuire
10. Midland
11. Peach Bottom
12. Perkins
13. Pilgrim
14. Perry
15. Salem
16. Sequoyah
17. Susquehanna
18. Rancho Seco
19. Turkey Point
20. Zimmer

Group II - Average

1. Arkansas
2. Bellefonte
3. Black Fox
4. Braidwood
5. Browns Ferry
6. Calvert Cliffs
7. Clinton
8. Brunswick
9. Davis-Besse
10. Duane Arnold
11. Fort Calhoun
12. Fitzpatrick
13. Ginna
14. Hartsville
15. LaSalle
16. Maine Yankee
17. Marble Hill
18. Nine Mile Point
19. Oconee
20 Oyster Creek
21. Palisades
22. Phipps Bend
23. Prairie Island
24. Quad Cities
25. River Bend
26. Robinson
27. San Onofre
28. Shearon Harris
29. Summer
30. Surry
31. St. Lucie
32. Skagit
33. Trojan
34. Vogtle
35. Watts Bar
36. WPPSS 3/5
37. Vermont Yankee
38. Monticello
39 Yellow Creek

---

[1]Bailly and Millstone Unit 3 are the only plants in Group IV that are in the early stages of construction.

Group I - Below Average

1. Allens Creek
2. Big Rock Point
3. Callaway
4. Comanche Peak
5. Cooper
6. Crystal River
7. Diablo Canyon
8. Dresden
9. Farley
10. Ft. St. Vrain
11. Grand Gulf
12. Hatch
13. Kewaunee
14. LaCrosse
15. North Anna
16. Palo Verde
17. Pebble Springs
18. Point Beach
19. South Texas
20. WPPSS 2
21. WPPSS 1/4
22. Wolf Creek
23. Yankee Rowe

CHARLES M. FERRELL

PROFESSIONAL QUALIFICATIONS

SITING ANALYSIS BRANCH

DIVISION OF ENGINEERING

I am a site analyst in the Siting Analysis Branch, Division of Engineering, U.S. Nuclear Regulatory Commission. My present duties in this position include the evaluation of site related environmental safety aspects of nuclear power generating facilities and design basis accident analysis. I graduated from Salem College in West Virginia in 1950 with a B.S. degree in physics and a teaching field in chemistry, biology, and mathematics. Upon graduation, I was drafted, and after completion of armored infantry training at Fort Knox, Kentucky, was assigned as a military physicist to the Radiological Division of the U.S. Army Chemical Corps at Edgewood, Maryland. I spent approximately two years in research involving nuclear weapon thermal radiation, nuclear radiation shielding studies and fallout analysis. I was released from active duty and worked for two years as a civilian physicist in Aerosol Physics (Aerobiology) Research at the U.S. Army Chemical Corps Biological Warfare Laboratory at Fort Detrick, Frederick, Maryland. In 1954, I applied for and was granted an AEC Fellowship in Radiological Physics at Vanderbilt University and the Oak Ridge National Laboratory in Tennessee. An additional year of graduate work in physics was taken at West Virginia University. Night school classes in Nuclear Engineering from the University of Maryland plus short summer courses from MIT in Air Pollution, Heat Transfer, and Nuclear Power Reactor Safety constitute the remainder of my formal education. In April, 1974, I completed a two week course in Pressurized Water Reactor Systems at the Westinghouse Training Center in Monroeville, Pennsylvania. I am a charter member of the Health Physics Society.

I have been a member of the AEC's (now NRC's) Regulatory Staff since 1956. Of these twenty-six years, five years were spent in duties involving the safe industrial and medical use of radioisotopes, in the evaluation of spent reactor fuel shipping casks and the promulgation of reactor fuel shipping regulations. Eight years were served as the Technical Assistant to the Office of Hearing Examiners, U.S. Atomic Energy Commission in which I assisted in approximately 40 hearings on nuclear power reactors, fuel reprocessing plants, and in addition contract appeals hearings on nuclear submarine components and nuclear equipment.

In January, 1969, I transferred to my present position. Since that time I have served as the site analyst on over 50 nuclear power plants, two U.S. Navy nuclear submarine reactors and a proposed nuclear powered crude oil tanker. I served as one of the technical reviewers of Chapter 7, "Assessment of Reactor Safeguards" in Applied Radiation Protection and Control by J.J. Fitzgerald, published under the auspices of the Division of Technical Information United States Atomic Energy Commission. I am one of the co-authors of the report "Demographic Statistics Pertaining to Nuclear Power Reactor Sites" NUREG-0348, and the report "Control of Heavy Loads at Nuclear Power Plants" NUREG-0612, published by the U.S. Nuclear Regulatory Commission.

I have testified in licensing hearings on seven nuclear facilities. These include San Onofre 2/3, Beaver Valley Unit 1, Hutchinson Island (now St. Lucie 1), Yellow Creek 1 and 2, Duane Arnold 1, Trojan Unit 1, and Allens Creek Unit 1.

