October 3, 1983

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CAROLINA POWER & LIGHT COMPANY	)	Docket Nos. 50-400 OL
AND NORTH CAROLINA EASTERN	)	50-401 OL
MUNICIPAL POWER AGENCY	)	
	)	
(Shearon Harris Nuclear Power	)	
Plant, Units 1 and 2)	)	

AFFIDAVIT OF JOHN J. MAURO AND GUY MARTIN, JR. IN SUPPORT OF APPLICANTS' MOTION FOR SUMMARY DISPOSITION OF INTERVENOR WELLS EDDLEMAN'S CONTENTION 29/30

City of New York)
) ss.
State of New York)

John J. Mauro, being duly sworn, deposes and says:

1. I am currently Director of the Radiological Assessment and Health Physics Department of Envirosphere Company, a division of Ebasco Services, Inc. My business address is Two World Trade Center, New York, N.Y. The statement of my background and qualifications is attached hereto as Attachment 1.

I have personal knowledge of the matters stated herein and believe them to be true and correct.

Guy Martin, Jr., being duly sworn deposes and says:

2. I am currently Manager of the Radiological Assessment and Health Physics Department of Envirosphere Company, a division of Ebasco Services, Inc. My business address is Two World Trade Center, New York, N.Y. The statement of my background and qualifications is attached hereto as Attachment 2. I have personal knowledge of the matters stated herein and believe them to be true and correct.

3. This Affidavit has been prepared in support of Applicants' Motion for Summary Disposition of Intervenor Wells Eddleman's Contention 29/30. The material and calculations contained within this affidavit have been prepared by us or under our direct supervision. The affidavit contains three sections. The first section consists of a general introduction to Appendix I compliance calculations. The purpose of this first section is to describe the general procedure by which Appendix I compliance calculations are made.

The second section describes the methodology of Applicants' radioiodine calculations used to demonstrate compliance with Appendix I to 10 C.F.R. Part 50. The purpose of the second section is to present the details of how Applicants' Appendix I compliance calculations were developed. This section is generally divided into discussions of: the

calculated radionuclide release rate (i.e., the Source Term), the dispersion of the radionuclides in the environment (the Dispersion Calculation) and the calculated dose to members of the general public (the Dose Calculation).

The third section of this affidavit compares the measured radioiodine releases at operating nuclear power plants with the releases predicted for each plant, prior to operation, using standardized calculational methods. The purpose of the third section is to demonstrate the reliability, and generally conservative nature, of the standard techniques Applicants used to predict radioiodine source terms at the Shearon Harris Plant.

I. General Introduction to Appendix I Compliance Calculation

4. Appendix I to 10 C.F.R. Part 50 provides numerical guides for design objectives and limiting conditions of operation to assist Applicants for, and holders of, licenses for light water cooled nuclear power reactors in meeting the requirement that radioactive material in effluents to unrestricted areas be kept as low as is reasonably achievable ("ALARA"). Compliance with Appendix I involves a demonstration that the plant design provides reasonable assurances that the liquid and gaseous effluents released will be below levels resulting in offsite exposures in excess of the Appendix I design objectives.

5. The methodology by which this demonstration of compliance is performed has been standardized and is set forth in a series of NRC regulatory guides which have been approved by the Advisory Committee on Reactor Safeguards; Regulatory Guides 1.109, 1.111, 1.112 and 1.113.1/ In addition, the NRC has an ongoing data collection and assessment program to determine whether changes to the standardized methods are warranted.

6. The methodology used to calculate the Source Term is described in Regulatory Guide 1.112 and NUREG-0017 and is implemented by a standard computer code called the Gaseous and Liquid Effluent (GALE).

7. Regulatory Guide 1.112, issued in April, 1976, states that NUREG-0017, and the associated GALE Code, is an acceptable method for calculating annual average releases of radioactive material from nuclear power reactors. Since that time, the GALE, or other codes developed from NUREG-0017, have been widely accepted and used almost exclusively throughout the nuclear industry to show compliance with 10 C.F.R. Part 50, Appendix I. The releases determined by GALE are based on:

1/ Regulatory Guide 1.109, "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents For The Purpose of Evaluating Compliance With 10 C.F.R. Part 50, Appendix I"; Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors"; Regulatory Guide 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors"; Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents From Accidental And Routine Reactor Releases For The Purpose of Implementing Appendix I".

(1) Standardized primary and secondary coolant activities derived from American Nuclear Society (ANS) 18.1 Working Group recommendations,

(2) The release and transport mechanisms that result in the appearance of radioactive material in waste streams, and

(3) Plant-specific design features used to reduce the quantities of radioactive materials ultimately released to the environment.

8. The methodology is based on prior operating experience and, as such, takes into account the full range of normal operating experience, including anticipated operational occurrences. Anticipated operational occurrences include equipment leakages and malfunctions which, based on prior industry experience, may be expected to occasionally occur and therefore need to be considered in the Source Term calculation. Though best estimates are appropriate, Attachments 3 and 4 reveal that the I-131 source terms predicted using the methods are on the average many times greater than actual operating experience. The first step is to calculate the Source Term, or the radionuclides released in liquid and gaseous effluents during normal operation. The second step is to calculate the atmospheric dispersion and aquatic dilution of released radionuclides calculated to be the Source Term. These methods have also been standardized as set forth in Regulatory Guide 1.111 for gaseous effluents and Regulatory Guide 1.113 for liquid effluents. The third and final step of the methodology

is to calculate the radiation doses to the general public attributable to the radionuclides dispersed in the environment. This step includes both calculation of concentrations in environmental pathways and the radiological doses to receptors. The methods used to calculate these doses are described in Regulatory Guide 1.109 and are implemented by standard computer codes entitled GASPAR and LADTAP. Contention 29/30 and this affidavit are solely concerned with the radioiodines in the Source Term.

9. Appendix I also requires the Applicants to establish an effluent and environmental monitoring program which is used to:

(1) quantify the radionuclide release rates to the environment,

(2) determine the concentration of radionuclides in important components of the environment, such as in air, water and foods, and

(3) calculate the resulting radiation exposures to the general population.

These monitoring programs have been standardized by the NRC in NUREG-0133 "Preparation of Radiological Effluent Technical Specifications For Nuclear Plants" and NUREG-0472 "Radiological Effluent Technical Specifications For PWR'S" and are implemented under the Shearon Harris Plant Environmental

Radiological Technical Specifications which delineate methods for setting radionuclide release limits and monitoring the releases to ensure compliance with those limits. These specifications also establish reporting levels for concentrations of radionuclides in the environment and surveillance methods to determine if and when these concentrations are exceeded.2/

10. Both the calculational methods used to assess compliance with the Appendix I design objectives, and the effluent and environmental monitoring program used to demonstrate compliance with the Limiting Conditions of Operation, must provide assurances that the calculated doses have not been significantly underestimated. Unless pathways ignored in the calculations have the potential to contribute significantly to the exposures, they need not be included in the dose calculation. Regulatory Guide 1.109 requires an assurance that at least 90% of the exposure is accounted for, in which case the analysis is considered complete.

11. Regulatory Guide 1.109 includes 14 equations and over 4000 calculational parameters, many of which are the subject of

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2/ Applicants are in the process of developing radiological technical specifications and procedures for the Shearon Harris Plant which conform to the guidelines contained in NUREGs-0133 and 0472. The technical specifications will be submitted to the NRC for approval as part of the licensing process. The plant operational procedures are also under development and will be submitted to the NRC Office of Inspection and Enforcement for review and approval.

continual research. As a result, at any point in time, some parameters may be marginally too high, or marginally too low. This may apply to generic parameters, such as dose conversion factors, or to site specific factors, such as bioaccumulation factors, but the standard methods in Regulatory Guide 1.109 are based on the necessity to ensure that the doses calculated using the standard methods will be reasonable, taken as a whole, and that there is assurance that the calculated doses are not significantly underestimated. Independent reviews of the methods outlined in Regulatory Guides 1.109, 1.111, 1.112 and 1.113 have revealed that although there are isolated parameters which may be considered liberal, they are compensated for by numerous assumptions and parameters which have been found to be conservative.

12. The calculational methods described in the above referenced NRC Guidelines are designed to result in a realistic estimate of radiation exposures.

