UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of LONG ISLAND LIGHTING COMPANY (Shoreham Nuclear Power Station, Unit 1)	Docket No. 50-322 (O.L.) (Emergency Planning Proceedings)

DIRECT TESTIMONY OF FRED C. FINLAYSON

ON BEHALF OF SUFFOLK COUNTY

REGARDING

CONTENTION EP 14

(ACCIDENT ASSESSMENT AND DOSE ASSESSMENT MODELS)

October 12, 1982

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Summary Outline of Direct Testimony of Fred C. Finlayson on Behalf of Suffolk County Regarding Contention EP 14 (Accident Assessment and Dose Assessment Models)

LILCO's dose assessment models neglect several important processes that lead to the accumulation of radiation doses. Most importantly, the models calculate whole body dose without regard either for the dose resulting from the inhalation of fission products or the dose resulting from ground contamination. At distances close to the plant, the eight-hour ground dose contributes about 70% of the whole body dose; beyond ten miles, the long term effects of inhalation exposure are the dominant source of whole body doses. Thus, the neglect of these exposure processes in LILCO's models may lead to significant underestimation of whole body doses as well as underestimation of doses to other human organs.

In addition, LILCO's dose assessment models also neglect important fission product source terms; apparently the noble gases and halogens are the only fission products that LILCO plans to use as input in calculating doses. However, significant accidents contributing to public risk will release many fission products in addition to the noble gases and halogens. A comparative analysis of dose estimates using the very limited set of fissi[illegible] products incorporated in the LILCO models and estimates [illegible] a more extensive realistic set of fission products fr[illegible] risk accident scenarios, indicates that the LILCO m[illegible]ld substantially underestimate the magnitudes of predic[illegible]es.

Finally, LILCO's estimates of noble gas and halogen release rates for use in the dose assessment models are based primarily upon arbitrary assumptions concerning the ratio of noble gases and halogens in the mixture. The estimates are not based on measurements that reflect the actual fission product makeup of escaping radioactive gases and vapors. Actual halogen release rates could be substantially different (either greater or smaller) from the estimated rates. Thus, the projected thyroid dose estimates produced by the LILCO models are inherently unrealistic.

Attachments to the Direct Testimony of
Dr. Fred C. Finlayson on Behalf of Suffolk County
Regarding Contention EP 14 (Accident Assessment
and Dose Assessment Models)

ATTACHMENT 1

Resume of Dr. Fred C. Finlayson

ATT. .HMENT 2

Table 1 - Typical Fission Product Inventory for a BWR of Shoreham Size

ATTACHMENT 3

Table 2 - SAI Categorization of Potential Severe Accident Releases for the Shoreham Facility

ATTACHMENT 4

Figure 1 - Projected Probabilities of Exceeding Specified Doses as a Function of Distance From a Severe Nuclear Power Plan Accident (SAI Category 1)

ATTACHMENT 5

Figure 2 - Noble Gas Whole Body Doses (DBA) CBA-1

ATTACHMENT 6

Figure 3 - Mean Whole Body Dose for Severe Core Melt Accidents

ATTACHMENT 7

Figure 4 - Whole Body Dose Components for Severe Core Melt Accidents

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Q. Please state your name and your position.

A. My name is Fred C. Finlayson. I am the Principal Associate of F. C. Finlayson & Associates, 12844 East Cuesta Street, Cerritos, California. A copy of my professional qualifications is attached to this testimony as Attachment 1.

Q. What is the purpose of this testimony?

A. The purpose of this testimony is to address certain of the concerns raised in Suffolk County's emergency planning contention EP 14--Accident Assessment and Dose Assessment Models. The contention states:

> LILCO's plan fails to provide reasonable assurance that adequate methods, systems and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use, and therefore does not comply with 10 CFR §50.47(b)(9).

In this testimony, I will address the issues in contention EP 14 associated with the adequacy of LILCO's methods of accident and dose assessment, in particular the methods used for dose assessment.

Q. Please describe your experience relative to methods for assessing consequences of a radiological emergency.

A. For the past ten years, I have been working extensively in a variety of projects related to the evaluation of reactor safety. Over the last six years, I have been responsible for directing, managing, and performing research projects in which reactor risk assessment and public health consequence assessment for severe nuclear power plant accidents were significant factors. For example, I supported the State of California, Office of Emergency Services, in the technical management of site unique probabilistic risk assessments and consequence analyses conducted for each of the nuclear power plant sites in California. I am currently a member of the review committee for the "PRA Procedures Guide" (NUREG/CR-2300) that is under development by the NRC, the American Nuclear Society (ANS) and the Institute of Electrical and Electronic Engineers (IEEE). As a member of the review committee, I have concentrated my efforts primarily upon evaluating the consequence assessment methods recommended in the PRA Procedures Guide.

In addition, I have been involved in the assessment of radioactively induced health effects and fallout patterns from nuclear weapons since 1960. More recently, I managed a program in which the relative consequences of severe nuclear power plant accidents in above-ground and underground facilities were examined. In all of these studies, state-of-the-art radioactive plume modeling and public health consequence assessment models have been used.

Q. Are you familiar with the methods, proposed to be used by LILCO, for assessing and monitoring actual or potential consequences of a radiological emergency?

A. I have reviewed Section 6.1 of the Shoreham Emergency Plan, Section 2 of the Shoreham Nuclear Power Station Offsite Dose Assessment Methodology for Emergency Applications prepared by Entech Engineering, Inc. (the "Entech model"), and the Shoreham Emergency Plan Implementing Procedure SP 69.022.01 Revision 0, Determination of Offsite Doses (effective date July 9, 1982). It is my understanding that these documents reflect the methods which LILCO will use to project offsite doses, prior to receipt of the results of offsite surveys and sampling, in the event of a radiological emergency. I have also reviewed Emergency Plan Implementing Procedures SP 69.023.01, Revision 0, Thyroid Dose Commitment Using TCS Air Sampler (effective date July 9, 1982), and SP 69.023.01, Revision 0, Downwind Surveys (effective date July 9, 1982), which reflect methods that LILCO proposes to use to determine thyroid doses based upon offsite surveys and sampling conducted during an emergency. As far as I know, these documents contain all the information provided by LILCO to the County with respect to the accident and dose assessment methods to be used by LILCO in the event of an accident.

Q. Please explain what is meant by the term "dose assessment model."

A. In the broadest sense, a dose assessment model is a mathematical tool by which the health impacts of accidentally released radioactive fission products upon humans can be determined as a function of: magnitude of the accidental releases (including both the quantities and specific constituents of the released fission products); the wind and weather conditions at the time of the accident; and, the distance of individuals from the site of the accident. Protective actions taken by the individuals, such as evacuation, and taking shelter from direct exposure to the radioactivity, can also have a significant impact on the magnitude of potential radiological doses that they might receive. The influences of protective actions on potential doses to exposed individuals are also customarily included in dose assessment models. Thus, the basic elements of a dose assessment model are: (1) an accurate description of the characteristics of the radionuclide release; (2) an acceptable mathematical model of the radioactive plume propagation and dispersion that can be used to define the concentration of radioactive fission products in the air and on the ground; (3) an adequate description of weather conditions, including atmospheric stability, wind speed and direction, and precipitation; (4) a model of the relationship between health effects and radioactive exposures; and, finally, (5) a method for evaluating the effects of protective actions taken by individuals to reduce radioactive exposures and/or doses.

Q. In your opinion, do the dose assessment models proposed to be used by LILCO prior to obtaining the results of offsite surveys provide an adequate method for assessing actual or potential offsite consequences of a radiological emergency condition?

A. Procedure SP 69.022.01 refers to two different LILCO dose assessment methods for estimating offsite doses. One is a computerized "radiation monitoring system" (of which the Entech model is evidently one part) and the other is a manual method. Neither the Entech model nor the LILCO Plan or procedures I have reviewed provide any details on the input to the computerized method, nor its output. I understand, however, that the manual model incorporates the same mathematical procedures for calculations to be performed as the computerized system does.

