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BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	Docket Nos. 50-413
DUKE POWER COMPANY, ET AL.	)	50-414
	)	
(Catawba Nuclear Station, Units	)	16 April 1984
1 and 2)	)	

PALMETTO ALLIANCE AND CAROLINA ENVIRONMENTAL
STUDY GROUP TESTIMONY OF STEVEN C. SHOLLY ON
EMERGENCY PLANNING CONTENTION NUMBER ELEVEN

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PALMETTO ALLIANCE AND CAROLINA ENVIRONMENTAL STUDY GROUP TESTIMONY OF STEVEN C. SHOLLY ON EMERGENCY PLANNING CONTENTION NUMBER ELEVEN

Q.01 Would you please state your name, position, and business address?

A.01 My name is Steven C. Sholly. I am a Technical Research Associate with the Union of Concerned Scientists (UCS) in Washington, D.C. My primary responsibility with UCS is in technical and policy analysis concerning risk assessment and emergency planning. My business address is: Union of Concerned Scientists, Dupont Circle Building, 1346 Connecticut Avenue, N.W., Suite 1101, Washington, D.C. 20036.

Q.02 Have you prepared a statement of professional qualifications?

A.02 Yes. My statement of professional qualifications is attached to this testimony.

Q.03 What is the purpose of your testimony?

A.03 This testimony, which is sponsored jointly by the Palmetto Alliance and the Carolina Environmental Study Group, addresses Emergency Planning Contention 11. That contention, as admitted by the Atomic Safety and Licensing Board in its Memorandum and Order of 29 September 1983, is worded as follows:

> The size and configuration of the northeast quadrant of the plume exposure pathway emergency planning zone (Plume EPZ) surrounding the Catawba facility has not been properly determined by State and local officials in relation to local emergency response needs and capabilities, as required by 10 CFR 50.47(c)(2). The boundary of that zone reaches, but does not extend past the Charlotte city limit. There is a substantial resident population in the southwest part of Charlotte near the present plume EPZ boundary. Local meteorological conditions are such that a serious accident at the Catawba facility would endanger the residents of that area and make their evacuation prudent. The likely flow of evacuees from the present plume EPZ through Charlotte access routes also indicates the need for evacuation planning for southwest Charlotte. There appear to be suitable plume EPZ boundaries inside the city limits, for example, highways 74 and 16 in southwest Charlotte. The boundary of the northeast quadrant of the plume EPZ should be reconsidered and extended to take account of these demographic, meteorological and access route considerations.

Q.04 What is the plume exposure pathway emergency planning zone?

A.04 The plume exposure pathway emergency planning zone ("plume EPZ") is an area surrounding a nuclear power plant for which emergency response plans are required in order to assure that prompt and effective actions can be taken to protect the public in the event of an accident from two principal pathways: (a) whole body external exposure to gamma radiation from the plume and from deposited materials, and (b) inhalation exposure from the passing radioactive plume. The plume EPZ should be about 10 miles in radius [NUREG-0396, pp. 27-28; NUREG-0654, Rev. 1, pp. 8-10].

Q.05 What is the overall objective of emergency response planning for nuclear power reactors?

A.05 The overall objective of emergency response planning for nuclear power reactors is to provide does savings (and in some cases immediate life savings) for a spectrum of accidents that could produce offsite doses in excess of Protective Action Guides[1/] [NUREG-0654, Rev. 1, p. 6].

Q.06 What protective actions for the general public are available to avoid or minimize exposures from the dose pathways of concern for the plume EPZ?

A.06 The principal protective actions available for the general public to avoid whole body and inhalation exposures are:

a. Evacuation -- expeditious movement of the population before plume passage to avoid exposure from a radioactive plume and exposure due to ground contamination by deposition from the plume;

b. Relocation -- expeditious movement of the population from contaminated areas after plume passage to avoid further exposure from ground contamination;

c. Sheltering -- expeditious movement of the population indoors before plume passage to reduce exposure from a radioactive plume and acute ground contamination by deposition from the plume, and to reduce inhalation exposure during plume passage (used in conjunction with relocation);

d. Respiratory protection -- use by the population of measures to reduce inhalation exposure during plume passage; and

e. Thyroid blocking -- use by the population (before plume passage) of potassium iodide to block the uptake of radioactive iodine by the thyroid gland.

The choice of protective actions in any given accident situation depends on a number of factors, including the magnitude and composition of the release from the plant (i.e., the source term), weather conditions at the time of and subsequent to the release, the amount of time available before plume passage, the distance of populated areas from the plant site, the speed with which various protective actions can be implemented, and the level of protection afforded by various protective actions.

Q.07 What is the spectrum of potential accidents at the Catawba Nuclear Station?

A.07 The spectrum of potential accidents at the Catawba Nuclear St-tion range from relatively trivial plant upsets thr ugh accidents involving severe core damage and large-scale melting of the core and subsequent breach of the containment. This spectrum of accidents is sometimes split into two large categories -- accidents within the design basis and accidents exceeded the design basis. Actual accident experience to date in nuclear power plants is briefly reviewed in the NRC Staff's Final Environmental Statement on the Catawba Nuclear Station (FES-Catawba) [NUREG-0921]. Cther references describe additional incidents in some detail in both commercial nuclear plants and experimental reactors [ORNL/NSIC-176; ORNL/NSIC-217 draft; and NUREG/CR-2497].

Q.08 What is the significance of this spectrum of potential accidents for emergency planning?

A.08 Nuclear power plants built in the U.S. are conservatively designed to respond to accidents as severe as design basis accidents without sustaining severe core damage. The general approach to this design process is based on the principal of providing multiple barriers to the release of fission products to the environment -- referred to as the "defense in depth" concept.

For the purposes of siting, extremely conservative design basis accident evaluations are mandated. The dose calculations for such evaluations are generally governed by the procedures set forth in a 1962 publication of the former U.S. Atomic Energy Commission [TID-14844]. Using a number of assumptions regarding the source term (i.e., the quantity and chemical form of radioactive materials available for release from containment), performance of engineered safety features, plume dispersion, and protective actions, calculated doses from design basis accidents must be demonstrated to be less than 25 Rem whole body and 300 Rem to the thyroid from iodine exposure for a two-hour period at the exclusion area boundary and the entire period of plume passage at the low population zone boundary.[2/]

In contrast, realistic evaluations of design basis accidents result in exposures significantly lower than these guideline levels. For example, the NRC Staff's FES-Catawba provides such calculated doses for design basis accidents at Catawba [NUREG-0921, p. 5-79]. The largest calculated doses for Catawba design basis accidents are 0.06 Rem whole body and 0.07 Rem to the thyroid at the exclusion area boundary. Not only are these doses significantly less than the siting guideline doses [10 CFR 100.11(a)(1) and (a)(2)], they are only small fractions of the Protective Action Guide doses (6% and 1.4%, respectively, for whole body and thyroid exposures).

Thus, even if these calculated doses are optimistic by a factor of ten, the estimated doses from a realistic evaluation of design basis accidents at Catawba will not exceed the Protective Action Guide doses at the exclusion area boundary. This observation leads to the conclusion that design basis accidents are not significant with respect to offsite emergency response.

As a practical matter, should a design basis accident actually occur, offsite officials may decide to implement precautionary protective measures such as sheltering or a limited evacuation of areas near the plant until conditions are stabilized and the potential for a release of radioactivity to the environment has diminished.

For accidents beyond the design basis, a range of possible offsite doses and consequences is possible. It is conceivable that a severe core damage accident could be successfully "bottled up" by the containment so long as containment heat removal systems function adequately and excessive amounts of noncondensible gases are not generated. On the other hand, accidents beyond the design basis could result in core melting and the release of radioactive materials to the environment ranging in quantity from trivial to very large. The magnitude of the release will depend upon the degree of core damage, the operating history of the core, the performance (or lack thereof) of engineered safety features, and the timing and mode of containment failure.