13. This methodology has been used by both the NRC and the Applicants to independently confirm compliance by the Shearon Harris Nuclear Power Plant ("SHNPP") with the design objectives of Appendix I to 10 C.F.R. Part 50. The NRC, in the Draft Environmental Statement (DES) at § 5.9.3.2 and Appendix D, and Applicants, in the Environmental Report (ER) at § 5.2, have both concluded that the Shearon Harris Plant is in compliance with the design objectives of Appendix I. We

supervised Applicants' Appendix I calculations and § 5.2 of the Environmental Report was prepared under our direction. We have reviewed Chapter 5.9 and Appendix D of the DES. In our opinion, the calculations of both Applicants and the NRC staff demonstrate compliance with Appendix I.

II. Applicants' Appendix I Compliance Methodology

A. The Source Term

14. The demonstration of Appendix I compliance initially requires the development of a Source Term according to the methodology of NUREG-0017. Among the factors considered by NUREG-0017 are the radionuclide concentration values for the primary coolant. The standard primary coolant concentration values recommended by the NRC for the purpose of assessing compliance with Appendix I reflect average failed fuel experience over a 3-1/2 year period (1970 to 1973) at 18 operating PWRs. The experience shows that the percent failed fuel ranged from .01% to .73% with an average of 0.12%. Accordingly, the standard calculation is based on the assumption that Shearon Harris will experience average failed fuel of about 0.12% over its operating life. However, over any one year period, the percent of failed fuel could be higher or lower.3/

3/ The radioiodine concentration level due to failed fuel is indirectly limited by the plant technical specifications and Limiting Conditions of Operation ("LCOs") established for the primary and secondary coolant. These specifications and LCOs control operational procedures which can reduce the concentra-

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15. The NRC has an ongoing program to calculate trends in fuel performance and the average percent of failed fuel is declining due to improvements in fuel fabrication and management methods and water chemistry control. For example, a review of fuel performance from 1978 to 1981 reveals an average failed fuel of about .01%. NUREG-0633, "Fuel Performance Annual Report for the Period through December, 1978"; NUREG/CR-1818, "Fuel Performance Annual Report, 1979"; NUREG/CR-2410, "Fuel Performance Annual Report, 1980"; NUREG/CR-3001, "Fuel Performance Annual Report, 1981". As a result, the standard methodology is somewhat conservative as borne out by operating experience which generally shows lower releases than predicted, as seen in Attachments 3 and 4.

16. An allegation has been made that Applicants' Source Term fails to account for radioiodines resulting from decay of xenon. Xenon isotopes produced in the core do not decay to iodines; rather iodines decay to xenon. Xenon levels build up during power transients and result in small pertubations in power levels.4/ The annual Source Term for iodine depends on

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tions. For the purposes of the NUREG-0017 standard calculation, a specific set of conditions and power levels is assumed for "normal operations" to arrive at a radioiodine release estimate.

4/ The coolant is continually degassed to draw off the xenon and, as indicated in NUREG-0017, this degassing operation is included in the Source Term derivation.

the average concentration of iodine in the coolant. This concentration is related to the average annual power production rate. Small pertubations in power level as a function of time are expected and are reflected in the average.

17. Regulatory Guide 1.109 states that only those pathways which contribute an additional dose increment equal to or more than 10 percent of the total dose need to be considered. As seen in Attachment 5, the only radioiodine which contributes over 10% to the offsite dose, relative to I-131, is I-133. The total offsite impact from I-130, I-132, I-134 and I-135 will be less than 7.1% of the impact from I-131 and I-133. This value is the upper limit to the contribution of the offsite dose by the neglected radioiodines: the actual impact will be less. The reason is this methodology neglects the decay of the radioiodines while in transit to receptor locations. Since I-131 has a longer half-life than the other radioiodines, the actual ratio of the activity of other radioiodine isotopes to I-131 will be less in the environment than in the core.

18. Applicants' release estimates are based on the calculated primary coolant concentrations. These, in turn, are based on actual operating experience at nuclear power plants. Thus, regardless of how the radioiodines are generated, they are properly accounted for. Most, in fact, do result from the decay of the primary fission products.

19. It is conceivable, however, that radionuclides which decay into radioiodines may escape from the primary coolant and decay later. Daughter radioiodines are formed by two distinct paths. I-129 and I-131 through I-138 are formed by the decay of the corresponding isotope of tellurium; I-113 through I-123 and I-125 are formed by the decay of the corresponding xenon isotope. However, these isotopes of xenon are not produced in any significant quantity during the fission process. I-127 is stable while I-126, I-128 and I-130 cannot be formed by the decay of other radionuclides.

If tellurium escapes from the primary coolant, it can decay inside or outside the plant. To estimate the release of radioiodines in the former case, we observe that Te-132 has the highest primary coolant concentration of any tellurium isotope. The ratio of daughter to parent specific activity for Te-132 to I-132 is also higher than any other tellurium-iodine pair. Consequently, we would expect that more I-132 activity would be released by this mechanism than any other radioiodine. Of 10 PWR plants reporting I-132 releases for 1979, the highest release, 9.7×10^{-3} Ci, was reported by Point Beach. Assuming for the purpose of this discussion that all of this activity was produced by the mechanism described above, we can conservatively assert that no more than .01 ci/yr of I-132 will be released at SHNPP by this mechanism. From the primary coolant concentration of the other tellurium isotopes, we can similarly

calculate the additional activities of other radioiodines released via this mechanism.

Isotope	Max Release (Ci/yr)
I-131	4.3×10^{-6}
I-132	1.0×10^{-2}
I-133	1.3×10^{-6}
I-134	1.4×10^{-8}

Te-135 through Te-138 are not significant primary or secondary fission products and are not found in the core or the primary coolant in any measurable quantities.

20. Another pathway for radioiodine release suggested or implied by Intervenor is from the decay of tellurium radioisotopes after their release from the plant. Only 3 PWR plants in the U.S. reported airborne releases of Te-132 for 1979. No releases of other tellurium radioisotopes were reported. The highest release, 2.39×10^{-5} Ci/yr of Te-132, was reported for Calvert Cliffs. Assuming this release rate for SHNPP, and estimating releases of other tellurium radioisotopes by the ratio of their concentration in the primary coolant, the additional effective releases of radioiodines are listed below.

I-129 - 7.8×10^{-15} Ci/yr

I-131 - 3.4×10^{-7} Ci/yr

I-132 - 8.2×10^{-4} Ci/yr

I-134 - 1.1×10^{-7} Ci/yr

21. The decay of Te yields the highest activity for I-132. The maximum calculated activity of I-132 from the decay of telluriurn 1.0×10^{-2} Ci/yr is equivalent to 1×10^{-4} Ci/yr of I-131. This is insignificant when compared to the SHNPP radioiodine Source Term of 4.8×10^{-2} Ci/yr. In addition, the rapid decay of I-132 in the ingestion pathway will make its contribution to the offsite dose even less. The total released radioiodine activities due to Te decay are thus seen to be completely insignificant compared to the SHNPP Source Term.

B. The Filter Factor

22. Radioiodine plant release estimates are adjusted to reflect the ability of air cleaning system filters to trap iodines prior to release.5/ A detailed description of the air cleaning system is provided in Attachment 6. Intervenor has made a number of allegations with regard to the ability of the filters to perform their intended function to trap airborne

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5/ Applicants' Appendix I compliance calculation uses a filter efficiency factor 90%. This value is in accordance with the guidance contained in NUREG-0017.

iodine gas and iodine particulates. In this section of our Affidavit we address certain of these allegations.

23. The actual in-place usage time of the High Efficiency Particulate Air (HEPA) filters will be much less than their life expectancy under normal environmental operating conditons, and the charcoal adsorber sections of the filter system employs gasket-less construction features. In addition, a quality assurance program applicable to safety-related equipment was adopted for this system.

24. The SHNPP air filter systems were designed and fabricated to the best industry standards and guidelines available at the time. Ebasco specifications for these units contains all the codes and standards used in the design, fabrication and testing of the units. Included in these standards and guidelines are (i) ANSI/ASME N509-1976 "Nuclear Power Plant Air Cleaning Units and Components", (ii) ANSI/ASEM N510-1975 "Testing of Nuclear Air Cleaning Systems" and (iii) ERDA 76-21 "Nuclear Air Cleaning Handbook" which are specifically recognized by Regulatory Guideline 1.140. In many ways, the systems exceed the minimum requirements of the standards. The systems are designed and fabricated to ensure the highest possible filtration efficiency. Typically the exhaust filter systems used during normal operation of the plant consist of filter casings, prefilters, HEPA filters, charcoal absorbers, fans and instrumentation. In cases where moisture entrainment

and high relative humidity (over 70%) are anticipated, demisters and electric heating coils are used to reduce the relative humidity below 70%. The materials and construction techniques used to manufacture Applicants' filter systems are listed in Attachment 6.