[Deposition of H. Mark Blauer, August 23, 1982, at 146-147]. The mathematical procedures in the LILCO model are based upon a straight-line trajectory model using Gaussian dispersion rules that were derived from the NRC Regulatory Guide 1-111 (Ref., Entech Model, p.5).

The manual method is based upon utilization of eight pre-calculated nomograms that provide the operators with pictographic tools for scaling measurements of the radioactivity of fission products escaping through the station ventilation system and the Reactor Building Standby Ventilation System (RBSVS), to estimate offsite whole body and thyroid doses at locations no more than ten miles from the plant site.[1/]

Many attempts have been made to develop accurate plume propagation and dispersion models. No thoroughly satisfactory methods for advance prediction of cloud motions exist as a result of the complexity of realistically modeling the motions of the earth's atmosphere. However, over the years, many simplifying modeling assumptions for assessing radioactive cloud motions have been made that have been broadly accepted by the technical community. The straight-line, Gaussian plume dispersion model adopted by LILCO is relatively unsophisticated, but

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1/ The current edition of SP 69.022.01 includes only five of the eight nomograms; it indicates that the remaining three will be provided "later."

one that is widely accepted in principle, within the technical community, for conducting long-term (annualized) risk assessments. These types of models are not generally useful for advance projections of real-time meteorological events -- especially those where the predicted motions of the cloud must be accurately portrayed over long distances. Changes in wind direction over time and distance will frequently cause the plume trajectory to deviate from a straight-line. Moreover, under certain very stable or nonstable atmospheric conditions, the Gaussian assumptions for dispersion begin to be invalid. When used for the advance projection of doses from real meteorological conditions, it is generally conceded that such models can be accurately applied over distances no greater than ten to twenty kilometers (or about five to ten miles in round numbers)(Ref: PRA Procedure Guide, NUREG/CR-2300 (Draft), Appendix D, p. 4).

The plume atmospheric dispersion and dose modeling methods that are outlined in the Entech model follow NRC Reg. Guide recommendations (Reg. Guides 1.111 § 1.145) reasonably well. However, it should be noted that the Reg. Guides cited were developed primarily for establishing long-term (annualized) risks from operation of nuclear power plants. For these circumstances, the use of straight-line trajectory, Gaussian plume models is relatively reasonable since, over the course of a year, clouds would follow meandering paths that would cover essentially all geographic locations around a site. Under these circumstances, straight-line trajectory models that incorporated directional probabilities based upon annualized wind-rose frequencies would be expected to provide results that would be similar to more sophisticated numerical methods that simulated actual meandering in cloud paths.

For purposes of predicting off-site doses following an accident, however, the straight-line, Gaussian plume models have far greater uncertainty in their results, than they do for the calculation of annualized average values of potential doses. Therefore, the plume dispersion aspects of the LILCO models are not particularly well suited for real time projection of cloud paths or for accurate estimation of the locations for particular radioactive exposure levels. They do, however, provide a first-order basis for estimation of cloud fission product concentrations that might occur in a straight line, downwind from an accident.

However, the LILCO models evidently include certain assumptions concerning the number and quantity of the constituent fission products in an accidental release that are not sufficiently general for the spectrum of severe accidents that emergency plans must consider. Therefore, in this respect the models are not currently applied in an adequate fashion.

Q. Please describe in more detail the inadequacies in LILCO's dose assessment models with respect to the assumptions concerning fission product releases.

A. My conclusions are based upon the limited amount of information available in the documents that I have reviewed relative to the LILCO fission product release assessments. These documents suggest that the fission product constituents that are considered in the LILCO analysis are limited to noble gases and some fractional releases of halogens (principally iodine isotopes). In fact, it appears that LILCO intends to use only these two categories of fission products without regard to potential releases from the reactor of a much larger spectrum of the fission product inventory under more severe accident conditions that would provide a greater challenge to emergency planning. The accidental releases in these latter categories would be expected to be the source of greater public risks than those associated with releases of only noble gases and a small fraction of the iodine fission product inventory.

A review of Shoreham Procedure 69.022.01, Section 3.4, indicates that the LILCO models assume a 100% release of noble gases and a 25% release of the halogen inventory within the containment building. They then apparently assume that internal stack filters will have a 99% efficiency for halogens. Thus, only 1% of the original 25% of the halogens released in the containment building would be released to the atmosphere (i.e., the fractional release of halogens to the atmosphere would be .0025 times the initial halogen inventory in the reactor core).

Q. What is the difference between LILCO's use of a fission product release inventory consisting of only noble gases and halogens, and the fission product release categories you have described for severe reactor accidents.

A. Table 1 (Attachment 2 to this testimony) contains a description of the inventory of the more significant fission products for a Boiling Water Reactor (BWR) such as Shoreham. Listed on Table 1 are the quantities of the more significant isotopes that are contained in the reactor core. Of the elements listed, those that are bracketed fall within the categories of "noble gas" and "halogen" (or iodines) as noted.

In the event of a severe accident, fractional releases of some or all the elements contained in the fuel could occur to the atmosphere. The precise contents of the release would vary depending upon the events involved in the accident. Accidental releases are customarily grouped into categories that reflect common accident characteristics. Such a grouping was performed for the Shoreham plant by Science Applications Inc. (SAI) in the Probabilistic Risk Assessment (PRA) that they conducted for LILCO. Table 2 (Attachment 3 hereto) has been reproduced from the SAI-PRA. It sets forth the release descriptions developed by SAI for five of the more probable classes of severe accidents (though all are projected to have low absolute probabilities).

The fraction of the total inventory of chemically and physically related groups of fission products that would be released in the event of an accident are shown in Table 2. The results shown in Tables 1 and 2 indicate that accidents can result in releases of many other fission products besides the noble gas and halogen categories. Although many of the fission product groups have relatively small release fractions, I will show below that calculations which exclude those fission products would lead to significantly lower dose estimates than those that are obtained from analyses that include their effects.

Q. Does the fact that certain fission products are not incorporated as inputs to the LILCO model have any impact on the whole body doses that would be calculated using that model?

A. If the LILCO whole body dose predictions are based exclusively upon the noble gas fission product releases, estimated doses would generally be relatively low compared with those that would be calculated with the fission product inventories shown in Table 2 -- i.e., those that SAI found to represent the releases of the more significant risk inducing accidents at

Shoreham. This is the result of two factors: The first is associated with the physical mechanisms by which individuals accumulate radiological doses; and the second is associated with the importance of fission product constituents besides noble gases and iodines that contribute to whole body doses. I will address each of these factors in turn.

There are five dominant ways in which people can accumulate radiation doses after an accidental release of radioactivity to the atmosphere:

1. Inhalation.
2. Exposure to external irradiation from the passing cloud (cloudshine).
3. Exposure to external irradiation from the deposited radionuclides (groundshine).
4. Ingestion, including contaminated vegetation, milk, milk products, and crops contaminated by root uptake.
5. Inhalation of resuspended radioactivity.

For estimating early effects such as deaths or injuries resulting from exposure to the radioactive cloud, the most important of these pathways are (1) inhalation from the cloud, (2) cloudshine, and (3) short-term exposure from contaminated ground (over periods lasting from hours to days). For estimating latent health effects such as cancers, the important pathways include (1) external exposure from contaminated ground (both short and long term), (2) inhalation exposure from the passing cloud and from the subsequent resuspension of radionuclides, and (3) the ingestion of contaminated foods.

In emergency planning for severe nuclear accidents, early effects, including deaths and injuries occuring within 30 to 60 days after exposure (or less) are most frequently of concern. Latent cancer effects from relatively short periods of external exposure to contaminated ground and inhalation exposure from the passing cloud can also be strongly influenced by protective actions associated with the emergency plan. An effective dose assessment model will include consideration of all such exposure processes.

The first and most significant weakness in LILCO's exclusive use of noble gases to estimate the whole body dose, is that the noble gases contribute almost exclusively to the cloud dose; they play an essentially negligible role in inhalation doses and do not contribute at all to ground doses. As previously noted, many of the chemically active radioactive fission products (as opposed to the chemicals, inert noble gases) that are inhaled into the lungs during the time of the passage of the cloud, as well as those which are on the ground, also contribute significantly to the whole body dose. Therefore, a realistic analysis of the whole body dose must consider the contributions from each of these elements -- materials inhaled into the lungs, the radioactive cloud emissions during its passage, and radioactive materials that have fallen out of the cloud and have been deposited on the ground. If the whole body dose assessment were limited to the contributions from noble gases, the contributions to health effects consequences from inhalation dose and ground dose would, for all practical purposes, be neglected.