Q.09 What magnitude of radiation exposure could result from core melt accidents in which the containment fails in the absence of emergency response?

A.09 A recent report from Sandia National Laboratories provides one perspective on accidents involving core melt with containment failure. Using the release categories for a pressurized water reactor from the Reactor Safety Study (RSS) [WASH-1400, Appendix VI], Sandia calculated bounding doses from such releases. The dose calculations were carried out using the CRAC2 accident consequence model [NUREG/CR-2326; NUREG/CR-2552; and NUREG/CR-2901], and provided estimates of whole body and thyroid doses at a distance of one mile from the release point assuming no protective actions for 48 hours. The doses presented represent the "peak" or maximum calculated doses based on 100 weather sequences. The doses thus calculated were [NUREG/CR-2925, p. 34]:

RELEASE CATEGORY	WHOLE BODY DOSE (REM)	THYROID DOSE (REM)
PWR-7	1×10^{0}	5×10^{0}
PWR-6	6×10^{1}	2×10^{2}
PWR-5	1×10^{3}	8×10^{3}
PWR-4	5×10^{3}	3×10^{4}
PWR-3	2×10^{4}	2×10^{4}
PWR-2	7×10^{4}	7×10^{4}
PWR-1A	8×10^{4}	9×10^{4}

Obviously, these accumulated dose levels would not be permitted to accumulate -- protective actions would be implemented to reduce the doses. The results do point out the need for protective actions (compared with the Protective Action Guide dose levels of 1-5 Rem whole body and 5-25 Rem thyroid) in core melt accidents in which the containment fails. The results also indicate that sheltering is not an adequate long-term protective action in areas close to the site for the more severe release categories (this is due to both the large initial exposure during plume passage and the accumulation of exposure from radioactive materials deposited from the plume on the ground during plume passage).

Q.10 What are the implications of the above for emergency planning for reactor accidents?

A.10 It can be concluded from the above information that core melt accidents dominate public risk considerations, and therefore, to a considerable extent, drive the size and configuration of the emergency planning zone. This is in accord with prior conclusions of probabilistic risk assessments such as the Reactor Safety Study [WASH-1400] and a comparative risk evaluation of accidents within and exceeding the design basis [NUREG/CR-0603].

Indeed, NRC regulations and joint NRC/FEMA emergency planning guidance reference NUREG-0396 as providing the technical basis for the size of the plume EPZ. This report is in turn based to a significant extent on a related Sandia Laboratories report [NUREG/CR-1131]. The dose versus distance and accident consequence calculations presented in NUREG-0396 and NUREG/CR-1131 are explicitly based on the characteristics of core melt accident release categories from the Reactor Safety Study. Thus, we need to look to analyses of offsite doses and consequences for core melt accidents at Catawba to gain perspective on the size and configuration of the plume EPZ.

Q.11 Which reactor served as the model for the calculations in NUREG-0396 and NUREG/CR-1131?

A.11 The accident probabilities and release characteristics used in NUREG-0396 and NUREG/CR-1131 are based on the results of the Reactor Safety Study [WASH-1400] analysis of a pressurized water reactor. The Surry Unit 1 reactor served as the surrogate in that analysis for all pressurized water reactors in the U.S.

Q.12 Briefly describe the Surry Unit 1 reactor and contrast it with the Catawba Nuclear Station reactors.

A.12 Surry Unit 1 is a three-loop Westinghouse pressurized water reactor with a thermal power output of 2,441 MWt. The plant has a dry subatmospheric containment with a design pressure of 45 psig.

The Catawba reactors are four-loop Westinghouse pressurized water reactors with a thermal power output of 3,412 MWt. The Catawba plants have ice condenser containments with a design pressure of 15 psig.

There are differences in design and the number and type of equipment provided in the two plants. These differences can be determined by comparing the Final Safety Analysis Reports and Safety Evaluation Reports for the facilities.

Q.13 How do the differences between Surry Unit 1 and the Catawba Nuclear Station reactors affect their performance in severe core damage or core melt accidents?

A.13 The NRC Staff's FES-Catawba states that the design and operating characteristics of the two plants are similar

[NUREG-0921, p. 5-36]. This may be accurate for normal operating conditions.

For performance under severe core damage or core melt accidents, however, the performance of the two plants can be expected to be different. Ideally, a probabilistic risk assessment (PRA) of the Catawba reactors would demonstrate this quite well, but no such analysis of the Catawba reactors has been prepared.

The next best choice is a PRA performed on a facility similar to the Catawba reactors. A PRA of the Sequoyah Unit 1 reactor was prepared by Sandia National Laboratories for the NRC under the Reactor Safety Study Methodology Applications Program (RSSMAP) in 1980 [NUREG/CR-1659, Vol. 1]. Sequoyah Unit 1 is, like the Catawba reactors, a 3,411 MWt four-loop Westinghouse pressurized water reactor with an ice condenser containment.

It would be reasonable to expect similar performance under severe accident conditions for Catawba and Sequoyah. There are two potentially important caveats here. The first is that the Sequoyah RSSMAP study did not consider so-called "external events" as accident initiators -- e.g., earthquakes, hurricanes, fires, etc. Because the events classified as "external events" are site- and plant-specific, the effects of such accident initiators are likely to be different for the Catawba and Sequoyah plants, despite their similarities in design.

In addition, there may be plant-specific features for Catawba that would result in differences between Sequoyah and Catawba in severe accident performance. Nonetheless, absent a plant-specific PRA for the Catawba reactors, the RSSMAP PRA for Sequoyah represents the best available guidance as to the performance characteristics of the Catawba reactors under severe accident conditions.

The differences in severe accident performance between Surry Unit 1 and Sequoyah Unit 1 (and, to the extent that the plants are similar, Catawba Units 1 and 2) were clearly identified in the Sequoyah RSSMAP report:

* Accident sequences involving transients were found to be important for Surry (indeed, one of the three dominant sequences was TMLB', a station blackout sequence). Only one transient accident sequence appears in the list of dominant accident sequences for Sequoyah [NUREG/CR-1659, Vol. 1, pp. 7-25 and 9-10].

* Overpressure failure of the containment for sequences in which containment engineered safety systems operate was found to be far more likely for Sequoyah than for Surry due to the lower containment design pressure and smaller containment volume of Sequoyah [NUREG/CR-1659, Vol. 1, p. 9-11].

* Although both Surry and Sequoyah use Westinghouse reactors, plant differences are manifested in significantly different dominant accident sequences [NUREG/CR-1659, Vol. 1, p. 9-12].

* Plant systems and design features which are important to risk are different for Surry and Sequoyah [Ibid.].

* Unlike the Surry plant, core melt accidents at Sequoyah caused by failure of emergency coolant injection or emergency coolant recirculation can fail the containment due to generation of noncondensible gases (a result similar to the Peach Bottom boiling water reactor, also analyzed in the Reactor Safety Study) [Ibid.].

* Unlike the Surry plant, failure of containment cooling following a small LOCA does not lead to core melt at Sequoyah (core melt at Surry for such sequences was predicted to occur due to boiling of sump

water leading to cavitation of emergency core cooling system pumps) [Ibid.].

* While there were only four dominant accident sequences for Surry, there were nine for Sequoyah [NUREG/CR-1659, Vol. 1, p. 9-13].

* Containment base melt through sequences can occur before above ground containment failure for Surry, whereas for Sequoyah an above ground containment failure is predicted to always precede containment basemat melt through. Containment failure by overpressurization is predicted to be a certainty for core melt accidents at Sequoyah if other containment failure modes are avoided [NUREG/CR-1659, Vol. 1, pp. 8-2 and 8-12].

Q.14 Which results do you recommend using as a basis for emergency planning for Catawba, Surry or Sequoyah?

A.14 Due to the differences in severe accident performance between Surry and Sequoyah, and the similarities between Sequoyah and Catawba, I recommend (in the absence of plant-specific results for Catawba) using the Sequoyah RSSMAP results as a basis for emergency planning for Catawba.