25. In his response to Applicants' Interrogatory 29-20(c), Intervenor Eddleman states that the NRC calculates releases via airborne effluents which are 1.5 times greater than CP&L's. In view of the fact that Mr. Eddleman implies that NRC supports his position that Applicants have underestimated their releases, the following offers 1) an explanation of why a difference exists between the Staff's and Applicants' value, and 2) the status of this difference. At the time of preparation of the SHNPP DES, the NRC's review of one of the radioactive waste handling systems was based on incomplete information. Based on their interpretation of that system operation, only partial credit was allowed for removal of radioiodines by that system. Consequently, the NRC's calculation of that system's contribution to the SHNPP gaseous Source Term was higher than the Applicants'. The NRC Staff has since completed a review of that radioactive waste treatment system and has revised its Source Term calculation. Discussions held with the NRC Staff revealed that their final Source Term estimate will be virtually the same as the Applicants'. This revised Source Term will be published in the NRC Final Environmental Statement on SHNPP.

26. In response to several of the NRC Staff's and Applicants' interrogatories pertaining to Contention 29/30, Intervenor states that effluent filtration and monitoring systems will not perform to the levels assumed by the Applicants and the NRC because of degradation resulting from exposure to radiation, heat and other environmental factors. As support for this allegation, Intervenor refers to various NUREG reports and publications by Robert L. Clough and Dennis T. Gillen. Our review of this material reveals that is not pertinent to the allegation and, if anything, provides evidence that the effluent filtration and monitoring systems will be able to perform as designed in their operating environment.

27. The reports and publications referred to by Intervenor were performed by Sandia Laboratories under contract to the NRC. The purpose of these studies was to evaluate the degradation of the insulation of safety related cabling in containment which is exposed to high doses of radiation. The objective was to determine whether improvements are needed in NRC guidance regarding acceptable methods to test the environmental qualifications of safety related equipment. These studies are unrelated to the filtration and monitoring systems associated with Appendix I compliance for the following reasons:

(1) The filters and monitors are not located within containment. As a result they will be exposed to doses which are at least 1,000 times smaller than the doses tested in these studies and are well below the levels where radiation damage was observed in these and other studies.

(2) The filters and monitors are not safety related equipment and if their performance degrades for any reason it will be determined during routine inspection and testing and corrective action taken.

28. Intervenor also contends that these studies show that as the dose rate goes down, the damage increases. As a result, Intervenor concludes that even if the doses experienced by the filters and monitors are low compared to the doses in Sandia studies the potential for serious degradation of performance still exists. This conclusion is incorrect because, while the Sandia studies show that degradation of equipment increases as dose rate goes down, this is true only over a specified dose range. This dose range is well in excess of the dose range to which the filters and monitors will be exposed in the operating environment. The Sandia studies also show that as the total dose goes down the degradation decreases. The lowest doses at which degradation was observed was in the Megarad range while the doses to the filters and monitors will be at least 1,000 times smaller and well below the level at which degradation was observed at any dose rate.

C. The Aquatic Dispersion Calculation6/

6/ The atmospheric dispersion calculation is discussed in the "Affidavit of Brian D. McFeathers In Support of Applicants' Motion for Summary Disposition of Intervenor Wells Eddleman's Contention 80," dated August 31, 1983.

29. To determine the exposure to individuals from radioiodines released to the Main Reservoir the steady state completely mixed dispersion model, equation 43, of NRC Regulatory Guide 1.113 was used. This model assumes that radioiodines are removed from the reservoir only by radiological decay and by discharge to the Cape Fear River.

30. Evaporation is not directly incorporated into this model since evaporation would not be expected to remove radioiodines from the water. However, evaporation has been incorporated in the determination of the annual average Main Reservoir discharge to the Cape Fear River, as discussed in ER Section 2.4.2.

31. Sedimentation, the chemical interaction of iodine with other materials, and stratification within the reservoir have also not been incorporated into this model.7/ The chemical interaction of the radioiodines with other materials would simply decrease the radioiodine concentration level within the Harris Reservoir due to the settling out, or sedimentation, or radioiodines chemically combined with other materials in the water. We can determine no effects of chemical interaction of radioiodines or stratification within the Harris Reservoir which would lead to underestimation of radioiodine

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7/ Applicants' dose calculation does, however, consider the exposure resulting from shore-line deposits and sedimentation. See Reg. Guide 1.109, equation A-5.

concentrations in the Harris Reservoir as alleged by Intervenor Eddleman.

32. In a similar fashion, stratification, if it occurred within the reservoir, would tend to increase the rate of dispersion within the Harris Reservoir. The effluent discharge point into the Harris Reservoir is a subterranean pipe below the surface of the reservoir. See ER § 5.2.1.2.1. Stratification, if it occurred, would result in thermal barriers separating the Harris Reservoir into different horizontal layers of water distinguished by their temperatures. The effect of a subsurface warm water discharge point into a stratified reservoir would result in disrupting the integrity of the temperature layers and an increase in mixing activity. This would tend to reduce the localized concentration of radioiodines and have no effect on the average concentration in the reservoir.

33. Consequently, considering the effects of chemical reactions between released radioiodines and other compounds in the Harris Reservoir would reduce the radioiodine concentrations for the aquatic medium and render Applicants' aquatic dispersion calculations less conservative than they are at the present time. Consideration of stratification would have no effect on the calculated doses.8/

8/ An evaluation of the limited data presently available to Applicants from their Harris Reservoir monitoring program indi-

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34. As stated in Regulatory Guide 1.113, the steady state completely mixed model is most applicable for long lived radionuclides. For short-lived radionuclides, such as I-131, this model is accurate only in predicting the total amount of activity in the Main Reservoir and not in the distribution of this activity. In order to show the range of concentrations in the Main Reservoir, both the average concentration and the concentration in the cooling tower discharge are provided in ER Table 5.2.2-3.

35. To determine the exposure of individuals to radioiodines in the Main Reservoir, the concentration of radioiodine in fish is of primary concern. Using the average water radioiodine concentration to determine the fish radioiodine concentration implies simply that the fish consumed by the exposed individual spends as much time in water with a higher than average concentration as in water with a lower than average concentration. This assumption is consistent with the Appendix I philosophy of determining exposures based on realistic assumptions.

36. If sedimentation and chemical interaction were incorporated in the model, the calculated doses would be smaller. If stratification were included in the models, it would have no

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cates that no stratification occurs within the Harris lake, however, should such stratification occur it would only lead to greater localized dispersion.

effect on the calculated doses. Using a multicompartment dispersion model is not required for calculating exposure via the fish ingestion pathway because fish are expected to reflect the average radionuclide concentration in the reservoir.

D. The Dose Calculation

37. Calculation of the dose to the public from releases of radioiodine from SHNPP was performed under our supervision, in strict accordance with Regulatory Guide 1.109. We address below several allegations regarding these calculations.

38. The accumulation of radionuclides by plants via root absorption is one of the pathways of exposure which was modeled. In performing this calculation it was assumed that the radioactivity in the plant due to root uptake is proportional to the average radioactivity of the soil. The constant of proportionality, or uptake factor, is obtained from measurements made of the concentration of radionuclides and stable elements in plants and soils. The uptake factors used are listed in Regulatory Guide 1.109 and are based on a report by Dr. Y. Ng. A review of Dr. Ng's report and several of the references cited in that report revealed that natural settings such as agricultural soils, meadows, forests and gardens (not sterilized soils as alleged by Intervenor) were used to derive the proportionality constants. Plant concentration of the different elements were also derived from material collected from the natural environment (not from transplanted plants

given insufficient time to accumulate radionuclides as alleged by Intervenor).