The second weakness associated with the exclusive use of noble gases for predicting whole body doses is that they represent a relatively small fraction of the significant fission product contributors to the total cloud dose that might be delivered to an individual at a given distance from the reactor under the risk-inducing accident scenarios shown in Table 2. As shown in Table 1 and Table 2, many fission products, in addition to noble gases, would be released in the event of a severe accident. A number of these other fission products, carried along in the cloud, could be significant contributors to the whole body cloud dose, such as Te-131 & 132; Sb-129; I-131, 132, 133, 134, & 135; and La-140, for example. Thus, if LILCO's assessment of whole body dose were limited to the noble gas contributions to the cloud dose, important health-effects contributions from fission products other than noble gases that might be carried in the cloud could be neglected.

Q. Can you quantify the effect of the LILCO model's neglecting certain fission products in the calculation of whole body doses?

A. Yes. Figure 1 (Attachment 4 hereto) shows the whole body dose, as a function of distance and probability, for typical Shoreham weather conditions given that one of the more severe accidents (SAI Category 1) has occurred. Figure 2 (Attachment 5 hereto) shows the results that are obtained if you calculate the whole body dose based exclusively upon a 100% release of noble gases, as the LILCO model assumes. It can be seen that there is a substantial difference in the values of the doses associated with the release of a full range of fission products as compared to the case where only the effects of noble gases are considered. For example, Figure 1 indicates that the probability of exceeding 200 rem (a whole-body dose at which most exposed individuals would show definite signs of early injuries) exceeds one percent out to a distance of about four miles for the SAI Category 1 accident scenario shown. In Figure 2, where it is assumed that only noble gases are released, the probability of exceeding a 200 rem dose becomes vanishingly small beyond 1 mile from the reactor. At 30 rem (a relatively large dose, but one for which ordinary laboratory or clinical methods would generally show no indications of radioactively-induced early injuries), the results shown in

Figure 1 indicate that there is a 1 percent probability of exceeding this dose out to about 30 miles; and a 10 percent probability (or greater) of exceeding 30 rem out to about 15 miles. In Figure 2, the probability of exceeding 30 rem becomes vanishingly small beyond 4 miles. Thus, the results shown in Figures 1 and 2 suggest that for the risk-inducing accidents involving the release of the complete spectrum of fission products, the distance at which there is an equal probability of exceeding a specified dose can be increased by a multiple of about five to ten.

Q. Please explain how the curves shown in Figures 1 and 2 were derived.

A. The curves in Figures 1 and 2 are based upon a series of calculations using the CRAC-2 code. The CRAC-2 code is a computerized risk assessment program that can be used for assessing doses on a probabilistic basis, or for direct evaluation of the long-term probability of health effects of all types from accidental releases of radioactivity from a nuclear power plant and/or the associated economic consequences. The operator of the code may define the fission product source term as broadly or as narrowly as he wishes within the limits of the 54 radioactive isotopes shown in Table 1. As indicated in Table 2, fission product releases may be categorized in terms of their probabilities and results for individual accident scenario categories evaluated singly, or several categories may be combined into integrated estimates of doses and consequences in accordance with their individual probabilities. Risks are annualized by using a statistical sampling of a year's worth of hourly weather data for the site as a basis for defining wind-speed, stability, and precipitation characteristics of the local weather at the site. The annualized risks are generally based on the results of consequence calculations for a sample of about 100 explicit hourly weather sequences taken over the representative year's worth of data.

The release characteristics for Figure 1 are exactly those shown for the SAI Category 1 release shown in Table 2. In performing the CRAC-2 calculations for the noble gas releases, the same set of atmospheric and meteorological conditions were used that were utilized in the assessment of the doses resulting from the more extensive set of released fission products used in the calculations for Figure 1. However, for the noble gas induced consequences shown in Figure 2, we have restricted the accidental releases from the plant to an exclusive release of 100% of the noble gases.

Q. How do the results reflected in Figures 1 and 2 relate to the results that would be obtained using the LILCO dose assessment models?

A. Aside from the mathematical models for estimating plume dispersion and fission product concentrations in the radioactive cloud that are described in the Entech model, as discussed above, it is difficult to be sure exactly what is embodied in LILCO's manual dose estimation procedures or in the computerized method. This is because the exact makeup of the fission products that are used as inputs by the Entech model are not presented in the document describing the model, nor is the basis for the development of the nomograms specified in the Procedures document. Assuming that the 100% noble gas fission product release is the basis for the whole body dose estimates shown in the LILCO nomograms, then for comparable fission product release rates the whole body doses predicted by the LILCO nomograms should be comparable to the noble gas-only curve shown in Figure 2. Accordingly, the LILCO whole body dose projections would apparently be substantially lower than projections based upon a more extensive fission product source term description, as shown by comparing Figures 1 and 2.

Q. Would the substantial difference in dose projection that you just mentioned have an impact on emergency planning decisions made by LILCO?

A. Yes, it seems very likely that it would. If the LILCO procedure is limited to assessment of doses which are based upon an assumed release ratio of noble gases to halogen fission products, then a more severe accident, with a more extensive fission product release, would probably result in doses greater than those predicted by the LILCO models. If lower doses were predicted by the use of the LILCO model than would have been predicted using a more significant description of released fission products, then the decisions necessary for successful protective actions might not be made in a timely fashion. The operators might not realize that dose projections exceeding Protective Action Guide (PAG) triggering levels could be reached in the early stages of the event. Even if subsequent measurements indicated that dose levels had exceeded expectations and PAG trigger levels were in fact exceeded (for instance, after offsite sampling had been done and reported back to the site), valuable time would have been lost as a result of initial reliance upon low dose projections which were not properly modeled. If, on the other hand, a more realistic dose modeling method were utilized, the projections of the action-triggering dose levels might be provided at an earlier point in the accident sequence, thus providing more time for implementing the necessary protective actions.

Q. Is there any other impact on the adequacy of LILCO's whole body dose assessment arising from the consideration of only noble gases?

A. Yes. As I mentioned before, there is an implicit assumption in LILCO's use of only noble gases in projecting the whole body doses that the dose is contributed exclusively by the cloud. In fact, if the breakdown between the contribution from the cloud and the two other major sources of dose in Figure 1 were shown -- those contributions from inhalation through the lungs and from fission products entrained on the ground -- we would find that both the inhalation and ground dose provided major contributions to the total whole body dose. Figure 3 (Attachment 6 hereto) and Figure 4 (Attachment 7 hereto) show a representative breakdown of contributions from inhalation, cloud and relatively short (8-hour) ground dose exposures. From Figure 4, it can be seen that at close-in distances the 8-hour ground dose contributes about 70% of the total whole-body dose. Beyond ten miles, the long-term effects of inhalation exposure to the cloud become the dominant source of whole-body doses. Over a period of a few days, contributions from each of the three elements of the whole body dose are quite similar in magnitude. For all practical purposes, both the inhalation and ground dose contributions are neglected if you assume you are only releasing the noble gases, or that only cloud doses are significant -- an explicit feature of the LILCO dose assessment models (Ref.: Entech Model, p.2).

Q. So far we've only been discussing whole body dose calculations. Have you reviewed the thyroid dose calculations contained in the LILCO dose assessment model?

A. Yes, I have reviewed those referenced in the Entech model, in SP 69.022.01 and in SP 69.023.01.

Q. According to the LILCO models, how are thyroid doses projected prior to receipt of offsite survey results?

A. The LILCO procedure SP 69.022.01 indicates that nomograms are used to project thyroid doses in the following manner. The first step in applying the nomograms requires the user to estimate the iodine release rate. Under normal conditions, the person responsible for making the thyroid dose predictions derives his initial estimate of the radioactive iodine source strength from a reading of the effluent monitor in the Reactor Building Stand-By Ventilation System (RBSVS) stack. This monitor measures the radioactivity being released from the stack in terms of the "counts per minute" of ionizing particles released by the fission products as they pass by the monitor. The Radiation Protection Manager (or an aide) then uses the nomograms to estimate the release rate of iodines.