Educational and Professional Qualifications

Homer Lowenberg
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission

My name is Homer Lowenberg. I am the Chief Engineer of the Office of Nuclear Material Safety and Safeguards. I am responsible for refinements of the technological base for improving and updating the licensing process and for the performance of generic and special studies in support of national and international policies and developments in the non-reactor areas of NRC's responsibilities. I am currently responsible for NRC's environmental review of the CRBR fuel cycle. In addition, I handle activities related to the fuel cycle aspects of the GESMO proceeding and LMFBR research; also, I participate in waste management aspects of the TMI-2 clean-up and in technical review of high and low level waste management programs.

I received the degree of Mechanical Engineer from Stevens Institute of Technology with distinction in Chemical Engineering and attended the Executive Development Program of Cornell University Graduate School of Business and Public Administration.

My professional career was initiated with 5 years of plant development and start-up activities for the Hercules Powder Company in smokeless powder, rocket propellants and high explosive operations.

Then I spent 20 years in the architect-engineering field with the Kellex Corporation which subsequently became Vitro Engineering Co. I was project manager for numerous nuclear facilities including AEC's Purex, Redox and Waste Metal Recovery reprocessing plants at Richland, Washington; the Italian and Swedish Reprocessing facilities; Consolidated Edison's Indian Point Nuclear Power Plant; the Indian Plutonium Laboratory; and a wide variety of nuclear and nonnuclear projects. When Vitro Engineering was sold to Ralph Parsons Co., I was manager of its New York operations.

I was Manager of Central Engineering for Atlantic Richfield Co.'s commercial nuclear activities for 5 years including planning, design and construction of all facilities for fuel material production, fuel assembly and manufacturing, fuel reprocessing and related functions.

I joined the Atomic Energy Commission in 1971 as an assistant director in the regulatory fuels and materials licensing area and continued with NRC upon its creation in 1974. As an assistant director I was responsible for initiating the Reactor-Fuel Cycle Rule (now 10 CFR 51, Tables S-3 and S-4).

I was the program manager and chief commission witness for the GESMO proceeding on widescale mixed oxide use in LWRS; a member of the U.S. delegation to the International Fuel Cycle Evaluation Working Group 4 on Pu reprocessing and recycle and on the TMI-2 Waste Management Task Force.

I am a professional engineer in the states of New York and Pennsylvania.

I was one of the editors of the Reactor Handbook, Volume II published by the AEC on Fuel Reprocessing and have been the program leader on numerous AEC and NRC projects that have been the subject of agency reports.

LEONARD SOFFER

PROFESSIONAL QUALIFICATIONS

SITING ANALYSIS BRANCH

DIVISION OF ENGINEERING

OFFICE OF NUCLEAR REACTOR REGULATION

I am Section Leader of the Site Analysis Section, Siting Analysis Branch, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission. My duties in this position include responsibility for the review and evaluation of the population characteristics of nuclear power reactor sites as well as the evaluation of potential hazards posed by nearby man-related activities.

I received a B. S. Degree (with honors) in Physics from the City College of New York in 1952 and attended graduate school at Case Western Reserve University in Cleveland, Ohio.

Before joining the Commission, I was employed for 21 years as a Physicist and Nuclear Engineer with the National Aeronautics and Space Administration (NASA) at the Lewis Research Center in Cleveland, Ohio. In this capacity, I performed analyses on radiation shielding and nuclear safety requirements for nuclear power systems intended for lunar and space applications. I assisted in the radiation shielding design of the NASA Plum Brook reactor, served on an agency-wide study team investigating the radiological safety aspects of using radioisotopes for space power generation, and was section leader of a group responsible for research on radiation shielding and radiological safety concerns. I also monitored contracts and occasionally lectured on radiological physics and shielding to others within NASA.

I joined the Commission staff in July 1973, and have participated in the detailed review of over 20 nuclear power plants. My responsibilities in this regard have included evaluation of the demographic characteristics and nearby facilities of sites as well as the independent assessment of the likelihood and consequences of various postulated accidents. I have prepared and presented testimony at hearings on the population density and use characteristics of sites as well as the radiological consequences of accidents. In my capacity as Section Leader, Siting Analysis Branch, I am responsible for reviewing the results of similar efforts by others.

Pertinent experience has also included participation in development of a draft standard entitled "Guidelines for Estimating Present and Forecasting Future Population Distributions Surrounding Power Reactor Sites", membership in the NRC Working Group that wrote the "Report of the Siting Policy Task Force" (NUREG-0625), and membership in a Siting Mission to Greece, to assist that Government in the development of demographic criteria for nuclear power plants.

I have also lectured on accident consequence assessment at several courses sponsored by the IAEA, have attended conferences devoted to population projection methodology for small geographic areas and have had discussions with expert demographers on this subject.

I have written about 12 technical papers on various topics related to radiological safety aspects of nuclear reactors. I am a member of the American Nuclear Society and the Population Association of America, which is the professional society of U. S. demographers.