UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	Docket Nos. 50-247-SP
CONSOLIDATED EDISON COMPANY	)	50-286-SP
OF NEW YORK (Indian Point, Unit 2)	)	
	)	
POWER AUTHORITY OF THE STATE	)	
OF NEW YORK (Indian Point, Unit 3)	)	

DIRECT TESTIMONY OF
SANFORD ISRAEL, JACK HICKMAN, GREGORY KOLB,
ROBERT G.EASTERLING, AND ALAN D. SWAIN
COMMISSION QUESTION 1 AND BOARD QUESTION 1.1

Q.1 Please state your name and business address for the record.

A.1 My name is Sanford Israel. My business address is U.S. Nuclear Regulatory Commission, Washington D.C. 20555

Q.2 Please identify your position with NRC and describe your responsibilities in that position.

A.2 I am a Risk Analyst in the Reliability and Risk Assessment Branch of the Division of Safety Technology within the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. My responsibilities are to provide risk perspectives based on a review of core-melt sequences and system reliabilities for various assigned tasks.

Q.3 Have you prepared a statement of your professional qualifications.

A.3 Yes, I have prepared the statement of my professional qualifications attached to this testimony.

Q.4 Please state your name and business address for the record.

A.4 My name is Jack Hickman. My business address is Sandia National Laboratories, Division 9412, Albuquerque, New Mexico.

Q.5 Please identify your position with Sandia and describe your responsibilities in that position.

A.5 I am Supervisor of the Nuclear Fuel Cycle Systems Safety Division at Sandia National Laboratories. In that position, I am responsible for the performance, evaluation and application of nuclear power plant system reliability analysis for programs being performed for the Nuclear Regulatory Commission.

Q.6 Please describe your education and professional qualifications.

A.6 A copy of my professional qualifications is attached to this testimony.

Q.7 Please state your name and business address for the record.

A.7 My name is Gregory Kolb. My business address is Sandia National Laboratories, Division 9412, Albuquerque, New Mexico.

Q.8 Please identify your position with Sandia and describe your responsibilities in that position.

A.8 I am a member of the technical staff within the Nuclear Fuel Cycle Systems Safety Divison. I am responsible for the performance and review of nuclear power plant systems reliability analyses which are a part of several research and technical assistance programs funded by the Nuclear Regulatory Commission.

Q.9 Please describe your education and professional qualifications.

A.9 A copy of my professional qualifications is attached to this testimony.

Q.10 Please state your name and business address for the record.

A.10 My name is Robert G. Easterling. My business address is Sandia National Laboratories, Albuquerque, New Mexico.

Q.11 Please identify your position with Sandia and describe your responsibilities in that position.

A.11 I am a staff member in the Reliability Department. My activities include research in statistical data analysis and in the application of statistical techniques to reliability and risk assessment.

Q.12 Have you prepared a statement of your professional qualifications?

A.12 Yes, I have prepared the statement of my professional qualification attached to this testimony.

Q.13 Please state your name and business address for the record.

A.13 My name is Alan D. Swain. My business address is Sandia National Laboratories, Albuquerque, New Mexico.

Q.14 Please identify your position with Sandia and describe your responsibilities in that position.

A.14 I am a member of the technical staff in the Statistics, Computing, and Human Factors Division where I provide technical leadership for the human factors group. My activities include the identification of the potential for human error in complex systems, the provision of design recommendations to reduce this potential, and the grantification of the human error potential for reliability and risk assessment studies.

Q.15 Have you prepared a statement of your professional qualifications?

A.15 Yes, I have prepared a statement of my professional qualifications attached to this testimony.

Q.16 Mr. Israel, what is the purpose of this testimony?

A.16 The purpose of this testimony is to provide frequencies of plant damage states, caused by internally initiated events excluding fire and sabotage, that are used in James Meyer's testimony on radiological releases. This testimony also provides an assessment of the IPPSS results in the area of plant damage states caused by internally initiated transients and accidents.

Q.17 Mr. Israel, what is the scope of this testimony?

A.17 This testimony discusses the probabilistic treatment of internally initiated reactor transients and accidents (except those initiated by fire and sabotage) leading to core melt. It assumes that the IPPSS is an accurate reflection of the current, as-built-and-operated plants (except for the contemplated modifications identified in the licensees' response to the NRC Staff First Set of Interrogatories Concerning Questions 1 and 2, dated June 25, 1982).

This testimony does not cover the adequacy of the probabilistic treatment of fires in IPPSS which is discussed in the testimony of Benjamin Buchbinder et al, or adequacy of the probabilistic treatment cf external events which is discussed in the testimony of Robert J. Budnitz. This testimony does not cover sabotage. Further the issue of pressurized thermal shock is discussed in the testimony on Board Question 1.4 and the issues of steam generator tube rupture is discussed in the testimony on Board Question 2.2.1.

Q.18 Please discuss what is meant by plant damage state.

A.18 A plant damage state is a group of accident sequences that result in core melt and have a common containment condition such as containment bypass prior to core-melt or core-melt with or without containment cooling available.