The use of the nomograms depends upon several assumptions that are not very explicitly defined. One is related to the constituency of the mixture of gaseous and vaporized fission products assumed to be escaping from the containment building.

Radioactivity measurements taken from the station ventilation system exhaust monitor are evidently assumed to be derived from only noble gas fission products (Xe and Kr). The nomograms based upon RBSVS monitor readings appear to incorporate an inherent assumption about the specific fission product ratios of the mixture of noble gases and halogens (principally Iodines) escaping from the stack. How the operator can tell from the radioactivity measured from these sources whether the ratios of noble gases to halogens is in accord with the assumption, or what he would or should do if it were not, is not described in the procedures.

Q. Is the technique used to estimate the iodine release rate in this procedure justified?

A. Not necessarily. The ability of an operator to estimate accurately the actual iodine release rate depends upon his recognizing that there may be various kinds of accidents at the plant. Some of them may release iodine quantities in proportions that differ from those that have been assumed in the methods incorporated into Procedure SP 69.022.01. Accordingly, the reasonableness of the iodine release rate estimates depends upon the adequacy and the accuracy of the measurement methods used. Unless a more sophisticated procedure is used to establish the fission product makeup of the gases escaping from the reactor containment building, it is not clear to me that the operator can accurately determine whether the reading in counts per minute given by the stack monitor was derived from gamma rays from iodine or from some other gamma emitting fission product. If the operator is to make reasonable estimates of the potential offsite thyroid doses, then it is very important that he be able to make accurate estimates of the actual radioactive iodine release rates. In my opinion, a more definitive measurement method would be a desirable improvement over LILCO's apparent reliance on an assumed fission-product make up of escaping gases. Procedure SP 69.022.01 does specify in Section 8.2.1.11 (p. 6) that if the radiation monitor is either inoperable or its reading is off-scale, a "grab" sample from some source (probably the RBSVS stack) is to be obtained as a basis for the source estimates for iodine or xenon release rates. I am not sure what equipment will be used by LILCO to assess the fission product makeup of the grab sample or how quickly such assessment could be accomplished. Under ideal circumstances, a sample of this type could be used to provide a more accurate definition of the constituent fission product makeup of the escaping gases. However, because the LILCO models identify the grab sample method as one to be used only if the less accurate radiation monitors are unavailable, LILCO appears to base its thyroid dose assessment primarily on iodine release rates using assumed rather than measured values.

Q. In your opinion, does the use of an assumed ratio of noble gases to iodines in the stack effluent represent a conservative approach for purposes of thyroid dose assessment?

A. This again involves the issue of the realism of the operator's projection of the iodine release rate from the containment. It is very important for an operator to have accurate estimates of the quantity of iodine actually released if his estimates of the thyroid doses are going to be accurate. If the operator makes his estimate of the iodine release rate from the nomograms of Procedure SP 69.022.01 based upon stack monitor count rate from the RBSVS, then the actual ratio of iodines to other gamma emitting fission products _is_ important. If the actual fractional quantity of iodine in the stack gases is greater than the assumed values upon which the nomograms are based, then the Radiation Protection Manager's estimate of thyroid dose would almost certainly be too low. Hence it does not appear that the approach used can always be considered conservative. If we have reason to believe that the operator's release estimates may not be accurate, then it would be more appropriate to use a more accurate source of data for defining the iodine release rate. If this could not be done in a timely fashion, then it would be more conservative to assume higher fractional concentrations of iodine to the releases for purposes of dose projections. In my opinion, those estimated concentrations should make use of the results of a study of severe accidents and their respective probabilities such as a thoroughgoing PRA.

Q. In your opinion, do the LILCO dose assessment models demonstrate that the methods to be used to assess thyroid doses prior to receipt of data from offsite survey teams are adequate?

A. As indicated above, there is a need for better methods of determining the iodine release rates from an accident. In addition, all the weaknesses previously described with respect to the size and propagation path of the radioactive cloud in the atmospheric dispersion methods of the Entech model are as germane to thyroid dose estimation as they are to whole-body dose estimates. The key question is whether the models have <u>demonstated</u> adequacy. In the LILCO documents I have reviewed, in my opinion, the documentation does not now demonstrate adequacy, although the methods could be the basis for reasonably adequate thyroid dose predictions, if the qualifying weaknesses were eliminated.

Q. Could the concerns you have identified with respect to LILCO's thyroid dose assessment have an impact on emergency planning decisions?

A. Yes. For the reasons I discussed above, the nomograms as they are now constituted could conceivably yield unrealistic doses estimates that might not meet protective action guides (PAG) criteria for implementing protective actions, when a more reliable estimate would have called for their implementation. Or, under other circumstances, unreliable dose estimates might trigger decisions to take protective actions unnecessarily.

Q. Do you have any other concerns about the adequacy of the LILCO dose assessment models?

A. Yes. According to SP 69.022.01, the dose estimates provided by the nomograms are restricted in their applicability to ten miles or less from the plant.

Q. What is the significance of limiting the LILCO dose assessments to distances of ten miles or less from the plant?

A. As seen in Figure 1, there is a high probability of exceeding the 1-5 Rem whole body dose protective action guide limits at distances beyond the ten-mile range. In fact, it shows a 50% probability of exceeding the 1 Rem PAG level at distances of thirty miles. Although it might be possible to extrapolate from the LILCO models to determine doses at distances beyond ten miles from the plant, such an extrapolation would require the operator to observe that the dose values at the ten-mile limit exceeded the PAG levels and then to use some simplified procedure for estimating how far beyond the arbitrary 10 mile boundary the dose levels would be exceeded. If advance preparation for such a procedure is not made, it is not clear what mechanism might be used in an ad hoc fashion to make the extrapolations. Consequently, I would recommend that advance preparations be made to provide the Radiation Protection Manager with this capability before he must make the judgment under emergency conditions.

Q. Dr. Finlayson, please summarize your testimony on EP 14.

A. In summary, LILCO has the basic mathematical tools for a relatively unsophisticated cloud dose assessment model that is based upon straight-line trajectory, Gaussian dispersion methods. As presently constituted, however, this model neglects several important processes that lead to the accumulation of radiation doses. For emergency planning purposes where rapid responses are required, the most important among these neglected irradiation processes are the inhalation process leading to internalization of fission products within the body, and the ground dose exposure process. As indicated in Figures 3 and 4, neglect of these exposure processes may lead to significant underestimation of whole body doses as well as underestimation of doses to other human organs.

In addition, the written evidence provided in LILCO's documentation of the dose assessment methods suggests that important fission product source terms are being neglected that should be included as input to the mathematical tools of the plume dispersion, radioactivity concentration, and exposure-to-dose conversion elements of the methodology. Apparently, the noble gases and halogens are the only fission products that LILCO plans to use as input to their dose assessment models. As indicated in Table 2, probabilistic risk assessments of the Shoreham plant have shown that the more significant accidents contributing to public risks will release many fission products in addition to the noble gases and halogens. The results of comparative analyses of dose estimates for the limited set of fission products incorporated in the current LILCO procedures and those for the more extensive releases from high risk accident scenarios indicate (as shown in Figures 1 and 2) that the LILCO procedures could substantially underestimate the magnitudes of predicted doses.

Noble gas and halogen release rates are primary inputs to LILCO's mathematical dose assessment tools. LILCO's principal method of estimating these release rates is based upon measurement of ionizing radiation from fission products escaping through the normal plant ventilation system and/or the reactor building standby ventilation system. The monitors in these ventilation systems do not provide an explicit method for determining the details of the fission product make up of the escaping radioactive gases and vapors. Thus LILCO's dose estimation procedures are based upon arbitrary assumptions concerning the ratios of noble gases and halogens in the mixture.