39. Intervenor has relied on the so-called "Heidelberg Report" (NRC Translation 520) as a credible source of information representing an overview of the literature concerning transfer of radionuclides through environmental pathways. This reliance is misplaced. The Heidelberg Report purports to assess the environmental radiological impact of a proposed PWR to be built near Wyhl, West Germany. A review of the Heidelberg Report and much of the correspondence regarding the report, revealed several gross inaccuracies and the fact that the literature referenced by it did not represent an accurate overview of the data concerning model transfer factors. In addition, many U.S. and German scientists and government agencies have severely criticized the entire document to the extent that the report has been thoroughly discredited by the scientific community. These criticisms are summarized in Attachment 7.

40. Applicants calculated no dose for the lung, but the lung dose would not be critical. Based upon information developed between 1959 and 1979, the International Committee on Radiation Protection has estimated radioiodine inhalation dose conversion factors for all the important organs of the body including the lung. By comparing the organ conversion factors for radioiodine it can be seen that the dose to the thyroid per unit activity inhaled would be several orders of magnitude

greater than the dose to the lung. It is, therefore, the thyroid that is the critical organ as was reported in the SHNPP FSAR and ER.

41. The dose attributable to vegetable consumption by Applicants for the maximum individual is certainly not underestimated as alleged by Intervenor. It is probably an overestimate, because of the many inherent conservative assumptions and parameters built into the dose calculation. It should be noted that these pertain only to the dose estimation and do not take into consideration the conservatisms assumed in the Source Term calculation and atmospheric dispersion calculation.

42. The vegetable consumption dose calculation is based on the assumptions that (1) a child consumes 26 kg/yr, (2) a teenager consumes 42 kg/yr, and (3) an adult consumes 64 kg/yr of leafy vegetables. For adults this is more than 2-1/2 lbs. per week of leafy vegetables every week of the year. In addition, all the vegetables are assumed to be obtained from the backyard garden. The methods by which these values were obtained are referenced in Regulatory Guide 1.109.

43. After radionuclides are initially deposited on vegetation surfaces, environmental processes, in addition to radiological decay, will begin to remove the deposited material. The environmental removal processes considered are usually wind and water removal, growth dilution and herbivorous grazing. The half-life assumed for this removal was 14 days,

however, recent work indicates that for iodine vapor and particulates a half-life of 8 days is most likely (C.W. Miller and F.O. Hoffman, Oak Ridge National Laboratory, (submitted to Health Physics) "An Examination of the Environmental Half-time for Radionuclides Deposited in Vegetation"). The fact that no removal was assumed for food preparation and cooking further underscores the conservatisms of the dose estimation. It has been estimated that food preparation can remove at least 50% of the total activity deposited on the surface of leafy vegetables (Hermes, A Digital Computer Code for Estimating Regional Radiological Effects from the Nuclear Power Industry, Report HEDL-TME-71-168, Hanford Engineering Development Laboratory, 1971).

44. Intervenor has assumed, in error, that the 60 day period used by Applicants in their dose calculation relates to the period of human consumption of vegetables. The 60 day period used in the calculation is the period radioiodines are assumed to accumulate on plant surfaces, and not the time during which the vegetables are anticipated to be consumed. This 60 day period approximates the maturation period of many food crops. Since only the matured leaves of vegetables, including collard greens, are consumed, they are not exposed to radioiodine deposition for the entire 60 day growth period of the plant. The dose calculation is, therefore, an overestimate because it assumes a conservative accumulation of deposited activity.

45. An additional source of overestimation in Applicants' dose calculation is the manner in which maximum doses are aggregated in ER Table 5.2.4-2. The maximum individual exposures from airborne releases reported by Applicants in ER Table 5.2.4-2 is a summation of all dose pathways. It should be noted that the maximum location for meat and milk pathway is different than the maximum location for the crop pathway. Implicit, in the Table, is the conservative and realistically impossible assumption that the maximum individual receptor lives in both maximum exposure locations at once. Any real maximum individual would have a total dose lower than those contained in the Table values.

III. Verification of the Models Used By Applicants to Predict I-131 Source Terms

46. The methods used to assess compliance with Appendix I include a Source Term calculation which conforms to the guidance provided in NUREG-0017. With any predictive model, there is always some question whether the calculated results are reliable. In order to demonstrate the reliability of the standard methods, Attachment 3 presents the gaseous radioiodine source terms predicted for specific plants, and compares those source terms to the radioiodine releases actually measured at the plants after they became operational. The comparison reveals that the predictive methods are reliable and overestimate the actual releases by an approximate average factor of 5.

47. Attachment 4 comparesthe predicted I-131 source terms for liquid effluents to the values actually measured at operating pressurized reactor plants in 1979. The comparison shows that in almost every case the predicted values are considerably higher than the measured values and overestimate actual releases by an average factor of approximately 10.

IV. Conclusion

48. Based on our familiarity with Applicants' Source Term derivation and the matters contained within this Affidavit, our review of the pleadings and discovery on Eddleman Contention 29/30 and a review of NRC guidance and Regulatory requirements regarding radioiodine releases during normal operations, we can state that we know of no factual basis for Eddleman Contention 29/30 or the various allegations of Mr. Eddleman with regard to the matters contained within this Affidavit. We believe that no significant defects or deficiencies exist with regard to Applicants' Appendix I compliance demonstration, the data used, or the NRC regulatory guidelines, models and methodology applied by the Applicants in that demonstration. We believe that Applicants' Appendix I calculation is an adequate mathematical model of the radioiodine releases, aquatic dispersion and dose estimates during normal operations of the Shearon Harris Plant.

JOHN J. MAURO

Marie A. Petraitis

MARIE A. PETRAITIS
Notary Public, State of New York
No. 01PE4605003
Qualified in Kings County
Cert. Filed in New York County
Commission Expires March 30, 1985

10/3/83

GUY MARTIN, JR.

Subscribed and sworn to before me this 3rd day of October, 1983.

NOTARY PUBLIC

My Commission Expires

MARIE A. PETRAITIS
Notary Public, State of New York
No. 01PE4605003
Qualified in Kings County
Cert. Filed in New York County
Commission Expires March 30, 1985

ATTACHMENT 1

JOHN JOSEPH MAURO
Certified Health Physicist

SUMMARY OF EXPERIENCE (Since 1970)

Total Experience - 12 years total experience. Six years experience in calculating projected exposures of radionuclides in the environment and evaluating the biological significance of these exposures. Six years experience in managing radiological and hazardous chemical consulting projects.

Professional Affiliations - Health Physics Society
ANSI Committee N18 - Guide to Standard Format and Content of Emergency Plans for Nuclear Power Generating Facilities

Education - PhD, New York University, 1973 - Biology, Radiological Health
MS, New York University, 1970 - Biology, Radiological Health
BS, Long Island University, 1967 - Biology, Bacteriology

Awards - Alvin Gruder Memorial Award
Founders Day Award

REPRESENTATIVE ENVIROSPHERE PROJECT EXPERIENCE (Since 1973)

1973 to Present
Radiological Assessment Scientist/Director Radiological Assessment and Health Physics Department

Responsible for evaluating the radiological impact of nuclear power plant operation for Ebasco's client utilities. Prepared the requisite sections of the Environmental Reports and Safety Analysis Reports and defended these analyses before the NRC, the Advisory Committee on Reactor Safeguards, State Site Certification Boards and the Atomic Safety and Licensing Boards. Provided these services for 10 nuclear power plants.

Responsible for the calculation of projected radionuclide inventories and liquid and gaseous source terms associated with various aspects of normal plant operation and hypothetical accidents. Experienced in the calculation of the projected environmental transport, distribution and concentration of radionuclides released in the liquid and gaseous effluent of nuclear power facilities; the calculation of the projected exposures to man and organisms other than man due to radionuclides in the environment; and the evaluation biological effects of exposure to radiation.

JOHN JOSEPH MAURO (Continued)

Experienced in the design of environmental radiological surveillance programs for nuclear power facilities. Has installed and calibrated radiation detection instrumentation at low level radiological laboratories and participated in the training of laboratory personnel.

Has managed the preparation of several emergency response plans and implementation procedures for commercial nuclear power facilities and for state agencies responsible for emergency response planning.

Has provided radiological and emergency response training to health physicists, nuclear engineers and members of state and local agencies responsible for emergency planning.

Has managed the preparation of the environmental/radiological technical specifications and offsite dose calculation manuals for several nuclear power facilities.