Q.15 What are the implications of using the Sequoyah accident progression analyses for Catawba in the context of emergency planning?

A.15 Accident progression (timing) results for sixteen accident sequences at Sequoyah are found in the RSSMAP analysis [NUREG/CR-1659, Vol. 1, p. 8-8]. In three of these sequences, containment failure occurs in about an hour or less (including Event V, the interfacing LOCA, in which the containment is bypassed at the time of accident initiation due to the nature of the accident). For the remaining thirteen sequences, core melt and containment failure are complete within roughly four hour of accident initiation for seven of the thirteen.

Thus, ten of the sixteen sequences analyzed will be accompanied by containment failure within about four hours or less. The remaining six have times for core melt and containment failure ranging from about five hours to thirteen hours. The full results of this analysis are provided as an attachment to this testimony.

Another important consideration is that at least five of the sequences leading to containment failure within about four hours (and four of the nine dominant accident sequences, for which in some cases no explicit progression calculations were presented) are assigned to release categories involving substantial fractions of the core inventory of the iodine, cesium-rubidium, tellurium-antimony radionuclide groups. These radionuclide groups tend to dominate accident consequences.

NUREG-0654 provides guidance on plume transit times within ten miles, providing a range of one to four hours [NUREG-0654, Rev. 1, p. 17]. For a twenty mile distance, these values can be doubled to two to eight hours. The city of Charlotte is in the range of ten to twenty-five miles, with the distance proposed in the contention for the extension of the plume EPZ of seventeen miles. At seventeen miles, the approximate plume transit times range from one and a half to six hours.

When the core melt accident timing considerations are combined with the plume transit times, we obtain time periods ranging roughly from five and a half to ten hours from the beginning of the accident to the arrival of the plume in the vicinity of Charlotte (assuming the wind is blowing in the direction of Charlotte). In some cases, as with Event V, the time period will be shorter; in other cases, where the release does not occur until about thirteen hours, the time will be longer.

In many cases, however, the range of roughly five to ten hours will apply. This time period will be reduced by the time consumed in diagnosing the accident, and the time consumed in notifying the public of the need to take protective actions, and any delay time between notification and the beginning of the implementation of the protective actions by the general public.

A crude indication of the time consumed in diagnosing the accident is provided in the "warning" time values used in accident consequence calculations. For the Sequoyah ice condenser release categories [NUREG-0773, p. 40], the warning time (the time available between notification of offsite authorities and the time of release) ranges between thirty minutes and two hours.

These time periods are probably on the pessimistic side of a distribution of potential time periods required for accident diagnosis. This pessimism is due to the adoption since the analyses were performed of the use of "Emergency Action Levels" [NUREG-0654, Rev. 1, Appendix 1] and symptom-oriented emergency procedures. These features, if properly used, should shorten the time required to diagnose an accident and activate emergency plans.

Nonetheless, it must be considered unlikely that plant operators will diagnose an impending severe core damage or core melt accident until either some core damage indication is annunciated in the control room or there is a clear indication of the failure of key safety functions (e.g., emergency core cooling). Thus, the five to ten hour period indicated above for accident progression and plume transit does not indicate the amount of time available for the implementation of protective actions beyond the present plume EPZ -- the latter time period will be less than five to ten hours, perhaps considerably so depending upon the circumstances.

Q.16 What sources of information are available on accident likelihoods and accident consequences (both doses and health effects) which can aid in an evaluation of emergency planning for Catawba?

A.16 The principal sources of information of accident likelihoods are completed PRAs for pressurized water reactors in the U.S., and documents which provide summaries of such information. The principal sources of information on accident consequences are NUREG-0396, NUREG/CR-1131, and NUREG-0921.

Q.17 What is the range of core melt accident and large release likelihoods for pressurized water reactors in the U.S. based on PRA results to date?

A.17 PRA estimates of core melt and large release likelihoods for U.S. pressurized water reactors were summarized in a memorandum prepared for the NRC Commissioners in January 1983 [Dircks]. The results for core melt likelihoods range from about 1:500 to 1:25,000 per reactor year, a range of roughly a factor of 50 (there are large uncertainties in the individual estimates). The results for large release likelihoods (i.e., a release with the potential to cause early fatalities offsite given nominal emergency response assumptions) range from about 1:1,000 to about 1:250,000, a range of roughly a factor of 250 (there are large uncertainties in the individual estimates).

Q.18 Where do the Catawba reactors fall within these ranges?

A.18 Absent a plant-specific PRA, it is difficult to have substantial confidence in any particular estimate for the Catawba reactors. Given the apparent similarities between Catawba and Sequoyah, one might have some confidence that the results would not differ dramatically. Such a judgment must be tempered by the recognition that plant-specific design and operational differences have been found to be important to risk in each PRA done to date. Simply accepting the Sequoyah results as completely applicable to Catawba ignores the possibility that risk outliers may be present at Catawba.

Further, it should be noted that the range of core melt and large release likelihoods presented in A.16 above did not include so-called "external events" for many reactors. External events, such as earthquakes, hurricanes, tornadoes, fires, etc., have been analyzed for only a few pressurized water reactors to date (Indian Point Units 2 and 3, Zion Units 1 and 2, and Seabrook Units 1 and 2). In these cases, external events have been found to be risk significant (and sometimes dominate risk), although the results are very site- and plant-specific (for example, the risk posed by Indian Point Units 2 and 3 was different both in magnitude and in the specific accident sequences which dominated risk) [IPPSS].

At most, therefore, one might conclude that the risk posed by the Catawba reactors is reasonably approximated by the Sequoyah Unit 1 RSSMAP PRA for internal events (there are large uncertainties associated with such a judgment). It is worth noting that if we assume that all the pressurized water reactors analyzed in PRAs meet NRC regulatory requirements, the range of performance in severe accident conditions implied by the ranges of core melt and large release likelihoods suggests that meeting NRC regulatory requirements does not equate to any particular level of risk as estimated in a PRA.

Absent site- and plant-specific analysis, it is not possible to judge whether the influence of external events will affect the comparison between Sequoyah and Catawba, or whether there are risk outliers for the Catawba reactors which render the comparison less robust. For emergency planning purposes, however, the Sequoyah PRA results provide the best available guidance.

Q.19 What are the implications of accident consequence analyses for emergency planning at Catawba?

A.19 NUREG-0396 serves as the explicit technical basis for the size of the plume EPZ, and therefore represents a logical starting place. In responding to this question, consideration of consequences will be limited to whole body exposure to gamma radiation.

Figure I-11 from NUREG-0396 (attached to this testimony) [NUREG-0396, p. I-38] presents curves of the conditional probability of whole body dose versus distance for core melt accidents. These curves are explicitly based on the source terms and relative probabilities of the Reactor Safety Study release categories PWR-1 through PWR-7. The curves result from a probabilistic weighting of separate curves for each release category. The doses were calculated based on straight line plume trajectory and an assumption of no protective actions, and were calculated using the CRAC ("Calculation of Reactor Accident

Consequences") computer model developed for the Reactor Safety Study [WASH-1400, Appendix VI; NUREG-0340; NUREG/CR-3185].

From Figure I-11 of NUREG-0396 conclusions for Catawba are possible if the assumption is made that these results reasonably represent Catawba. This assumption is somewhat questionable since the results are for release characteristics and relative probabilities for Surry rather than for a reactor with an ice condenser containment. The release likelihoods for release categories PWR-1 through PWR-3, however, are not very different between the Surry and Sequoyah analyses (there are large differences for release categories PWR-4 and PWR-5). Another consideration is that the curves will be slightly conservative for Catawba since the WASH-1400 consequence calculations were carried out for a 3,200 MWt core, whereas the Catawba core is somewhat larger at 3412 MWt.