Actual halogen release rates could be substantially different (either greater or smaller) than the estimated rates that are based upon the ventilation system monitor measurements. Since the projected thyroid dose estimates are directly related to the magnitude of the input halogen release rates, the output doses will be inherently unreliable if the input data is not precisely defined.

LILCO's projected whole body dose estimates could lead decision makers to fail to institute important protective actions since they may be substantially lower than actual doses in the field. On the other hand, LILCO's thyroid dose estimates may be inherently unreliable, leaving decision makers without a reasonable basis for instituting protective action decisions. In both cases, improvements in the methods for predicting such doses appear to be needed.

ATTACHMENT 1

FRED C. FINLAYSON
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REACTOR SAFETY ASSESSMENT
NUCLEAR POWER PLANT
PROBABILISTIC RISK ASSESSMENT
REACTOR ACCIDENT CONSEQUENCE ASSESSMENT
ENERGY SYSTEMS DESIGN AND ANALYSIS

BACKGROUND SUMMARY

Dr. Finlayson has extensive experience in the field of assessment of the safety and risks of nuclear power reactors. He recently provided technical direction of the probabilistic risk analyses conducted for the State of California's evaluation of Emergency Planning Zone requirements. He was the principal investigator and program manager of the NRC's first investigation of the adequacy of human engineering in nuclear power plant control rooms under severe accident conditions. He is currently conducting an investigation for the NRC of the feasibility of instituting a specific reporting system for human errors in nuclear power plants. Dr. Finlayson was also the manager of The Aerospace Corporation program that provided systems integration and technical direction of the California Energy Commission's study of underground nuclear power plant designs, costs, and their relative effectiveness in reducing the consequences of extremely severe accidents.

Dr. Finlayson has been a consultant to the NRC, U.S. General Accounting Office, and other federal and state governmental agencies on nuclear safety related issues such as site-specific risk analyses, human engineering, large-scale reactor test program design and effectiveness, sabotage, waste transport hazards, and a wide variety of other related topics. He is a member of the review committee for the "PRA Procedures Guide" (NUREG/CR-2300) for probabilistic risk assessments for nuclear power plants that is being prepared under a joint NRC-industry-technical society effort. He served as a member of the NRC's 1980/1981 LOFT Special Review Group and was consultant to the NRC's Rogovin Special Inquiry Group in their investigation of human engineering factors associated with the Three Mile Island incident. He has performed several assessments of the design and effectiveness of ECCS for LWRs, including the analysis conducted for the American Physical Society's Review Committee (1975) on Light Water Reactor Safety.

EDUCATION

BS Mechanical Engineering, Brigham Young University, 1958
PhD Mechanical Engineering, Northwestern University, 1964

EXPERIENCE

The Aerospace Corporation, Los Angeles, CA (1972-Present). Dr. Finlayson is currently Manager, Nuclear and Geothermal Systems, Energy and Resources Division. In this capacity, he is responsible for nuclear, geothermal, and energy conservation projects.

He directed the systems engineering and technical management efforts for the recent California study of statewide nuclear power plant risks and associated emergency planning zone requirements; and directed a similar program for a major study of underground nuclear power plant siting. He was also the program manager for an assessment of the impact of plutonium fuel cycle safeguards, and an evaluation of nuclear control room human engineering. He has also managed and performed systems analyses of industrial process heat applications of geothermal power as well as conceptual design and evaluation studies of hybrid solar/geothermal power systems. Studies of local and national energy consumption patterns and the effectiveness of selected conservation measures have also been performed under his direction.

Physics International Company, San Leandro, CA (1968-1972). Dr. Finlayson directed and conducted research in strategic and tactical weapon systems survivability/vulnerability, numerical analyses of the propagation of strong shocks in geologic media and structural materials, and structure-medium interactions.

The Aerospace Corporation, Los Angeles, CA (1964-1968). Dr. Finlayson conducted investigations of ground based system survivability to all relevant effects of nuclear weapons.

The General American Transportation Corporation, Chicago, Ill (1960-1964). Dr. Finlayson conducted research on the interactions of strong shocks in air and earth materials with above-ground and buried structures.

PROFESSIONAL ACTIVITIES

Dr. Finlayson is a registered Professional Nuclear Engineer in the State of California. He is a member of the American Nuclear Society and the Institute of Electrical and Electronics Engineers (Reliability Society).

PUBLICATIONS

"Closures for Hardened Protective Hangers", AFSWC-TDR-62-77, MRD Division of General American Transport Corporation, Niles, Illinois, August 1962 .

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ATTACHMENT 2

Table 1

Typical Fission Product Inventory for a BWR of Shoreham Size

NUMBER	NAME	GROUP	PARENT	INITIAL(CURIES)	HALF-LIFE(DAYS)
1	CO-58	7		5.595E+05	7.130E+01
2	CO-60	7		3.371E+03	1.921E+03
3	KR-85	1		4.979E+05	3.919E+03
4	KR-85M	1		2.345E+07	1.867E-01
5	KR-87	1		4.272E+07	5.278E-02
6	KR-88	1		5.768E+07	1.167E-01
7	RB-86	4		3.611E+04	1.865E+01
8	SR-89	6		7.191E+07	5.200E+01
9	SR-90	6		3.878E+06	1.026E+04
10	SR-91	6		9.285E+07	3.950E-01
11	Y-90	8	SR-90	4.160E+06	2.670E+00
12	Y-91	8	SR-91	8.768E+07	5.880E+01
13	ZR-95	8		1.117E+08	6.550E+01
14	ZR-97	8		1.171E+08	7.000E-01
15	NB-95	8	ZR-95	1.055E+08	3.510E+01
16	MO-99	7		1.241E+08	2.751E+00
17	TC-99M	7	MO-99	1.071E+08	2.508E-01
18	RU-103	7		9.330E+07	3.959E+01
19	RU-105	7		6.158E+07	1.850E-01
20	RU-106	7		2.169E+07	3.690E+02
21	RH-105	7	RU-105	4.181E+07	1.479E+00
22	SB-127	5		5.790E+06	3.800E+00
23	SB-129	5		2.036E+07	1.808E-01
24	TE-127	5	SB-127	5.589E+06	3.896E-01
25	TE-127M	5		7.379E+05	1.090E+02
26	TE-129	5	SB-129	1.910E+07	4.861E-02
27	TE-129M	5		5.024E+06	3.340E+01
28	TE-131M	5		9.608E+06	1.250E+00
29	TE-132	5		9.510E+07	3.250E+00
30	I-131	3	TE-131M	6.553E+07	8.040E+00
31	I-132	3	TE-132	9.645E+07	9.521E-02
32	I-133	3		1.380E+08	8.667E-01
33	I-134	3		1.513E+08	3.653E-02
34	I-135	3		1.301E+08	2.744E-01
35	XE-133	1	I-133	1.381E+08	5.290E+00
36	XE-135	1	I-135	2.850E+07	3.821E-01
37	CS-134	4		9.458E+06	7.524E+02
38	CS-136	4		2.933E+06	1.300E+01
39	CS-137	4		4.903E+06	1.099E+04
40	BA-140	6		1.261E+08	1.279E+01
41	LA-140	8	BA-140	1.288E+08	1.676E+00
42	CE-141	8		1.145E+08	3.253E+01
43	CE-143	8		1.114E+08	1.375E+00
44	CE-144	8		6.867E+07	2.844E+02
45	PR-143	8	CE-143	1.091E+08	1.358E+01
46	ND-147	8		4.897E+07	1.099E+01
47	NP-239	8		1.388E+09	2.350E+00
48	PU-238	8	CM-242	8.760E+04	3.251E+04
49	PU-239	8	NP-239	1.936E+04	8.912E+06
50	PU-240	8	CM-244	2.170E+04	2.469E+06
51	PU-241	8		4.066E+06	5.333E+03
52	AM-241	8	PU-241	2.718E+03	1.581E+05
53	CM-242	8		1.027E+06	1.630E+02
54	CM-244	8		[illegible]	[illegible]

ATTACHMENT 3

Table 2
SAI Categorization of Potential Severe
Accident Releases for the Shoreham Facility