Q.19 Why does the Staff emphasize core melt accidents in it's risk assessment?

A.19 The core of an operating nuclear power plant contains radioactive materials which, if ineffectively contained, can cause harm to the population and environment in the vicinity of a plant. Even after an operating reactor is shut down, a mechanism for releasing radioactivity exists. This is called the "afterheat" or "decay heat" produced in the fuel after the nuclear chain reaction ceases. This decay heat diminishes gradually once a nuclear reactor is shut down, but within the first hours or days after shutdown, the decay heat released within the fuel has the potential to melt the fuel and breach each of the several barriers used to obstruct the release of radioactive materials. Such a phenomenon may take place if the decay heat in the fuel is not dissipated in controlled ways.

Because a "core melt" accident has the potential to release large quantities of radioactivity, it is the principal cause for concern among potential nuclear reactor accidents.

We do not differentiate between severe core damage and core-melt events and we will refer to both types as core-melt events. This assumption is conservative because the potential radioactive release for events that stop at severe core damage may be less than that for core melt events. The analyses have not been refined to differentiate the fraction of events that may terminate at severe core damage.

The timing of the core melt with respect to containment failure has an impact on the consequences because it affects the radioactive releases to the environment. If the containment building remains intact following core melt, the potential radioactive releases would be reduced and the consequences to the public would be small compared to sequences where the containment building fails above ground level or is bypassed prior to core melt or shortly thereafter.

Q.20 How were the plant damage states derived?

A.20 The Licensees performed a risk assessment of their plants (IPPSS) which included the frequencies of transients and accidents resulting in core melt. The Staff contracted with Sandia National Laboratories to review the core melt analysis portion of IPPSS and to derive modified estimates for core melt sequences as appropriate. Sandia grouped the core melt sequences into plant damage state categories based on the availability of containment cooling following core melt.

Q.21 Messrs. Kolb/Hickman, please describe the scope of the Sandia review of internally initiated events presented in the IPPSS and your findings.

A.21 The Sandia review of the IPPSS is presented in NUREG/CR-2934, Review and Evaluation of The Indian Point Probabilistic Safety Study. The IPPSS estimated the frequency of several hundred core melt accident sequences initiated by internal events. Because of the very large number of sequences considered in the report, and the time limitation placed on our review, it became necessary for us to focus on a subset. The subset which received the most extensive review were those identified in the IPPSS to dominate the internal event core melt frequency or the important plant damage state frequencies. There were 18 of these sequences, 10 for Indian Point 2, and 8 for Indian Point 3. The review of these sequences relied heavily on our past PRA experience. This experience aided us in searching for subtle methodological problem areas, potential omissions, and important analysis assumptions which could have a significant impact on the sequence frequency estimates. Our review discussed in NUREG/CR-2934 also entailed an evaluation of the basic building blocks of the IPPSS internal event analysis; namely, initiating events, fault trees, event trees, human errors, data, common cause and sequence analysis. These building blocks were reviewed to determine if possible errors, unrealistic assumptions, or omissions made by the IPPSS analysts could allow additional sequences to the above mentioned 18 to become important.

Q.22 Based on your review of the IPPSS, what is your overall impression of the internal event analysis presented in that study?

A.22 It is our opinion that the IPPSS internal event analysis is a state-of-the-art analysis performed by competent analysts and from it much is to be learned. Their treatment (modeling, assumptions, completeness, etc.) is comparable with other state-of-the-art probabilistic risk assessments. We did identify areas in our review where we thought alternative modeling or calculational techniques should be considered; and based on these, calculated revised estimates of the dominant accident sequence frequencies.

Q.23 With reference to the internal event building blocks mentioned previously, describe the major findings of your review.

A.23 Initiating events are plant occurrences which require a rapid reactor shutdown and subsequent safety system operation to prevent core melt. We found the initating events covered in the IPPSS to be relatively complete and their frequency estimates to be reasonable compared to those addressed in previous PRA's. However, an exception to this was found. We found no indication that the IPPSS considered an initiating event caused by a pipe break in the component cooling water system. This event was found in our review to be an important contribution to the core-melt frequency.

Fault trees are logic models which describe the various ways safety systems can fail. We reviewed all the fault trees presented in the IPPSS and found them in general to be a reasonable representation of the Indian Point safety systems. However, we felt some changes in the logic structure of the fault trees for the service water system, auxiliary feedwater system, and fan coolers were appropriate. The IPPSS analysts agreed with this conclusion and we factored these changes into our plant damage state frequencies. System unavailabilities presented in our evaluation NUREG/CR-2934 compare reasonably with estimates from other studies. We also noted that the IPPSS analysis was unduly conservative by not taking any credit for the main feedwater system to remove decay heat following a reactor scram.