This reservation aside, given a core melt accident there is about a 30% likelihood (about one chance in 3) of exceeding the 1 Rem whole body PAG at 10 miles, and about a 20% likelihood (about 1 chance in 5) of exceeding the 5 Rem whole body PAG at 10 miles. Another way of stating this is that there is about 1 chance in 5 to 1 chance in 10 of needing to implement protective actions beyond the present 10-mile plume EPZ given a core melt accident.

Further, again based on Figure I-11 from NUREG-0396, there is about a 10% likelihood (one chance in 10) of exceeding a 50 Rem whole body dose at 10 miles; such a dose is a factor of ten greater than the upper bound whole body PAG dose of 5 Rem. The likelihood of exceeding a dose of 200 Rem whole body (which is in the range of early fatality threshold without medical intervention) at 10 miles is about 3% (about 1 chance in 30) given a core melt accident.

Additional perspective can be gained, however, by separating the PWR release categories into those involving direct releases to the atmosphere (i.e., PWR-1 through PWR-5) and those involving releases resulting from basemat melt through (i.e., PWR-6 and PWR-7). This was done in NUREG/CR-1131 [NUREG/CR-1131, Figures 5.2, 5.3, 5.9, and 5.10, attached to this testimony] for the mean (average over 91 weather sequences) and 95% (value equalled or exceeded in only one weather sequence out of twenty) cases.

Given a core melt accident with a basemat melt through release (examining Figures 5.2 and 5.3 from NUREG/CR-1131, using Curve A representing no protective actions), the average distance to which the 1 and 5 Rem whole body PAG doses will be reached is about 1-2 miles and 0.4 miles, respectively. In the 95% case, the distances are about 6 miles and 2 miles, respectively. In addition, in the 95% case (equalled or exceeded only 5% of the time), the distance to which a 50 Rem whole body dose is exceeded is about 0.2 miles.

Given a core melt accident with a release to the atmosphere (examining Figures 5.9 and 5.10 from NUREG/CR-1131, using Curve A representing no protective actions), the average distance to which the 1 and 5 Rem whole body PAG is reached is about 100 miles and 80 miles, respectively. Moreover, a 50 Rem whole body dose is reached at about 20 miles, and a 200 Rem whole body dose is reached at about 8 miles. In addition, a 500 Rem whole body dose (510 Rem is the so-called "LD-50/60" dose in WASH-1400, that dose sufficient to result in early fatalities to 50% of those exposed within 60 days) is reached at about 3 miles.

In the 95% case (equalled or exceeded only 5% of the time), the 1 and 5 Rem whole body PAG doses do not appear on the graph, but a 10 Rem dose is reached at about 100 miles. A dose of 50 Rem is reached at about 50 miles. A 200 Rem dose is reached at about 20 miles. A 500 Rem dose is reached at about 10 miles.

A very approximate overall perspective can be gained as follows. According to data contained in NUREG/CR-2239 [NUREG/CR-2239, p. A-21], the wind rose for Catawba (based on data from 6/30/71 through 6/30/72) would place winds blowing toward Charlotte from Catawba (compass headings of NNE, NE, and ENE) about 35% (3.5×10^{-1}) of the time.

Release categories PWR-1 through PWR-3 dominate the above relationships where the PWR-1 through PWR-5 releases are probabilistically weighted. Based on the Sequoyah RSSMAP PRA, the approximate likelihood of a PWR-1 through PWR-3 release is about 1 in 25,000 (4×10^{-5}) [Dircks; NUREG/CR-1659, Vol. 1, p. 9-13]. The overall core melt probability is about 1 in 17,000 per reactor year (6×10^{-5}). Thus, the conditional likelihood of a large release given a core melt is approximately 2 in 3 (6.7×10^{-1}).

Thus, combining the likelihood of a large release (PWR-1 through PWR-3) with the likelihood of the wind blowing in the direction of Charlotte at the time of the release, a very approximate overall likelihood of a large release occuring with the wind blowing toward Charlotte is about 1 in 72,000 per reactor year (1.4 x 10-5). In addition, combining the conditional likelihood of a large release given a core melt with the likelihood of a the wind blowing toward Charlotte at the time of the release, we obtain a conditional probability (given a core melt) of a large release with the wind blowing toward Charlotte of about 1 chance in 4 (2.3×10^{-1}).

On average (the mean case), when a large release occurs with the wind blowing toward Charlotte, the dose at 10 miles will be about 100 Rem whole body and the dose at 20 miles will be about 50 Rem whole body if no protective actions are taken. In the 95% case (with a likelihood of 1 chance in 20, or 5×10^{-2}), the dose at 10 miles will be about 500 Rem and the dose at 20 miles will be about 200 Rem. This case has an approximate overall likelihood (based on calculations above) of about 1 in 1.4 million and a conditional probability (given a core melt accident) of about 1 in 90 (1.1×10^{-2}).

The absolute probability values derived above are very uncertain, and assume that the results from the Sequoyah RSSMAP PRA are competely applicable to Catawba (which they may not be, but they are certainly more representative than Surry's results). The conditional likelihoods have less uncertainty (being dependent only upon the relative likelihood of a large release given a core melt and the likelihood of the wind blowing toward Charlotte), and are therefore more robust.

Q.20 What are the implications of the information provided in response to Q.19 for the configuration of the plume EPZ at Catawba?

A.20 Given a large release with the wind blowing toward Charlotte, even in the mean (average) case protective actions will be necessary beyond the existing 10 mile EPZ because whole body doses will be above the PAG levels in the absence of protective actions. Protective actions would also be needed beyond the existing 10 mile EPZ if the wind was blowing in any other direction from Catawba at the time of the release.

The question of whether Charlotte should be included within the plume EPZ (as opposed to other areas outside the plume EPZ) turns on the relative difficulty of implementing protective actions. In response to Q.15 above, I indicated that the time from accident initiation to the transit of the plume through a distance from 10-17 miles from Catawba would be roughly 5-10 hours. I also indicated that the actual time between when a warning could be given and plume transit would be less than the range of 5-10 hours, perhaps substantially so depending upon circumstances. Thus, the range of 5-10 hours would represent an optimistic upper bound case (i.e., with almost immediate warning to offsite authorities when the accident starts, an immediate decision to implement protective actions, and prompt communication of this information to the public).

In the worst case, assuming only minimal (30 minutes) warning time before the release occurs, the plume will complete its transit of the Charlotte area in about 2-6.5 hours. Further, the time available to implement protective actions will be reduced by the time consumed in notification of the general public of the need to take action. The length of time required to notify the residents of the city of Charlotte to take protective actions is open to speculation at this time (however, some fraction of the population will be watching television or listening to the radio at any given time and will receive broadcast warnings; further, fire and civil defense sirens could be sounded, and police and other emergency vehicles with sirens could be pressed into service). Emergency planning and active public education could improve notification times.

Given these considerations, for some accidents (namely, those in which containment failure occurs within about four hours or less of the start of the accident) it does not appear that evacuation would be a feasible option. Assuming the population delays one hour before evacuating [NUREG-0921, p. F-3], more time will be lost between the start of the accident and plume transit of Charlotte. Evacuation efforts would need to be concentrated within the existing 10-mile EPZ where the residents of that area are at greater risk (due to higher exposure levels).

However, as Figures 5.9 and 5.10 from NUREG/CR-1131 demonstrate, sheltering with relocation six hours after plume passage provides roughly equivalent protection to evacuation. Curves B and D represent sheltering with different sheltering factors, and Curves C and E represent evacuation at an effective speed of 10 mph (the NRC Staff's consequence estimates in NUREG-0921 assume an effective speed of 6.7 mph based on evacuation time estimates for the existing 10 mile EPZ) with delay times of five and three hours, respectively.

Even the least favorable of these four emergency reponse sets provides dose reductions of a factor of about 3-5 for the mean case (given an atmospheric release) and a factor of about 3 for the 95% case in the 10-20 mile distance interval. The least favorable set assumes sheltering with shielding factors of 0.75 for cloud exposure and 0.33 for ground exposure). The most favorable shielding factors assumed were 0.5 for cloud exposure and 0.08 for ground exposure.