DRAFT - PRELIMINARY

Table 4.2

SUMMARY OF SNPS PRA RESULTS, OVERPRESSURE EVENTS

ACCIDENT CLASS	(a) CORE VULNERABLE FREQUENCY	(b) CONDITIONAL FREQUENCY OF RELEASE	(c) RELEASE BOUNDS	(d) ACCIDENT RELEASE DESIGNATOR	(e) FRACTION	(f) TIME OF RELEASE	(g) DURATION OF RELEASE	(h) WARNING TIME FOR EVACUATION	(i) ELEV OF RELEASE	(j) AVG ENERGY OF RELEASE	RADIONUCLIDE RELEASE FRACTIONS Kr-Xe	OI⁻	I[l]	I⁻	Cs[m]	Te[n]	Sr[o]	Ru[p]	La[q]
1	2.7×10^{-5}	8.0×10^{-2}	UPPER	$C1R_1T_{1\gamma}'$	.33	2	6[u]	1.5	60	10	.99	3.0×10^{-4}	2.4×10^{-2}	2.1×10^{-4}	3.9×10^{-2}	1.7×10^{-1}	3.9×10^{-3}	2.2×10^{-2}	3.8×10^{-3}
			BASE CASE	$C1R_1T_{1\gamma}'$	.34	3	1	7	60	6	.98	2.97×10^{-4}	1.8×10^{-2}	1.5×10^{-5}	5.1×10^{-2}	2.1×10^{-1}	5.1×10^{-3}	2.5×10^{-2}	5.1×10^{-3}
			LOWER	$C1R_1'T_1c$	.33	8	1	7	75	.3[v]	.15	9.6×10^{-6}	2.0×10^{-6}	1.4×10^{-7}	5.4×10^{-6}	2.4×10^{-5}	5.1×10^{-7}	3.0×10^{-6}	5.1×10^{-7}
2	1.1×10^{-5}	7.0×10^{-2}	UPPER	$C2R_1VSFT_{1\gamma}'$	.1	25	20	2	60	50	.98	6.4×10^{-4}	1.15×10^{-2}	3.1×10^{-2}	8.2×10^{-2}	1.5×10^{-1}	3.0×10^{-3}	1.5×10^{-2}	2.3×10^{-3}
			BASE CASE	$C2R_1T_{1\gamma}'$	.8	30	20	2	60	50	.98	6.4×10^{-4}	1.14×10^{-2}	3.1×10^{-2}	8.1×10^{-2}	1.5×10^{-1}	2.9×10^{-3}	1.5×10^{-2}	2.2×10^{-3}
			LOWER	$C2R_1'T_{1\gamma}'$	.1	30	30	2	60	3[v]	.96	6.4×10^{-4}	3.1×10^{-3}	1.4×10^{-2}	3.1×10^{-2}	3.5×10^{-2}	6.4×10^{-4}	2.5×10^{-3}	3.2×10^{-4}
3	3.6×10^{-7}	8.4×10^{-1}	UPPER	$C3R_1T_{1\gamma}'$	.33	2	6[u]	1.5	60	10	.99	6.3×10^{-3}	8.0×10^{-3}	4.7×10^{-3}	4.4×10^{-2}	1.7×10^{-1}	5.0×10^{-3}	2.0×10^{-2}	3.9×10^{-3}
			BASE CASE	$C3R_1T_{1\gamma}'$	.34	8	1	7	60	6	.99	6.3×10^{-3}	1.1×10^{-2}	3.5×10^{-5}	4.6×10^{-2}	1.7×10^{-1}	4.2×10^{-3}	2.1×10^{-2}	3.1×10^{-3}
			LOWER	$C3R_1T_1c\overline{\epsilon}$	.33	8	1	7	75	.3[v]	.49	3.1×10^{-3}	1.5×10^{-3}	2.9×10^{-5}	1.0×10^{-2}	4.3×10^{-2}	1.1×10^{-3}	5.2×10^{-3}	1.0×10^{-3}
4	6.1×10^{-6}	4.3×10^{-1}	UPPER	$C4R_1VSFT_{1\gamma}'$	.1	1.5	10[u]	1	60	60	.99	6.4×10^{-4}	2.3×10^{-2}	3.5×10^{-2}	1.0×10^{-1}	2.1×10^{-1}	4.1×10^{-3}	2.2×10^{-2}	3.4×10^{-3}
			BASE CASE	$C4R_1T_{1\gamma}'$	.8	1.5	10[u]	1	60	60	.99	6.4×10^{-4}	2.3×10^{-2}	3.5×10^{-2}	1.0×10^{-1}	2.1×10^{-1}	4.1×10^{-3}	2.2×10^{-3}	3.4×10^{-3}
			LOWER	$C4R_1T_1c\overline{\epsilon}$	.1	1.5	10[u]	1	60	3[v]	.98	6.4×10^{-4}	7.4×10^{-3}	1.3×10^{-2}	2.9×10^{-2}	4.2×10^{-2}	5.9×10^{-4}	4.0×10^{-3}	4.1×10^{-4}
5	2.0×10^{-8}	≈1.0	UPPER	$C5R_1\overline{z}$[r]	.1	1	1	0	60	6	.99	6.3×10^{-3}	6.6×10^{-2}	6.5×10^{-1}	6.9×10^{-1}	5.9×10^{-1}	1.8×10^{-1}	3.9×10^{-2}	6.1×10^{-3}
			BASE CASE	$C5R_1\overline{z}$[s]	.45	1	1	0	50	6	.9	5.1×10^{-3}	4.6×10^{-1}	6.6×10^{-2}	6.9×10^{-1}	5.9×10^{-1}	1.8×10^{-1}	3.9×10^{-2}	6.1×10^{-3}
			LOWER	$C5R_1\overline{z}$[t]	.45	1	2.5	0	75	.3[v]	.99	2.5×10^{-3}	3.5×10^{-1}	4.0×10^{-2}	4.3×10^{-1}	3.0×10^{-1}	1.0×10^{-1}	1.2×10^{-2}	2.0×10^{-3}

FOOTNOTES

(a) Obtained from functional event tree quantification.

(b) Given that the core has become vulnerable, this is the conditional probability that a substantial release occurs.

(c) Releases are treated as distributed with simple discrete model (see uncertainties). Values are shown for the best estimate and two bounding cases.

(d) See Appendix H for discussion of nomenclature.

(e) Treated as a discrete distribution with three values [illegible] probability of either the point estimate or the bounds.

(f) in hours

(g) in hours

(h) in hours

(i) in meters

(j) in 10^6 BTU/hr

(k) Background on the isotope groups and release mechanisms is presented in WASH-1400, Appendix VII.

(l) Includes I and Br.

(m) Includes Cs and Rb

(n) Includes Te and Sb

(o) Includes Ba and Sr

(p) Includes Ru, Rh, Co, Mo, Tc

(q) Includes Y, La, Zr, Nb, Ce, Pr, Nd, Np, Pu, Am, Cm

(r) Case where significant deflagration has occurred in secondary containment causing a reducing atmosphere to exist; difference is in chemical form of iodine.

(s) Case where no significant deflagration in secondary containment has occurred.

(t) Extrapolated from data for release from containment terminated after .3 to .6 hr.

(u) Release characteristics are such that an initial puff occurs during the first hour followed by another puff towards the end of the release duration.

(v) Low energy represents slower blowdown after a smaller containment breach.