Has managed numerous consulting projects in the area of decommissioning, environmental monitoring, environmental dosimetry in-plant health physics for the nuclear fuel cycle.

Has performed several toxic chemical studies in support of nuclear power plant control room design and in the assessment of the radiological and chemical toxicity of low level radioactive wastes.

PRIOR EXPERIENCE (2 years)

Private part-time consulting work.

Worked on the radiological effects sections of a nuclear power plant environmental impact report. Drafted answers to interrogatories from an AEC licensing hearing.

New York University Institute of Environmental Medicine
Assistant Research Scientist (2 years)

Performed research work on the life history of white perch in the Hudson River. Aided in the cataloging and enumeration of invertebrates in the Hudson River. Participated in a project to develop techniques to determine and ability of various micro-organisms to organify inorganic mercury.

Publications and Presentations

Mauro, J J and M E Wrenn 1972. A review of radiocesium in aquatic biota. Presented at the Health Physics Society Annual Meeting, Las Vegas, Nevada, June 12-16.

JOHN JOSEPH MAURO (Continued)

Mauro, J J and M E Wrenn 1973. Reasons for the absence of a trophic level effect for radiocesium in the Hudson River Estuary. Presented at the IRPA meeting held in Washington, D.C. in October. Published in the proceedings of that meeting.

Mauro, J J, and J Porrovecchio. Numerical criteria for in-plant as low as is reasonably achievable proceedings of the 9th Mid-Year Topical Symposium of the Health Physics Society.

Mauro, J J, D Michlewicz and A Letizia 1977. Evaluation of environmental dosimetry models for applicability to possible radioactive waste repository discharges, Y/OW1/SUB-77/45705, September.

Mauro, J J 1978. Comparison of gaseous effluent standards for nuclear and fossil fuel power production facilities. Proceedings of the December 1978 Annual Meeting of the American Nuclear Society.

Mauro, J J, J Thomas, J Ryniker and R Fellman 1979. Airborne uranium, its concentration and toxicity in uranium enrichment facilities, K/PO/SUB-79/31057/1, February.

Mauro, J J, K E Lind, J D Levine, L Yemin, H J Howe, Jr and C W Pierce 1979. Safety related research required to support future fusion research reactors. Presented at the Annual Meeting of the American Nuclear Society-San Francisco, November.

Mauro, J J and E P O'Donnell 1979. A cost-benefit comparison of nuclear and nonnuclear health and safety protective measures and regulations. Nuclear Safety, Vol. 20 No. 5, September-October.

Mauro, J J 1980. A real time computer program for offsite radiological impact assessment. Presented at the 1980 Annual Meeting of the American Nuclear Society. TANSAO 34 1-899.

Mauro, J J, R Bhatia and G Martin 1980. Effects of containment purge on the consequences of a loss of coolant accident. Presented at the 1980 Annual Meeting of the American Nuclear Society. TANSAO 34 1-899.

Mauro, J J and S Marschke 1980. Radiocesium transport into reservoir bottom sediments - a licensing approach. Presented at the 1980 Annual Meeting of the ANS. TANSAO 34 1-899.

Mauro, J J and D Michlewicz 1981 deployment concepts for Real Time Environmental Dosimetry Systems. Presented at the 1981 Annual Meeting if the Health Physics Society.

Mauro, J J and E P O'Donnell 1982. The role of the Architect/Engineer in the Emergency Planning Process. Presented at the Annual Meeting of the American Nuclear Society. June 6-10, 1982.

JOHN JOSEPH MAURO (Continued)

Mauro, J J and W R Rish 1982. Dealing with Uncertainties in Examining Safety Goals for Nuclear Power Plants. In NUREG/CP-0027. Proceedings of the International Meeting on Thermal Reactor Safety.

Mauro, J J, S Schaffer, J Ryniker, and J Roetzer. Survey of Chemical and Radiological Indices Evaluating Toxicity. National Low-Level Radioactive Waste Management Program. DOE/LLW-17T. March, 1983.

ATTACHMENT 2

GUY MARTIN, JR

Manager
Radiological Assessment

SUMMARY OF EXPERIENCE (Since 1965)

Total Experience - Ten years participation in preparation of engineering safety analyses of radiological and toxic chemical protection. Six years in cost analysis for insurance premium determination.

Professional Affiliations - American Society of Mechanical Engineers
Health Physics Society
American Nuclear Society
Intern Engineer in New York State

Education - M.S,. Nuclear Engineering, Polytechnic Institute of New York, 1976
B.M.E., City College of the City of New York, 1974

REPRESENTATIVE EBASCO PROJECT EXPERIENCE (Since 1973)

Manager - Radiological Assessment Department

Areas of complete responsibility include the preparation of Safety Analysis and Environmental Reports (part of the Construction Permit and Operating License application for nuclear power plants) sections dealing with impact analyses of toxic chemical and radiological releases. Such analyses are performed for both routine plant operation and under accident conditions. In this regard, conduct reviews of radwaste handling systems, air handling and cleanup systems and habitability systems. Estimate radionuclide releases from plant effluents and calculate radiological doses to biota and man. Calculate the in-plant dose rates to equipment and personnel from airborne radionuclide exposure and perform "as low as is reasonably achievable" (ALARA) reviews of air cleanup systems. Perform safety reviews of engineered safety systems, their specifications and operation from a radiation protection viewpoint and provide design recommendations based on assessed radiological doses and established nuclear safety criteria. Perform the analysis of transport of toxic chemicals postulated to be released accidentally and calculate their concentration in critical locations of a power plant and, provide technical feedback on required protection level. Assist responsible disciplines in determining toxic chemical detector specifications based on worker and equipment protection criteria.

Additional area of responsibility include the preparation of radiological environmental surveillance programs. In this regard, prepare detailed surveillance program description based upon site specific critical pathways of exposure, specify samples, frequency and types of analysis to be performed, prepare vendor and/or laboratory bid request, contractor selection, assist in monitoring equipment selection and review periodic surveillance reports submitted by contractors.

GUY MARTIN, JR (Cont'd)

Participate in the defense of the licensing documents. Activities involve preparation of responses to intervenor interrogatories, testimonies and furnishing technical support to Ebasco's clients in licensing presentations (e.g., ACRS, ASLB etc.) and, safety and environmental hearings.

Experience to date spans over the following Ebasco projects:

Client	Project	Reactor Type
Houston Lighting & Power Company	Allens Creek 1	BWR
Comision Federal de Electricidad	Laguna Verde 1 & 2	BWR
Louisiana Power & Light Company	Waterford 3	PWR
Carolina Power & Light Company	Shearon Harris 1-4	PWR
Washington Public Power Supply System	WNP 3 & 5	PWR
Florida Power & Light Company	St Lucie 1 & 2	PWR

In addition to the above responsibilities on Ebasco's major nuclear projects, participated in the following projects on a supervisory capacity:

Decommissioning Requirements for Nuclear Waste Repository Licensing

A study was conducted whose principal component was the establishment of comprehensive data base covering D&D as it applies to a high-level waste repository located in specific geological formations. The report prepared by Envirosphere presents a compilation of the various regulations and existing industry guidance and experience pertinent to D&D. In addition, a discussion of the various D&D alternatives which may apply to such waste repository facilities along with D&D strategies and scenarios were presented.

Remedial Action Project - Kellex Laboratory Site

Supervised (on part-time basis) the health physics activities related to the decontamination work. Such activities included personnel and environmental monitoring, field radiation surveys and quality control.

GUY MARTIN, JR (Cont'd)

PRIOR EXPERIENCE (6 years)

Equitable Life Assurance Society of the US
Cost Analyst

Work involved calculating and analyzing cost of various activities performed throughout the company; assisting departmental managers in their budget preparation work. Made statistical studies for determination of activity costs and providing company's actuaries support information for premium determination.

Publications

Martin, G and J Thomas 1978. Meeting the dose requirements of 10CFR100 for site suitability and general design criteria 19 for control room habitability: a parametric approach. Transactions of American Nuclear Society 24th Annual Meeting. Vol. 28.

Martin, G, D Michlewicz and J Thomas 1978. Fission 2120: a program for assessing the need for engineered safety feature grade air cleaning systems in post - accident environment. Proceedings of 15th DOE Nuclear Air Cleaning Conference.