According to NUREG/CR-2239 [NUREG/CR-2239, pp. A-5 and A-7], Catawba was placed into a sheltering region with shielding factors of 0.6 and 0.2 for the Sandia siting study calculations. Thus, the actual sheltering result for Catawba would lie somewhere between curves B and D on Figures 5.9 and 5.10 in NUREG/CR-1131.

Doses might be reduced further if infiltration of radioactive particulates can be minimized by shutting down ventilation systems, moving to basements or the interior areas of buildings, and blocking cracks in doorways with cloth or paper. Inhalation doses could be reduced further with ad hoc respiratory protection [NUREG/CR-2272]. These measures should be evaluated in more depth. Implementation of such measures would require an adequate program of public education.

These considerations suggest that an emergency plan for Charlotte should consider sheltering with prompt relocation from contaminated areas after plume passage for the relatively fast-moving accidents. For accidents in which the containment is not projected to fail for ten hours or more, evacuation appears to be a more realistic alternative.

Q.21 What should be the principal considerations for an emergency plan for Charlotte involving nuclear accidents at Catawba?

A.21 Several key considerations emerge from the above discussions. First, redundant communications links with the utility and other offsite emergency response organizations are needed. Second, prompt access to radiation monitoring equipment is needed to locate contaminated areas from which prompt relocation must occur and to avoid having persons relocating after plume passage into contaminated areas (airborne monitoring from a helicopter would be a good choice if available). Third, some consideration should be given to possible egress routes to facilitate relocation and evacuation. Fourth, consideration needs to be given to means of public notification and the content of emergency messages (this requires liason with local media).

Public education is most important, not only so that the public will know what may be expected of them, but so that if the recommended protective action is sheltering, the public will understand the benefits of sheltering and relocation, and understand the reasons why this option has been selected. The latter is very important since vehicles provide essentially no shielding against gamma radiation and minimal protection against infiltration of radioactive particulates, and it is most undesirable to have people in vehicles in a traffic queue be overtaken by a radioactive plume.

An emergency plan incorporating these features for Charlotte need not be painstakingly detailed or extremely expensive. Existing emergency plans may already incorporate some of the functions required, and the remainder could be developed without significant expenditure of resources. What is required is a recognition of the need for the plan, the benefits which could derive from it in the event of an accident, and a commitment from the city of Charlotte, the Applicant, and Federal, state, and local planners to cooperate in the development of a plan for Charlotte and its integration into the overall emergency plan.

Q.22 What are your conclusions regarding the necessity of extending the plume EPZ to include the city of Charlotte?

A.22 Based on considerations of the possible performance of the Catawba reactors under core melt accident conditions, the conditional likelihood of a severe release occuring with the wind blowing toward Charlotte given a core melt accident, the benefits which can be obtained from the implementation of even minimal protective actions, and the modest effort involved, I recommend that the plume EPZ be extended as recommended in the contention.

As a practical matter, the planning done for the 10-17 mile area of Charlotte will be applicable to the remainder of the city as well. The preparation of such a plan will have a salutary effect as well -- the planning for sheltering and relocation for radiological emergencies will to a great extent be useful in other emergencies (such as those involving toxic materials spills).

END NOTES

Protective Action Guides (PAGs) are projected doses -- doses that would be received by the population of no protective actions are taken -- established by the U.S. Environmental Protection Agency (EPA) in 1975 for exposure to airborne materials released in nuclear accidents. For exposure of the general population to whole body gamma radiation, the EPA has established a range of PAGs from 1 to 5 Rem whole body exposure. For thyroid exposure of the general population, the EPA has established a range of PAGs from 5 to 25 Rem thyroid exposure. According to EPA guidance, the lower range of these PAGs should be used when there are no major local constraints in providing protection against exposure, especially to sensitive populations. In no case, however, should the upper range of these PAGs be exceeded in determining the need for protective action. The PAG doses do not include that dose which has unavoidably occurred prior to making dose projections [EPA 520/1-75-001, pp. 2.1-2.8].

2/ Among the assumptions made are: (a) a source term consisting of 100% of the core inventory of noble gases, 50% of the core inventory of iodine, and 1% of the remaining core inventory, (b) no consideration of natural attenuation processes in containment, (c) no consideration of the impact of engineered safeguards features such as containment sprays on fission product behavior, (d) containment isolation and leakage at a constant 0.1% per day, (e) time invariant fifth percentile meteorology, and (f) no protective actions for the exposed population.

R E F E R E N C E S

DIRCKS
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Table 8-3

MARCH Results for Ice Condenser PWR Accident Sequences[a]

SEQUENCE	ECCS START	ECCS STOP	CSS START	CSS STOP	CORE UNCOVERY	CORE MELT START	CORE MELT END	VESSEL FAILURE	ICE MELT COMPLETE	CONT FAILURE	CONCRETE MELT START
AD-α	-	-	1	-	[illegible]	5	20	46	[illegible]	20	46
AD-γ	-	-	1	66	1	5	18	66	>1[illegible]	66	67
AD-δ	-	-	1	361	1	5	18	66	>1[illegible]	361	67
AHF-δ	1	129	1	129	177	202	225	269	[illegible]	219	269
S_1HF-δ	1	126	1	126	[illegible]	177	215	259	[illegible]	219	269
S_2D-δ	-	-	1	703	46	62	118	161	[illegible]	703	182
S_2H-γθ	1	50	1	110	62	80	96	110	[illegible]	110	260
S_2H-δθ	1	50	1	798	62	80	96	110	[illegible]	798	260
S_2HF-α	1	109	1	109	128	152	180	206	[illegible]	180	324
S_2HF-δ	1	109	1	109	128	152	180	193	[illegible]	301	193
S_2HF-δθ	1	109	1	109	128	152	180	193	[illegible]	197	314
TMLB'-δ	-	-	-	-	184	200	232	238	[illegible]	328	238
TMLB'-δθ	-	-	-	-	184	200	232	238	[illegible]	244	384
TML-γ	238	-	1	238	184	200	232	238	>1[illegible]	238	238
TML-δ	238	-	1	660	184	200	232	238	>1[illegible]	[illegible]	238
V[b]	-	-	-	-	[illegible]	38	57	91	[illegible]	-	91