ATTACHMENT 3

ATTACHMENT 4

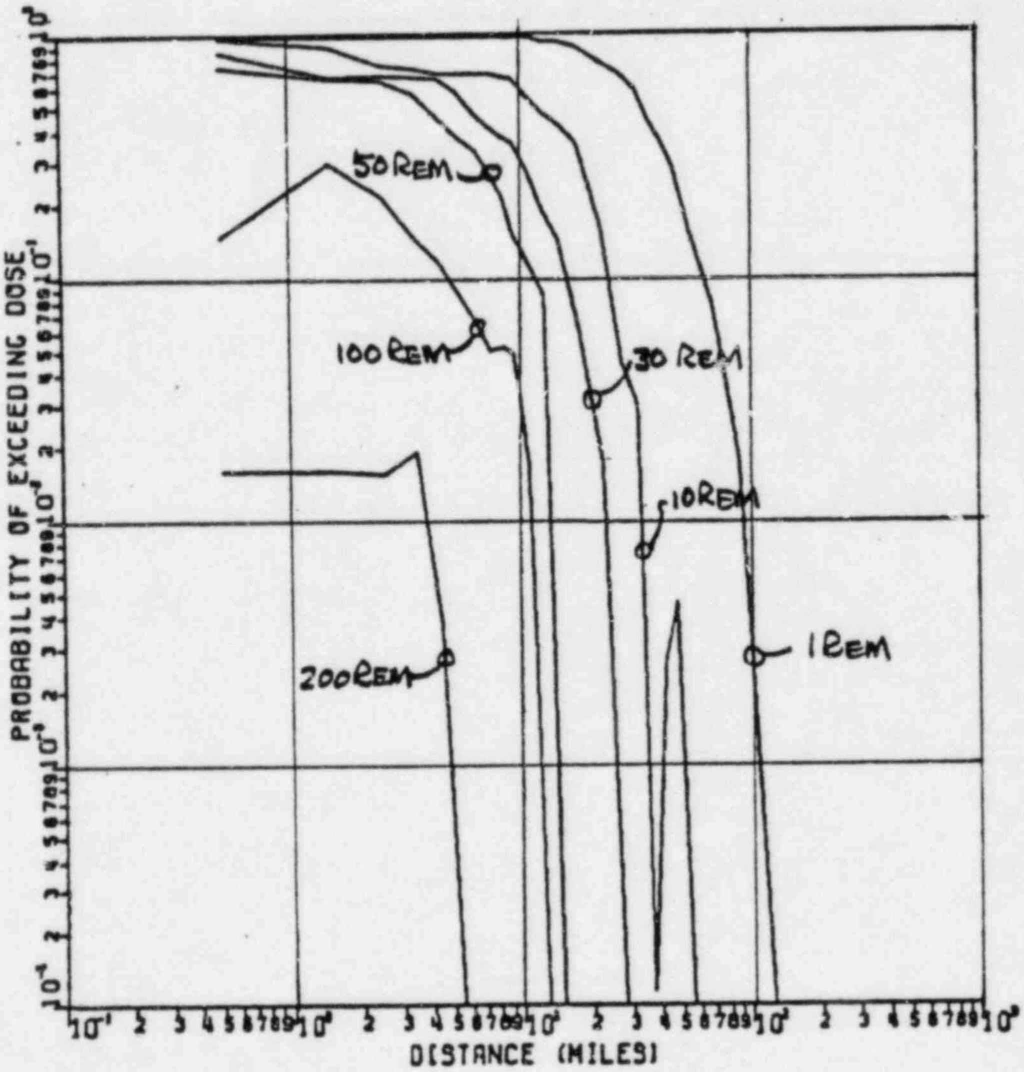

Figure 1

Projected Probabilities of Exceeding Specified Doses as a Function of Distance From a Severe Nuclear Power Plant Accident (SAI Category 1)