Letizia, A P, G Martin and J F Silvey 1979. - Implications for nuclear facilities of changes being initiated in the NRC standard atmospheric diffusion model. Proceeding of the 41st Annual Meeting of the American Power Conference.

Bhatia, R K, Mauro, J, Martin, G. Effects of Containment Purge on the Consequences of a Loss-of-Coolant Accident. Transactions of American Nuclear Society 1980 Annual Meeting.

ATTACHMENT 3

AIRBORNE RADIOIODINE SOURCE TERMS

UNIT	PREDICTED(1,3) (Ci/Yr - unit)	MEASURED (Ci/Yr)(2) Average	Range
Arkansas 1	.048	.14	.003-.74
Arkansas 2	.17	.0047	.0047
Beaver Valley	.014	.021	.0001-.072
Calvert Cliffs (2 units)	.25	.27	.035-1.0
Crystal River	.12	.0071	.0025-.019
Davis-Besse	.12	.0021	.00026-.0057
D.C. Cook (2 units)	.10	.028	.005-.055
Ft. Calhoun	.065	.011	.0016-.02
Haddam Neck	.04	.019	.0017-.05
H.B. Robinson	-	.063	.0004-.3
Indian Point 1 & 2	.36	.22	.005-.81
Indian Point 3	-	.0084	.0039-.013
J.M. Farley	.049	.032	.022-.041
Kewaunee	.081	.12	.00062-.66
Maine Yankee	-	.14	.0021-.94
Millstone 2	.105	.0059	.0-.013
North Anna 1	.095	.045	.032-.057
Oconee (3 units)	.80	.062	.0033-.18
Palisades	.79	.1	.01-.38
Point Beach (2 units)	-	.049	.0025-.28
Prairie Island	.137	.0093	.0009-.021
Rancho Seco	-	.013	.005-.032
R.F. Ginna	.11	.039	.01-.17
Salem	.21	.016	.0-.04
San Onofre	-	.17	.00014-1.6
St. Lucie 1	1.0	.22	.01-.52
Surry	2.1	.097	.0076-.35
TMI 1	-	.035	.01-.14
Trojan	.24	.028	.01-.051
Turkey Point (2 units)	.80	.44	.03-1.8
Yankee Rowe	-	.077	.0-.53
Zion (2 units)	.20	.033	.005-.07
Average (Ci/Yr-unit)	.34 ci/yr-unit	.065 ci/yr-unit	

FOOTNOTES

(1) The predicted values were obtained from the FES for each plant and are based on calculations performed by the NRC using industry wide standard methods. The values are for I-131 except where indicated.

(2) The average and range are inclusive over the years of operation from 1970 to 1979. The values are a slight overestimate because they include I-131 and particulates with half lives greater than 8 days.

(3) Value not available is denoted by "-".

ATTACHMENT 4

I-131 RELEASES IN LIQUID EFFLUENTS IN 1979

PLANT	PREDICTED(1,3) (Ci/Yr-Unit)	MEASURED(2) (Ci/Yr)
Arkansas 1	9.2	.28
Arkansas 2	.26	.24
Beaver Valley 1	.34	.0008
Calvert Cliffs 1 & 2 (2 units)	.27	.65
D.C. Cook 1 & 2 (2 units)	.47	.012
Crystal River 3	2.0	.06
Davis-Besse 1	2.37	.0035
J.M. Farley 1	.48	.0013
Ft. Calhoun 1	1.8	.019
R.E. Ginna 1	.27	.0093
Haddam Neck 1	.36	.067
Indian Point 1 & 2 (2 units)	2.06	.079
Indian Point 3	-	.059
Kewaunee	.51	.00059
Maine Yankee 1	-	.41
Millstone 2	.9	.12
North Anna 1	1.2	.16
Oconee 1, 2 & 3 (2 units)(?)	.2	.14
Palisades 1	-	.00038
Point Beach 1 & 2 (2 units)	-	.088
Prarie Is. 1 & 2 (2 units)	3.8	.00076
Rancho Seco 1	0	.0
H.B. Robinson 2	-	.0037
Salem 1	1.43	.019
San Onofre 1	-	.025
St. Lucie 1	.17	.048
Surry 1 & 2 (2 units)	12.15	.064
TMI 1	-	.14
Trojan 1	.21	.012
Turkey Pt. 3 & 4 (2 units)	10.2	.020
Yankee Rowe 1	-	.0041
Zion 1 & 2 (2 units)	.81	.011
Average (Ci/Yr-unit)	2.1	.065

(1) From the Final Environmental Statements

(2) From NUREG/CR-2227

(3) Value not available is denoted by "-".

ATTACHMENT 5

IDENTIFICATION OF RADIOIODINE ISOTOPES
SIGNIFICANTLY CONTRIBUTING
TO DOSE ESTIMATION

There are several hundred known nuclides, most of which are radioactive and only some of which are produced in a nuclear reactor. The quantity of fission products in the reactor is dependent upon the decay rate of the radionuclide, measured by its half-life, and its fission yield. Fission yield can be interpreted as the probability of producing a given nuclide as a result of the fission process. Fission yield is frequently displayed as a curve based on a large number of experiments. The fision yield curve can be found in most texts on nuclear physics or engineering, as in John Lamarsh, Introduction to Nuclear Engineering at 68, (Addison-Wesley 1975). This text displays fissions as a function of the mass number of the nuclide (the sum of the number of neutrons and the number of protrons in the nucleus). The fission yield curve has two peaks. For thermal neutron fission of uranium - 235 ("U-235"),1/ they occur near mass numbers 93 and 135. The curve drops off sharply below 93 and above 135 and drops to a low point near mass number 120. The fission yield of a

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1/ Isopotes of an element are denoted by the periodic system (U for uranium, I for iodine Te for teluium) followed by the atomic weight (the combined total of neutrons and protons), hence: I-131, U-235, U-238, etc.

specific nuclide such as Iodine-131 ("I-131"), is only a portion of the total yield of all nuclides of mass number 131.

Yield is also dependent on the neutron to proton ratio. Neutron-proton ratios near that of uranium-235, i.e. 1.55, are favored as primary fission.

It is instructive to consider this curve in investigating the question of whether Applicants have underestimated radioiodine releases. The Source Term calculated includes all radionuclides which make a significant contribution to the calculated doses. It includes both primary fission products and their decay products. An attempt is not made to identify every radionuclide which might occur as a result of the fission process. Instead, conservatisms in dose calculations are such that those radionuclides which are significant contributors to dosage are predicted to produce releases and resulting doses as large as, or larger than, those which have actually been measured.

In the Environmental Report and the DES, the analyses are limited to I-131 and I-133. The totality of theoretically possible radioiodines include I-115 to I-140. The following shows that of these 26 nuclides, only I-131 and I-133 are significant and require explicit consideration. For the purpose of this discussion, it is convenient to first consider I-115 through I-128 as a group and I-136 through I-140 as another. The other isotopes of iodine are considered individually.

Consideration of the fission yield curve indicates that nuclides with mass numbers between 98 to 128 occur in less than .001 fissions. This is as compared to nuclides with mass numbers of 131 which have a fission yield of .08. Accordingly, based on mass numbers alone I-115 to I-128 are produced at a much lower rate than I-131. When this low production rate is considered along with their relatively short half lives, it becomes apparent that the core inventories of I-115 to I-128 are extremely small compared to that of I-131 and therefore do not require explicit consideration.

The iodine isotopes I-136 through I-140 are produced in significant quantities as a result of their favorable neutron-proton ratio and position on the fission product yield curve. However, due to the fact that they have half-lives ranging from 1.5 sec to 85 sec, they will not significantly affect the core inventory of radioiodines. With half-lives this short, even if released, by the time they reach the environment, the vast majority of these radioisotopes will have decayed to another nuclide; specifically an isotope of xenon with the same mass number.

With the elimination of these isotopes, I-129 through I-135 remain. I-127 is stable and does not require consideration. I-129 was not included in the Appendix I compliance calculation because of its low core inventory, 1.4 curies as opposed to 8.0 curies of I-131. Assuming that this relationship is maintained in the primary coolant and throughout the

plant, then the I-129 release will be 8.1 curies per year. This is why I-129 was not explicity addressed in estimating radioiodine release. Of the iodines from I-130 to I-135 only I-131 and I-133 received explicit consideration because of their relatively greater contribution to offsite dose.