[a] All times in minutes

[b] Ice bed bypassed after steam explosion

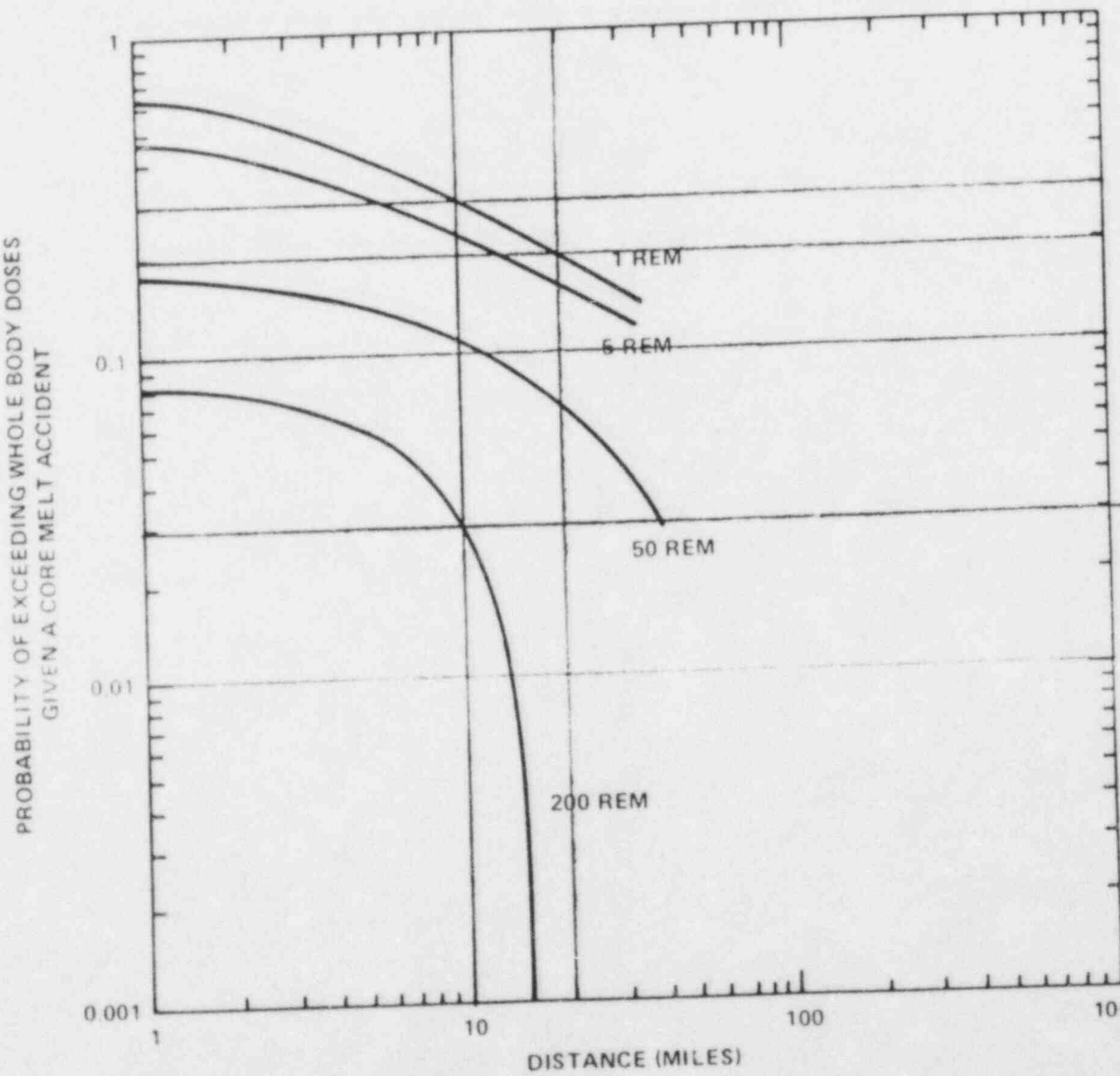

Figure I-11. Conditional Probability of Exceeding Whole Body Dose Versus Distance. Probabilities are Conditional on a Core Melt Accident (5×10^{-5}).

Whole body dose calculated includes: external dose to the whole body due to the passing cloud, exposure to radionuclides on ground, and the dose to the whole body from inhaled radionuclides.

Dose calculations assumed no protective actions taken, and straight line plume trajectory.

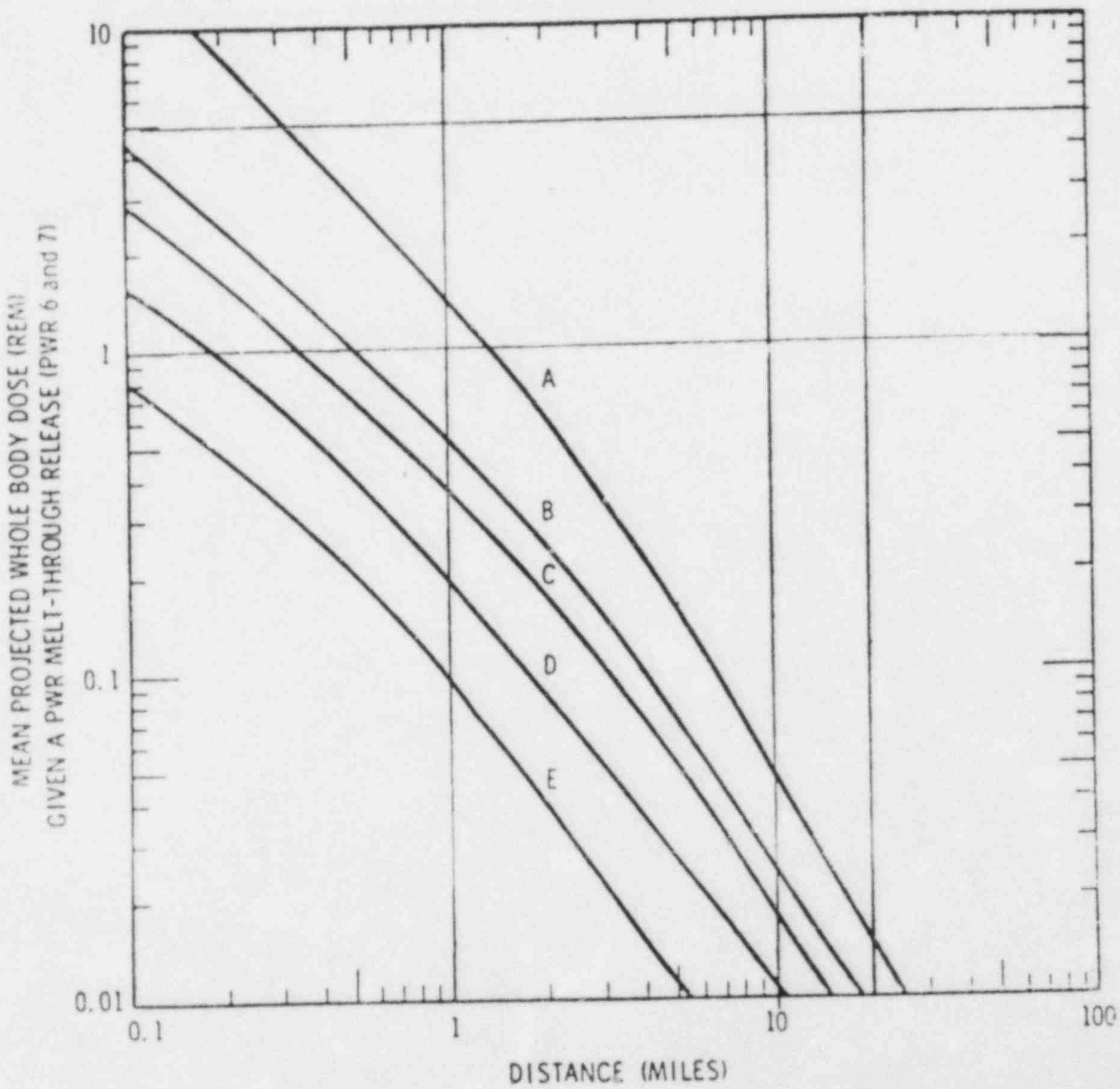

Figure 5.2 Conditional Mean Projected Whole Body Dose Versus Distance for Sheltering and Evacuation Strategies. Projected Doses are Conditional on a PWR "Melt-Through" Release (PWR 6 and 7).

Curve A Individual located outdoors without protection. SF's (1.0, 0.7). 1-day exposure to radionuclides on ground.

Curve B Sheltering, SF's (0.75, 0.33), 6-hour exposure to radionuclides on ground.

Curve C Sheltering, SF's (0.5, 0.08), 6-hour exposure to radionuclides on ground.

Curve D Evacuation, 5 hour delay time, 10 MPH.

Curve E Evacuation, 3 hour delay time, 10 MPH.