ATTACHMENT 5

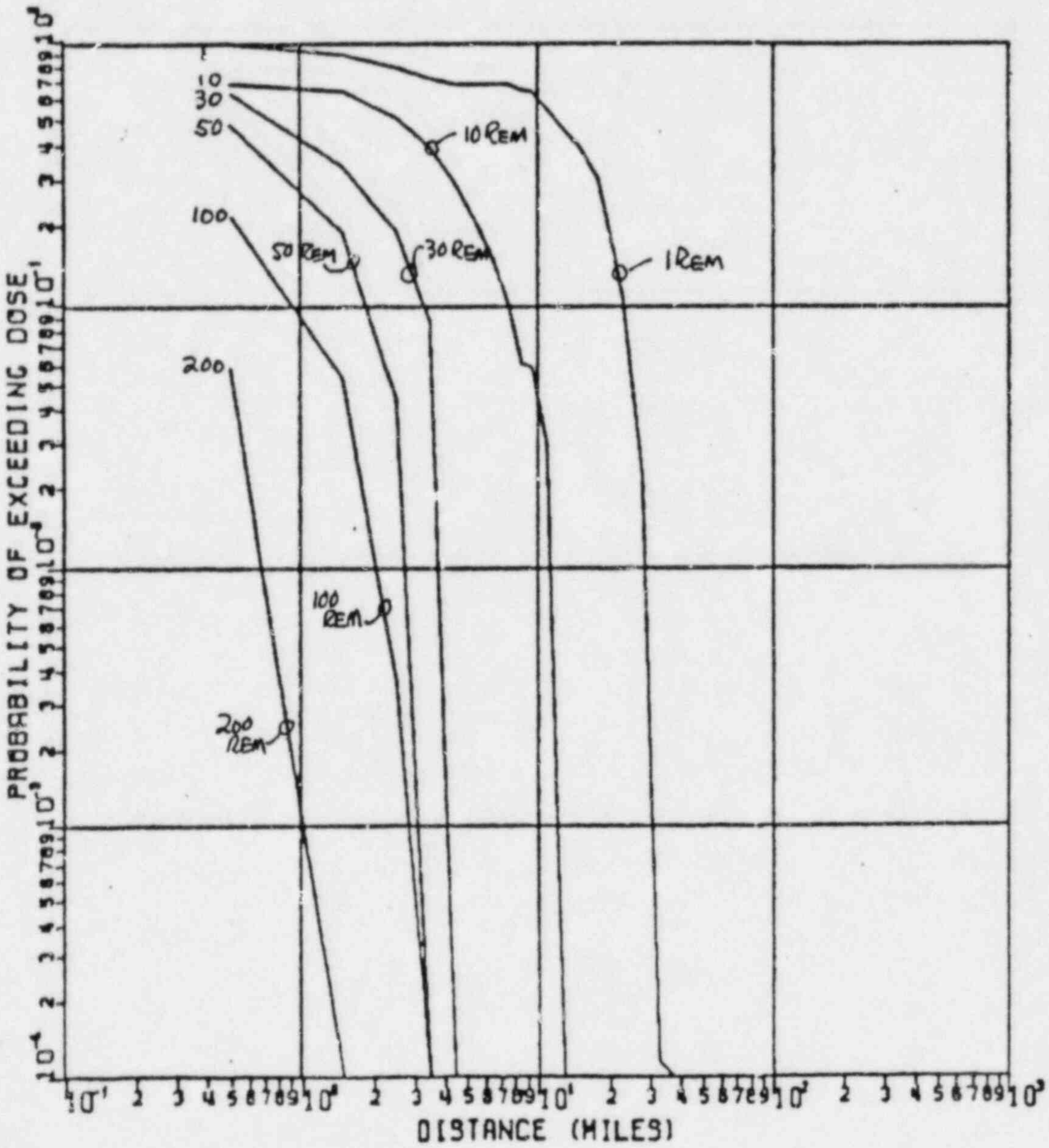

Figure 2

Noble Gas Whole Body Doses (DBA) CBA-1

ATTACHMENT 6

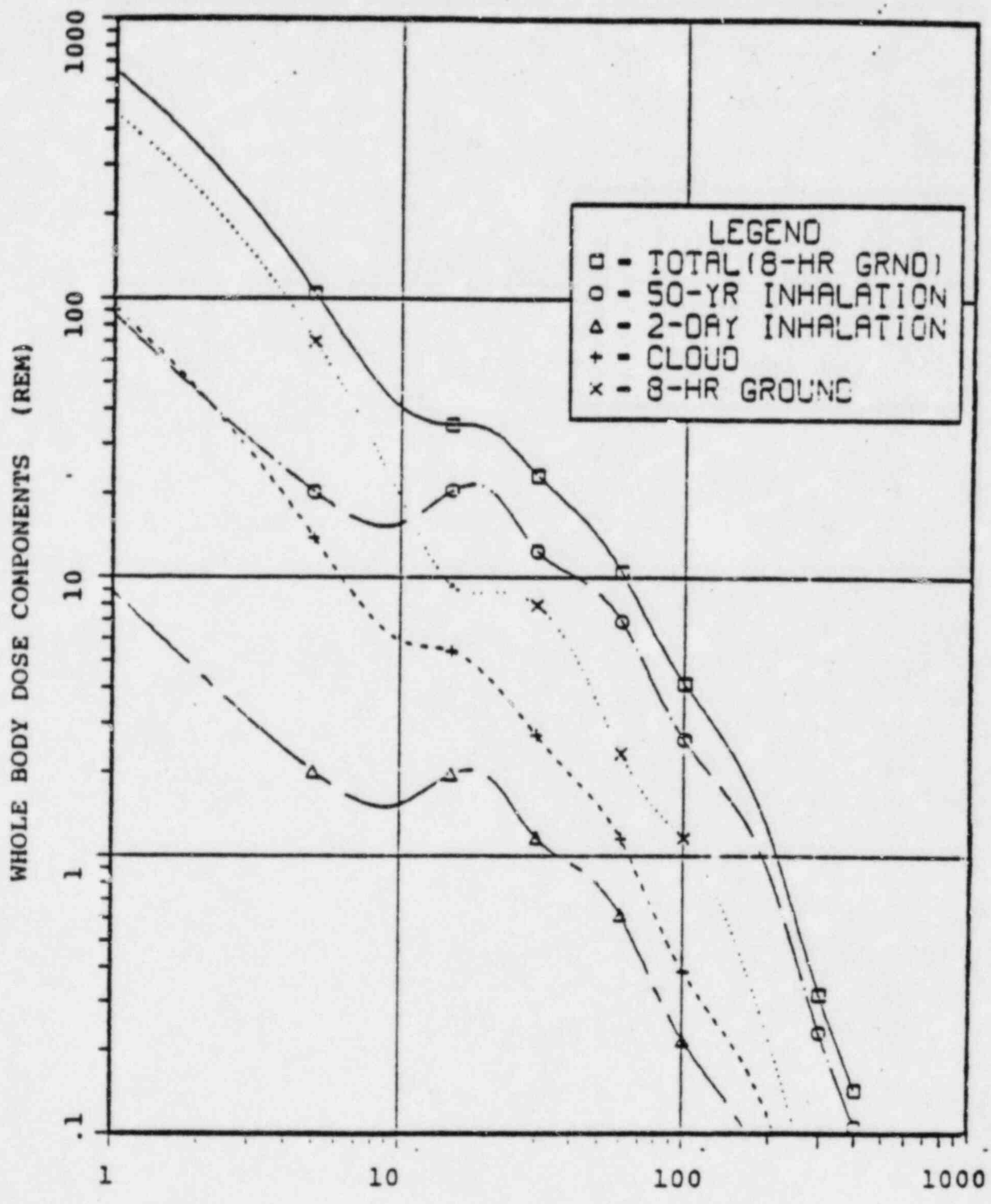

Figure 3

Mean Whole Body Dose for Severe Core Melt Accidents

ATTACHMENT 7

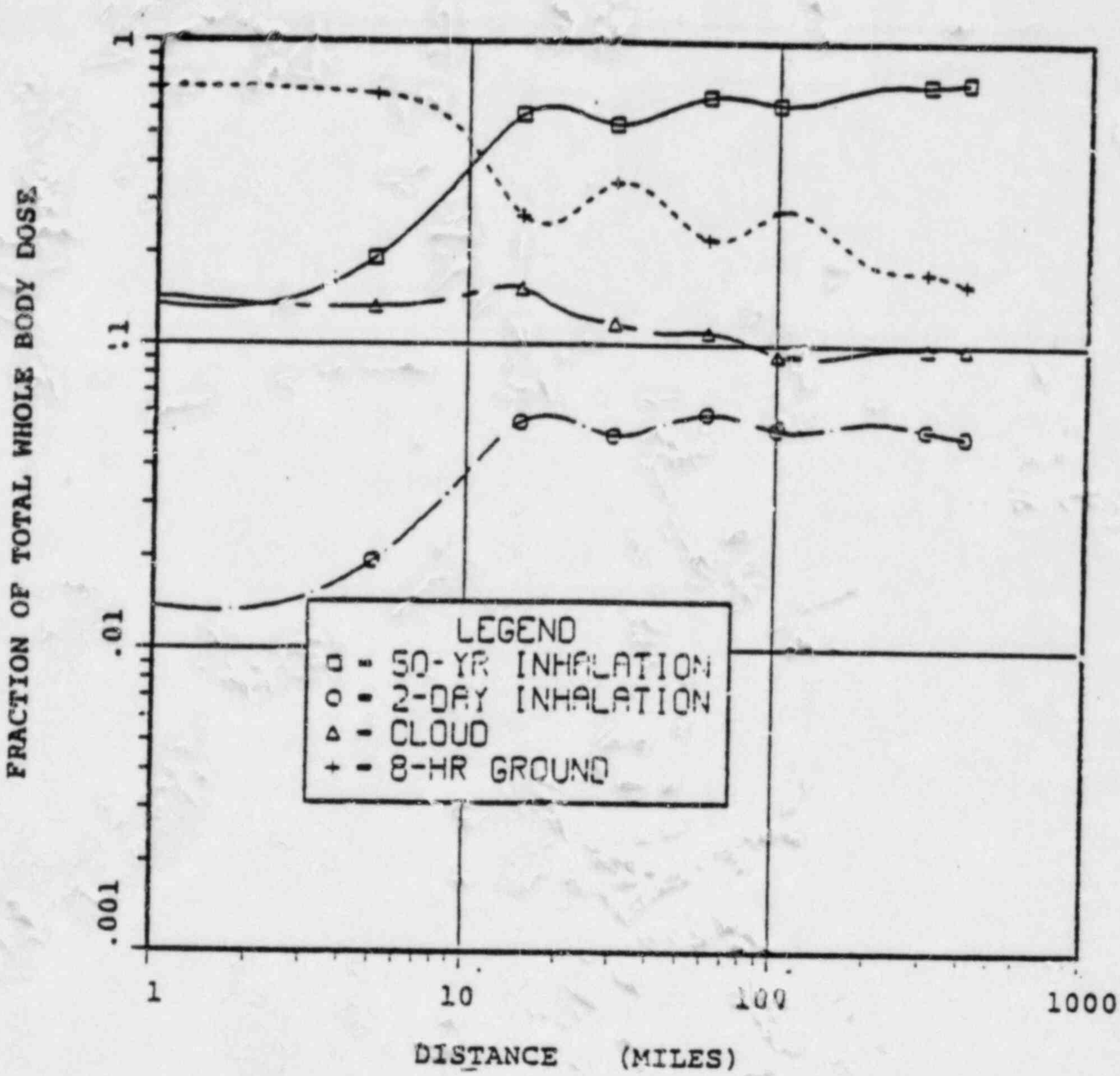

Figure 4

Whole Body Dose Components for Severe Core Melt Accidents

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

DOCKETED
USNRC

'82 OCT 22 P4:31

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

In the Matter of LONG ISLAND LIGHTING COMPANY (Shoreham Nuclear Power Station, Unit 1)	Docket No. 50-322 (O.L.) (Emergency Planning Proceedings)

CERTIFICATE OF SERVICE

I hereby certify that copies of the following items were sent on October 12, 1982 by first class mail, except as otherwise noted, to the persons listed below:

1. Direct Testimony of Andrew C. Kanen, Dr. James H. Johnson, Dr. Stephen Cole and Dr. Kai T. Erikson On Behalf of Suffolk County Regarding Contentions EP 2B and EP 5B (Traffic Congestion Issues).

2. Direct Testimony of Dr. Kai T. Erikson and Dr. Stephen Cole On Behalf Of Suffolk County Regarding Contention 5A (Role Conflict).

3. Direct Testimony of Gregory C. Minor On Behalf Of Suffolk County Regarding Contentions EP 10B and 10C (Radiation Monitoring) and EP 14 (Dose Assessment).

4. Direct Testimony of Dr. Fred C. Finlayson On Behalf of Suffolk County Regarding Contention EP 14 (Accident Assessment and Dose Assessment Models).

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