The maximum potential offsite impact of each radioiodine, relative to I-131, can be determined by:

$$RI\ (i) = \frac{CI\ (i) * DCF\ (i)}{CI\ (I\text{-}131) * DCF\ (I\text{-}131)}$$

RI (i) = relative impact of radioiodine i

CI (i) = core inventory of radioiodine (FSAR, Table 15.0.9-1)

DCF (i) = dose conversion factor of radioiodine I (NRC Regulatory Guide 1.109) retention in the body is included in this term.

Evaluating this equation produces the following:

Isotope	RI
I-130	7.8×10^{-4} 2/
I-132	1.4×10^{-2}
I-133	3.8×10^{-1}
I-134	5.6×10^{-3}
I-135	7.0×10^{-2}

2/ For I-130 the relative impact was calculated using the primary coolant concentrations from FSAR, Table 11.1.1-5 rather than the core inventory.

ATTACHMENT 6

SHNPP FILTER SYSTEM CHARACTERISTICS

I. FILTER MATERIALS

The following materials were used in the fabrication of Applicants' filter units:

COMPONENT	MATERIAL
Casing except floor (above 10,000 Cubic feet per minute - "CFM")	Carbon Steel American Standard for Testing Materials ("ASTM")
Casing floor (above 10,000 CFM)	ASTM-A240, #304 Stainless Steel ("SS")
Casing (10,000 CFM or less and units in Containment Building	ASTM-A240, #304SS
Mounting Frames for Filters, Demisters & Electric Heating Coils	ASTM-A276, #304SS
Charcoal Adsorber	304SS
HEPA Filter Frame	Cadmium Plated Steel
HEPA Filter Separator	Aluminum Type 1145-H19
HEPA Filter Face Guard	Galvanized Steel Screen
Prefilter Frame	Galvanized Steel
Demisters	#304SS

GASKETS

COMPONENT USED FOR	GASKET MATERIAL
HEPA Filters	Neoprene
Prefilters	Neoprene
Access Doors & Electric Heating Coils	Cohrlastic R-10480 Gr. Medium Silicone Rubber

SEALS

COMPONENT USED FOR	SEAL MATERIAL
HEPA Filters	Self Extinguishing Rubber Base or Urethane
Pre-Filters	Fire Retardant Foam

Corrosion problem was adequately addressed by using appropriate materials and coatings where necessary to prolong the life of the units.

II. FILTER UNIT CONSTRUCTION

A. Housing

All air cleaning units were built as prefabricated, skid-mounted assemblies, and where required, units were fabricated as split units to facilitate handling and shipping. The housing assemblies consist of 2/16"-thick skin to reduce the amount of external stiffeners which are required for accommodating

pressures and seismic loads. Where necessary, stiffeners are provided on the exterior of the housings so as to ensure a smooth interior for ease of future decontamination. Housings were prefabricated, complete with internal filter racks, absorbers, lights, access doors, drains and instruments.

The internal filter rack assemblies were fabricated as separate assemblies and installed into housings by seal welding to the housing skin. Similarly, the charcoal absorbers were fabricated as separate assemblies and mounted and sealed within the housing by seal wedling to the housing skin.

Access Doors are provided on either side of the housing assemblies to allow adequate space for maintenance personnel to enter the housing wearing protective clothing, remove individual spent filter elements and replace with new filters. In sections which require no servicing access space, adequate internal space is provided to allow maintenance personnel to perform necessary in-place testing and inspection activities.

B. Moisture separator/Prefilter/HEPA Rack Construction

The filter element racks form a grid framework providing support for individual elements, and a smooth surface for gasket sealing. Clamping mechanisms provide for gasket compression and filter release for changeout. The moisture separator and HEPA racks consist of a grid formed of welded stainless steel structural channel. The prefilter racks consist of a peripheral welded channel frame with vertical channel stiffeners

between which are mounted modular prefilter clamping frames. With the use of these types of rack construction, a permanently smooth, rigid, gasket-sealing surface is ensured.

Each rack is seal-welded to the outer casing to completely eliminate air bypass around filter elements. Each HEPA filter element is individually clamped to the rack with the use of quick-acting clamping method. Filters may be removed without disturbing adjacent filters. Elements are supported and clamped in such a way that a plastic filter disposal bag may be slipped over the filter prior to performing filter removal operations. No rivets, screws or caulking-type materials were used, and all-welded construction is employed throughout. No leaks through the housing or around the filter sealing faces are permitted.

C. Charcoal Adsorber

The adsorber is designed especially for purposes of attaining extremely high removal efficiencies of radioactive iodines and iodine compounds from contaminated air passing through the filtration systems. Completely shop-fabricated of stainless steel, the design employs adsorbent beds which are constructed by welding to eliminate the dependence on mechanical gaskets or elastomeric seals. The adsorber consists of a series of vertically oriented parallel adsorbent beds mounted within the air stream. Air enters through vertical slots in the upstream face of the adsorber, passes left or right through

the adsorbent bed, and out through adjacent downstream slots. The adsorber design combines the use of extended surface, low-air velocity parallel beds with a configuration that assures uniform adsorbent packing densities and ease of adsorbent filling and emptying. Individual adsorbent beds are sized to permit a face velocity of 40 feet per minute and an air-to-adsorbent residence time of 0.25 seconds per each 2 inches of adsorbent. The air slots which permit entry and discharge of air are normally 2 inches in width. Typically, the normal exhaust filter systems have 4" beds.

The area containing the adsorbent is completely open and without baffling or internal structural support. Structural members are installed only within the air slots between individaul beds. Chance of bypass due to uneven adsorbent packing density or solid-to-adsorbent interfaces is virtually eliminated. Further, bulk filling is possible without the interference of internal bed separators.

All materials used in the fabrication of FILTRAD adsorbers are stainless steel, Type 304. Each bed consists of 26 gauge perforated material with bed sides made of solid formed channel. Proper bed spacing is maintained by forming bed supports into the perforated bed screens which project into the air slots. No support or spacers are mounted in the bed volumes containing the adsorbent.

It should be noted that the gaskets and sealants for the filters are also replaced with the filters at the time of filter replacements.

III. INSPECTION

The units were inspected during various stages of fabrication by the Quality Assurance inspectors of both the Vendor and Ebasco to check compliance with the specifications and all Ebasco approved vendor design drawings and procedures. The units were inspected at the fabrication shop before shipment to the job site. Then they were packaged and shipped to the job site in accordance with approved procedures. Prefilters, HEPA Filters, Demisters, Electric Heating Coils and Fans were shipped in separate packages.

All the units and packages were inspected at the job site by CP&L at the time of delivery and damages, if any, were reported to vendor immediately in accordance with approved procedures.

IV. TESTING

A. Shop Tests

DOP and R-11 injection and test ports were provided both upstream and downstream of each filter rack.

After the units are installed complete with all the filters the following tests are performed:

B. Air-Aerosol Mixing Uniformity Test

This test is made only upon completion of initial installation, modification or major repair of the system. The purpose is to verify that tracer (DOP or refrigerant gas) injection and sample ports are located so as to provide proper

mixing of the tracer in the air approaching the component stage (HEPA filter bank or adsorber stage) to be tested. This test is performed before all the other in-place leak tests outlined below.

C. Leak Test For Hepa Filter Banks

This test is used during acceptance testing of the air cleaning system, after any filter replacement, or after any maintenance activity in the filter housing to verify (1) that the filters have not been damaged, (2) that they have been installed properly, (3) that there are no leaks in the is determined from the ratio of downstream to upstream concentration at time zero. Any leaks detected are repaired and the test repeated.

Again the method and procedure are not described in detail since it is not warranted here other than to mention of their existence and usage.

D. Laboratory Testing of Adsorbent

This test is used to establish the condition of the adsorbent in the adsorber system. A test is made of a sample of adsorbent before it is loaded into the adsorber beds to establish the initial point for comparison of future surveillance test results and in addition to verify the suitability of the adsorbent actually used.

During surveillance tests, test canisters containing the charcoal are removed from the adsorbers and sent for laboratory

testing and the results at that point are compared with the initial point and with each other, and curves showing the variations of adsorbent condition with time are prepared. Activated charcoal for the Air Cleaning Units is provided by CP&L and the initial condition of the charcoal shall be obtained from the charcoal vendor. The arrangements for subsequent surveillance tests of the samples shall be made with the vendor. The frequency of tests shall be established by CP&L with the help of the charcoal vendor.