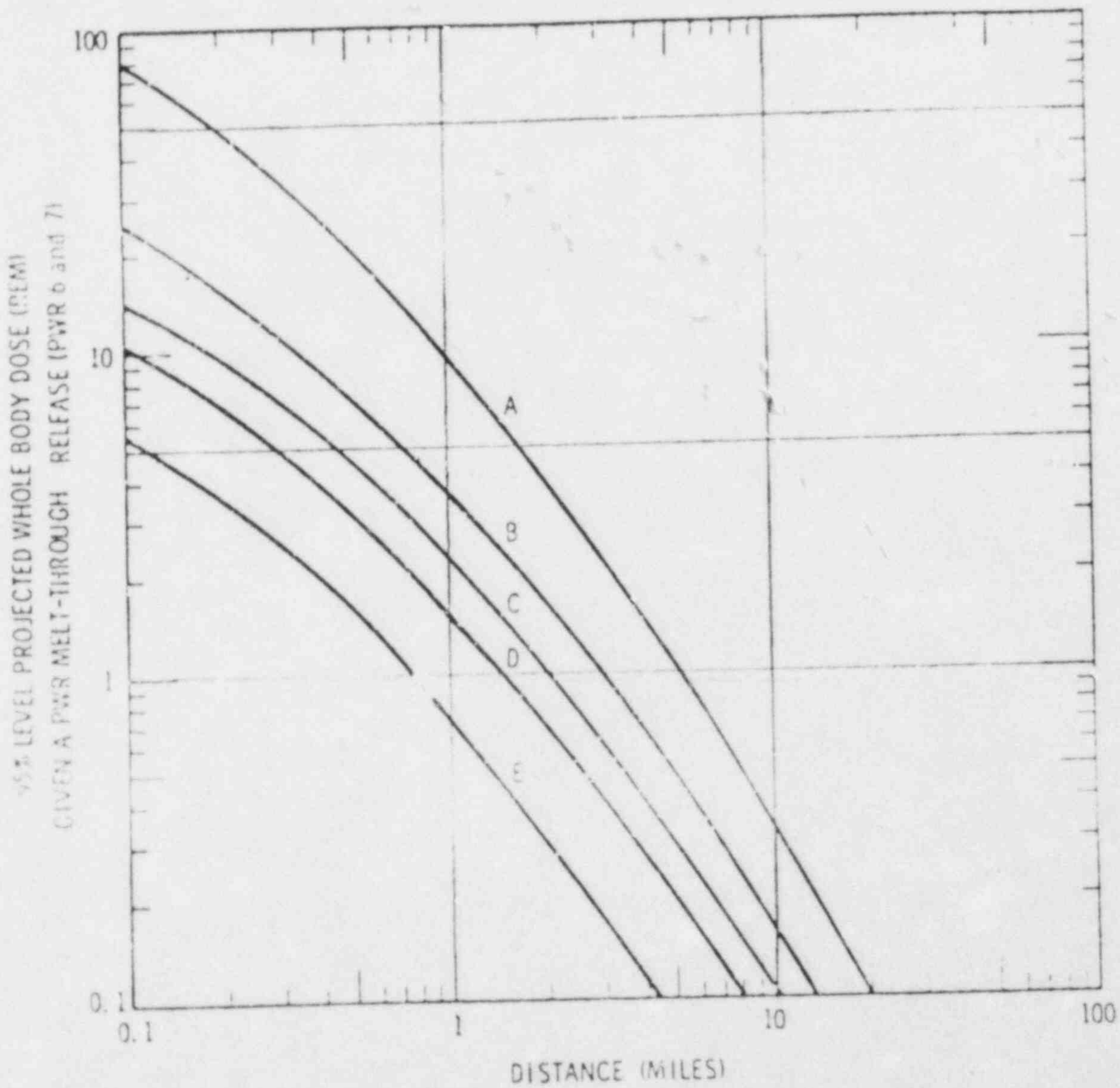

Figure 5.3 Conditional 95% Level Whole Body Dose Versus Distance for Sheltering and Evacuation Strategies. Projected Doses are Conditional on a PWR "Melt-Through" Release (PWR 6 and 7).

Curve A Individual located outdoors without protection. SF's (1.0, 0.7). 1-day exposure to radionuclides on ground.

Curve B Sheltering, SF's (0.75, 0.33), 6-hour exposure to radionuclides on ground.

Curve C Sheltering, SF's (0.5, 0.08), 6-hour exposure to radionuclides on ground.

Curve D Evacuation, 5 hour delay time, 10 MPH.

Curve E Evacuation, 3 hour delay time, 10 MPH.

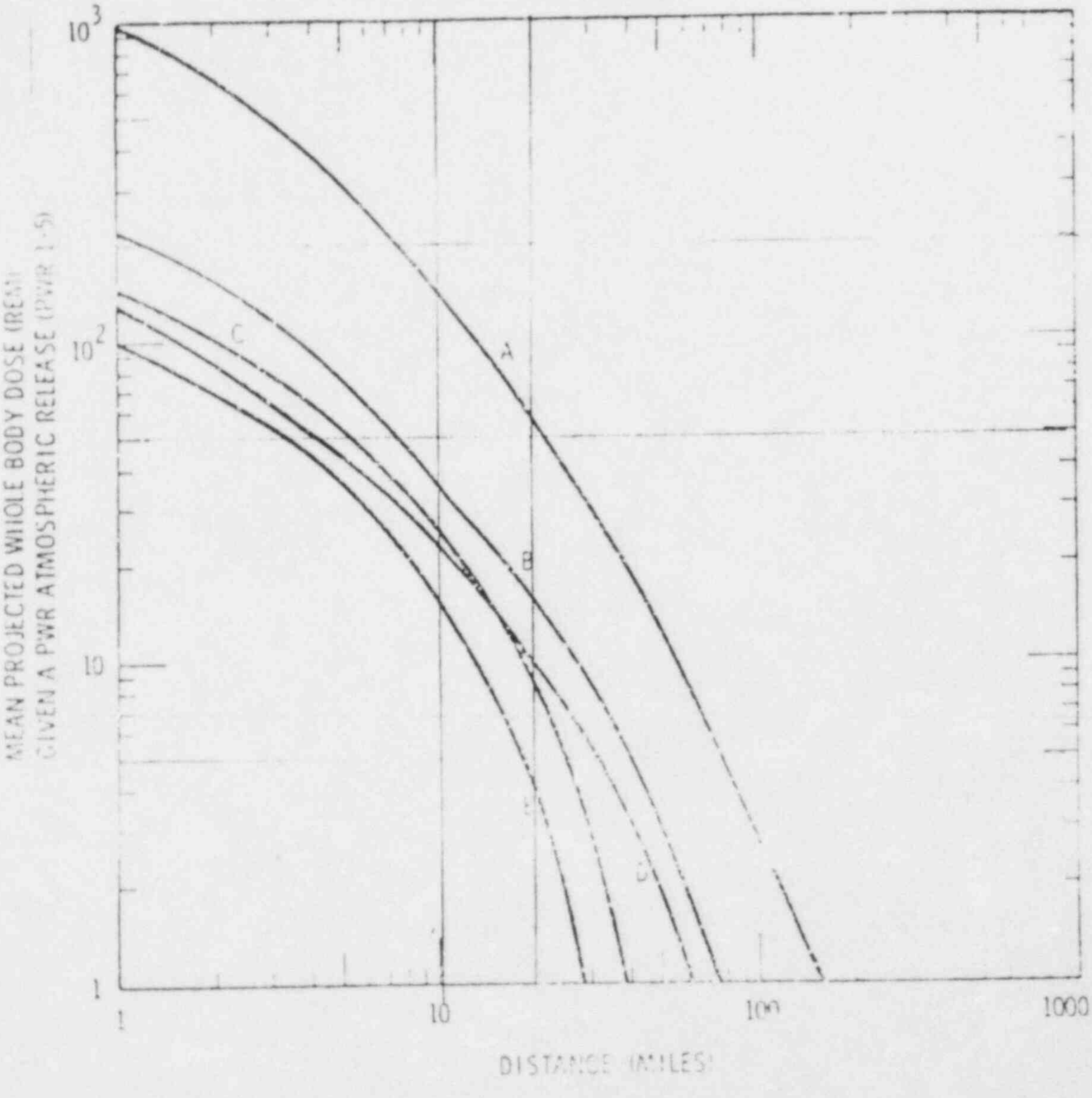

Figure 5.9 Conditional Mean Projected Whole Body Dose Versus Distance for Sheltering and Evacuation Strategies. Projected Doses are Conditional on a PWR "Atmospheric" Release (PWR 1-5).