The following table shows the tests and minimum test frequencies for the filters:

E. Test and Minimum Test Frequency

Test	Recommended Test Frequency
Duct and Housing Leak Test	Acceptance (1)
Mounting Frame Pressure Leak Test	Acceptance (1)
Airflow Capacity, Distribution, and Residence Time Tests	Acceptance (1)
Air-Aerosol Mixing Uniformity Test	Acceptance (1)
In-Place Leak Test, HEPA Filters	Acceptance after each filter change and at least annually (1,2)
In-Place Leak Test, Adsorbers	Acceptance, after each adsorber change, and at least annually (1,2)
Laboratory Testing of Adsorbent	Acceptance, before each adsorber change, and at least annually (1,3)
In-Place Tests, Prefilters	Not required

NOTES:

(1) Acceptance tests are made after completion of initial construction and after any major system modification or repair

(2) More frequent (e.g., 6 months) testing may be required following initial startup of the system until a pattern is established.

(3) Adsorbents are tested before installation or replacement of adsorbers to establish suitability. Samples for laboratory testing are taken from the system at the same time as routine in-place testing of the installed system to verify the condition of the adsorbent.

All the above tests are conducted at the time of first installation of the units and at intervals shown in the above table in order to ensure that the contaminants are not by-passing the filters and that the filters are operating efficient.

The following are the efficiencies of the various filters based on vendor information.

Filter	Minimum Efficiency	Reference
Prefilters	55 - 60%	AAF
HEPA Filters	99.97% (On 0.3 u DOP)	Flanders
Charcoal	99.0% (for CH3 I-131)	Barnebey Cheney (Tech. Bulletin 717)
Charcoal	99.9% (for I-131)	Barnebey Cheney (Tech. Bulletin 717)
Charcoal	99.9% (for F112 leak test)	Barnebey Cheney (Tech. Bulletin 717)

V. REPLACEMENT SCHEDULES

A. Filter Replacement Schedule

The following is the initial filter replacement schedule based on past experience and on data presented in ERDA 76-21. The actual conditions encountered during operation of the units may affect this schedule depending on the actual filter loading rates. However, the maximum intervals at which they shall be replaced are also indicated.

Filter	Replacement Interval
Prefilter	Whenever the Differential Pressure Gage across the filter shows fully loaded condition or when the differential pressure alarm goes off) or two years, whichever comes first.
HEPA Filter	*Whenever the Differential Pressure Gage across the filter shows fully loaded condition or when the differential pressure alarm goes off) or two years, whichever comes first.

*The Differential Pressure Gages are monitored daily

CP&L will prepare the filter replacement schedule based on the actual operating conditions subject to the maximum periods indicated above.

B. Gasket and Seal Replacement Schedule

The only gaskets and seals that are present in the units, the deterioration of which due to thermal, radiation and humidity conditions of the air handled by the air cleaning units are

at the Prefilters and HEPA filters where the particulate matter is trapped. The charcoal adsorber section where the gaseous radioactive iodines and iodine compounds are trapped is of all-welded gasketless construction. The access doors have gaskets which are not directly located in the air stream and have a life of more than 40 years in the operating environment specified for the units, according to the vendor data. Certification was provided by the vendor to this effect. Also, these gaskets are inspected for signs of deterioration and wear every time the filters are inspected and replaced. If any damage is noticed the gasket is replaced. As the filter housings are maintained under negative pressure during operation, no leakage of contaminated air to the outside is possible even if an improbable leak exists in the door gasket.

As can be seen from the filter replacement schedule given in the previous paragraph the filters are replaced every two years at the latest. The silicone, neoprene and urethane materials used for gaskets and sealants in the filters last many times longer than the filter replacement period of two years under the normal operating environment given below.

Temperature: 104°F

Total Integrated Radiation Does:

$8x10^{-4}$ Rads

(40 years normal & 1 year post LOCA)

Humidity: Less than 70%

ATTACHMENT 7

Summary of Critiques of the "Heidelberg Report"

I. NRC Review, NUREG-0668

The NRC critique of the Heidelberg Report focused on the key elements of the source terms, atmospheric dispersion factors and dose conversion factors. The NRC found that the liquid and gaseous source terms used in the report were many times greater than the average source terms from operating plants in the U.S. and therefore do not reflect U.S. operating experience.

The NRC found that the atmospheric dispersion factors (X/Q) were derived using data assembled from five separate meteorological stations. No single station was used to obtain the basic wind speed, stability class and wind direction joint frequency data. As a result, combinations of wind speed, direction and stability class are used which have no meaning in reality. Using approximations, the NRC believes that the peak X/Q used in the Heidelberg Report may be high by a factor of 10 or more.

The NRC found that the soil to plant concentration factors (B_{iv} values) were not supported by the literature cited. Specifically, the Cs and Sr B_{iv} values were selected at the high end or well beyond the high end of the experimental data.

Also, the dose conversion factors for Cs-137 and Sr-90 are much higher than those used by the NRC and are not supported by experimental data. The NRC factors are based on International Council on Radiological Protection ("ICRP") guidelines.

As its final critique, the NRC reviewed environmental radiological surveillance data around operating reactors in the U.S. the NRC determined that if Heidelberg models were valid, then high, easily detectable levels of I-131 and Cs-137 would be found in the vicinity of operating reactors, when in fact they are not.

II. The University of Heidelberg

Dr. E. K. F. Brutz, Dean - Faculty of Biology of the University of Heidelberg, in a letter to the NRC, stated that the Heidelberg Report was not prepared or sponsored by the University, but rather was prepared by a group of graduate students aided by a few junior faculty members. The group represented itself as being sponsored by the University, against the direct instructions of the President of the University. Dr. Brutz referred to the group as a "bunch of students setting out to prove their philosophy right" and coming up with "data based more on fancy than fact." He was also apologetic that the NRC had to "go through the pains having to referee such silly claims."

The degree to which the University was concerned with the Heidelberg Report is demonstrated by the legal action brought by the University to restrain the authors of the Report from

referring to their group as the Tutorium Environmental Protection at the University of Heidelberg.

III. Ministerium Fur Arbeit Gesundheit Und Sozialordnung Baden (a German government nuclear power plant licensing agency)

J. Narrog of the German licensing agency was extremely critical of the Report, referring to it as "less a serious scientific report but rather a public relations paper of opponents against nuclear energy. All European institutions, which dealt with the report, came to similar statements."

On a more technical side, J. Narrog demonstrated that there was no agreement between the average radionuclide release rates for operating power plants in Germany and the Heidelberg Report source terms. In addition, site specific meteorological data have shown the Heidelberg dispersion factors to be at least 2 fold too high. Finally, J. Narrog stated that environmental surveillance programs at operating plants in Germany have "never showed numbers in the magnitude of the estimated concentration values of the Heidelberg Report for CS-137, CS-134 and I-131."

IV. Federal Energy Production Office - Nuclear Installation Safety Division (Germany)

Drs. J. Czarencki, J. Halter, H. Pfeiffer, H. Fritz-Niggli and H. Brunnen published an extensive critique of the Heidelberg Report. Quoting from the critique, "The selection

of the literature is one sided, obsolete and incomplete . . . . Insufficient or missing scientific knowledge and deficient knowledge of the literature result in serious assumptions and assertions and incorrect conclusions . . . . In its conclusions, it is based on unsubstantiated statements, incorrect assumptions, unidentifiable calculations, exaggerated and unrealistic numerical values for the parameters used."

V. National Radiological Protection Board, Harwell, England:

G. S. Linsley echoed many of the above statements. In addition, he cited experimental work in England which supported the B_{iv} values used by the NRC.

VI. Society for Reactor Safety (GRS)

The German Society for Reactor Safety was very critical of meteorological modeling used in the Heidelberg Report. They believe the Report's long-term diffusion factor is high by a factor of 3, and the deposition velocity of airborne particles is high by a factor of 4. The critique also included a retraction statement made by D. Teufel, co-author of the Heidelberg Report, regarding previous statements made in support of their dose conversion factors. Teufel admitted to the administrative court that, "[t]his cannot be checked by us precisely. The statement was probably made a little prematurely. It fell back to me, to my graduate work, in which this statement was made and which I must now retract."