Curve A Individual located outdoors without protection. SF's (1.0, 0.7). 1-day exposure to radionuclides on ground.

Curve B Sheltering, SF's (0.75, 0.33), 6-hour exposure to radionuclides on ground.

Curve C Evacuation, 5 hour delay time, 10 MPH.

Curve D Sheltering, SF's [illegible], 6-hour exposure to radionuclides on ground.

Curve E Evacuation, 3 hour delay time, 10 MPH.

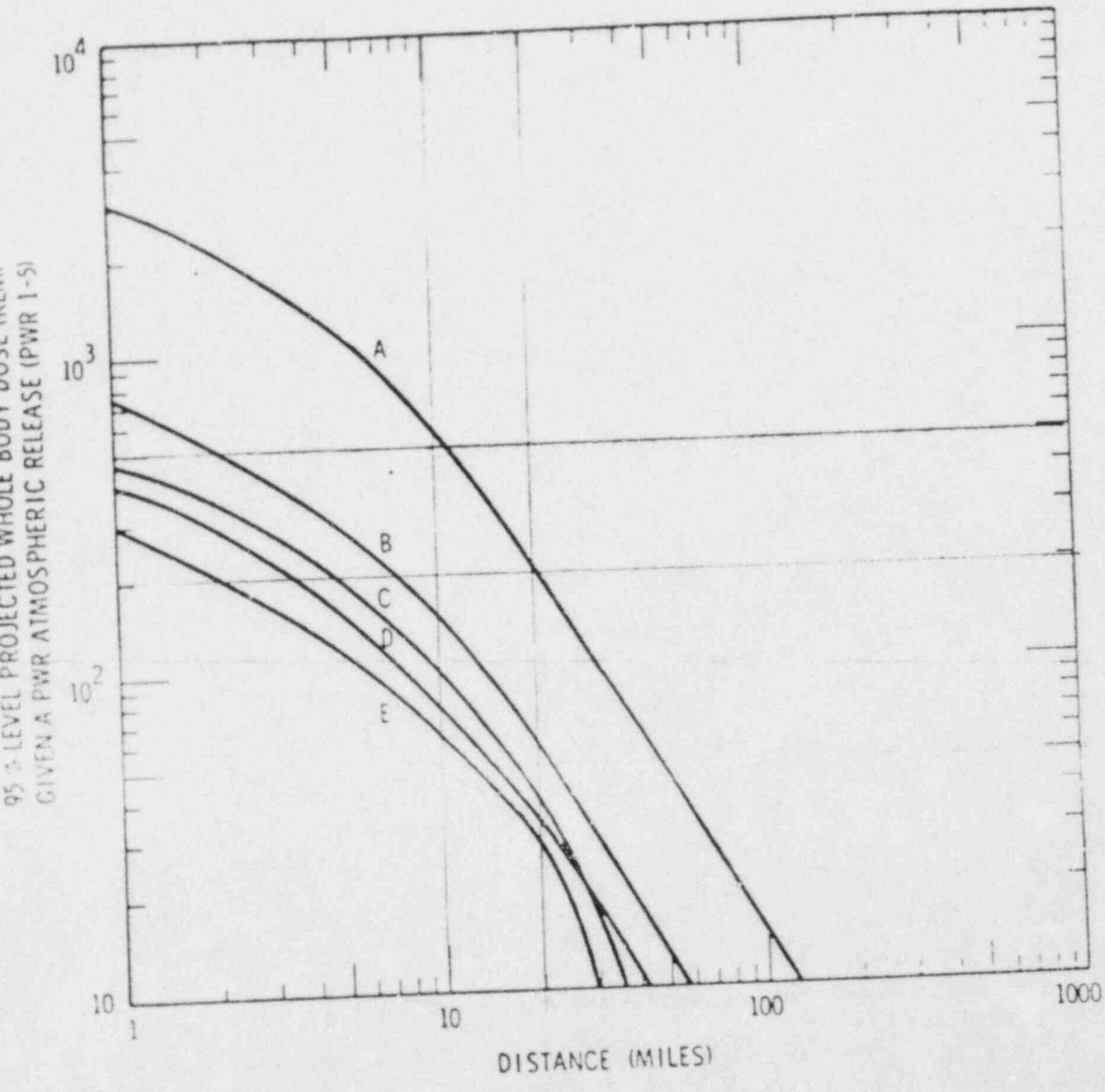

Figure 5.10 Conditional 95% Level Projected Whole Body Dose Versus Distance for Sheltering and Evacuation Strategies. Projected Doses are Conditional on a PWR "Atmospheric" Release (PWR 1-5).

Curve A Individual located outdoors without protection. SF's (1.0, 0.7). 1-day exposure to radionuclides on ground.

Curve B Sheltering, SF's (0.75, 0.33), 6-hour exposure to radionuclides on ground.

Curve C Evacuation, 5 hour delay time, 10 MPH.

Curve D Sheltering, SF's (0.5, 0.08), 6-hour exposure to radionuclides on ground.

Curve E Evacuation, 3 hour delay time, 10 MPH.

STATEMENT OF PROFESSIONAL QUALIFICATIONS -- STEVEN C. SHOLLY

My name is Steven C. Sholly. I am a Technical Research Associate with the Union of Concerne Scientists (UCS), Dupont Circle Building, 1346 Connecticut Avenue, N.W., Washington, D.C. 20036. I joined the UCS staff in February 1981.

My primary responsibilities are UCS are technical and policy analysis concerning probabilistic risk assessment and radiological emergency planning. In addition, I monitor nuclear safety research in several other ares, including severe accident research, accident mitigation systems, and alternative reactor designs. I am also a regular contributor to UCS's newsletter, Nucleus.

Prior to joining UCS, I served as Research Coordinator and Project Director of the TMI Public Interest Resource Center (TMIPIRC) in Harrisburg, Pennsylvania. TMIPIRC was created after the Three Mile Island accident by concerned citizens groups in Pennsylvania. At TMIPIRC, I was responsible for directing research and public education activities associated with the proposed restart of TMI Unit 1 and the cleanup of TMI Unit 2.

In addition to this experience, I taught secondary school science for two years. I also have two years experience in wastewater treatment, including experience as Chief Process Operator of a 5.0 MGD tertiary treatment facility. In the latter capacity, I obtained state certification to operate activated sludge wastewater treatment plants (Pennsylvania Class B, Type 1 certification).

I have provided testimony before Congress and a special committee of the New York State Assembly on radiological emergency planning matters. I have also testified before

Congress on safety issues associated with steam generators in pressurized water reactors.

During the Indian Point Units 2 and 3 Special Investigation in 1983, I provided expert testimony on behalf of UCS and NYPIRG on filtered vented containment systems (jointly with Dr. Gordon Thompson), severe accident consequences, and comparative risk analysis of nuclear power plants. Most recently, I provided supporting evidence (principal evidence by Dr. Gordon Thompson) on emergency planning and probabilistic risk assessment in the Sizewell B Inquiry in the United Kingdom on behalf of the Town and Country Planning Association.

I am a 1975 graduate of Shippensburg State College (now Shippensburg University), Shippensburg, Pennsylvania. I received a B.S. degree in Education (majors in Earth and Space Science and General Science, and minor in Environmental Education). I have also completed graduate coursework in land use planning. I am a resident of Columbia, Maryland.

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NUCLEAR REGULATORY COMMISSION DOCKETING & SERVICE
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BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	Docket Nos. 50-413
DUKE POWER COMPANY, ET AL.	)	50-414
	)	
(Catawba Nuclear Station, Units 1 and 2)	)	16 April 1984

CERTIFICATE OF SERVICE

I hereby certify that copies of PALMETTO ALLIANCE AND CAROLINA ENVIRONMENTAL STUDY GROUP TESTIMONY OF STEVEN C. SHOLLY ON EMERGENCY PLANNING CONTENTION NUMBER ELEVEN in the above captioned matter have been served upon the following by deposit in the United States mail this 16th day of April 1984.

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