Event trees are logic models which delineate the various combinations of safety system failures following an initating event leading to core melt and/or containment failure. These combinations are known as accident sequences. We reviewed all of the event trees and found the structure of most to be appropriate. We made changes in several for purposes of calculating our revised estimates. They were: 1) steam generator tube rupture, 2) loss of service water, 3) loss of component cooling water, and 4) anticipated transients without scram (ATWS). Differences in 1) and 4) above had the most impact on the results. The IPPSS steam generator tube rupture event tree did not include a containment bypass sequence caused by a stuck open secondary safety valve. The IPPSS ATWS event tree gave credit for an ATWS fix for which the utilities decided to defer implementation. We performed a revised analysis which considered sequences involving a steam generator tube rupture and stuck open safety valve, as well as, ATWS sequences which did not include the fix. We quantified these sequences and included them in our final plant damage state frequencies.

The human is the most difficult nuclear plant "system" to analyze. He can have both a positive and negative influence on the course of various accidents. Because of the very large number of human activities possible, we focused our human error review on those activities identified in the IPPSS to have a major impact on the dominant accident sequences. This investigation revealed that either no or limited procedures existed for several of the activities, e.g., feed and bleed core cooling, loss of component cooling water. In our revised estimates, we assigned bounding human error probabilities for these situations. Four activities which were important and identified as having emergency procedures were reviewed in some depth. These activities dealt with switchover from injection to recirculation following a LOCA. Of the four, our revised estimates resulted in higher human error probability estimates for two of them and lower estimates for two of them.

We commend the IPPSS for greater use of plant specific initiating event and component failure data than found in many past PRA's. Our evaluation indicated that the Bayesian methodology produced reasonable point estimate failure probabilities based on our comparison of the IPPSS estimates of the dominant accident sequences to estimates based primarily on the IPPSS--reported data. Our evaluation also considered statistical confidence limits on the occurrence frequencies of these sequences. These limits, based primarily on IPPSS--reported data, identify a range of sequence frequencies that are consistent with the data considered. We found that the IPPSS point estimates generally fell within these ranges.

Common cause events result in failure of multiple safety systems or subsystems which compromise designed redundancy. A common cause failure could be the result of a test or maintenance error, a common support system, or an unidentified cause. The IPPSS modeled the more important of these common cause failures. The IPPSS analysts however, did not have available at the time they performed the work some common cause data sources which have recently been made available to us (e.g., Common Cause

Fault Rates for Pumps, EGG-EA-5289 and Precursors to Potential Severe Core Damage Accidents, NUREG/CR-2497). These sources suggest higher common cause system failure in some cases. Our revised results take into account these more recent common cause data sources.

Sequence analysis requires the logical combining and quantifying of the initiating event and the fault trees for each accident sequence defined by the event trees. At this point in our review, we requantified the dominant accident sequences based on the results of the review of the basic building blocks of the IPPSS. Each accident sequence was then assigned to a plant damage state. Comparing our results with those found in the IPPSS reveals some differences.

Q.24 What are the internal event plant damage state frequency estimates derived by the Sandia review, and how do they compare with the IPPSS estimates?

A.24 Table 1 summarizes the revised frequency estimates and compares them to the IPPSS. These frequency estimates were extracted from Tables 5.2-1 and 5.2-2 in our final report, NUREG/CR-2934, "Review and Evaluation of the Indian Point Probabilistic Safety Study," December 1982.

Q.25 What are the uncertainties in the results?

A.25 Uncertainties can be divided into three types: data, modeling, and completeness. Each of these types is addressed below with respect to internally initiated events.

Table 1

IPPSS and Revised Internal Event Plant Damage State Comparison

	IPPSS Estimates		Sandia Estimates	
Plant Damage State	IPS	IP3	IP2	IP3
Containment Bypass Prior to Core Melt	4.6(-7)	4.6(-7)	2.1(-7)	2.1(-7)
Core Melt Without Containment Cooling	1.1(-6)	7.1(-7)	6.7(-7)	5.7(-7)
Early Core Melt With Containment Cooling	5.4(-5)	1.8(-5)	1.2(-4)	1.8(-4)
Late Core Melt With Containment Cooling	3.4(-5)	1.1(-4)	1.0(-4)	1.0(-4)
Steam Generator Tube Rupture With Stuck Open Secondary Safety Valve	--	--	5.2(-7)	2.0(-7)

Data

Data uncertainties arise from a lack of infinite data pertaining to initiating event frequencies and subsystem and component failure probabilities. These types of uncertainties were evaluated for our revised list of Indian Point dominant accident sequences by using the Maximus methodology, referenced in our report, to calculate statistical confidence limits for the frequency of these sequences, and for the plant damage state frequencies. The results are presented in Chapter 5 of NUREG/CR-2934.

Modeling

Modeling uncertainties stem from the inadequacy of the PRA logic models to perfectly represent reality. Some of the more important modeling uncertainties are evaluated in NUREG/CR-2934 via a sensitivity study. The sensitivity issues addressed there are: 1) feed and bleed core cooling, 2) core melt/systems interaction, and 3) reactor coolant pump seal LOCA.

Completeness

Uncertainties associated with completeness are related to the inability of the analyst to evaluate perfectly and exhaustively all contributions to core melt because of oversights due to lack of knowledge or the limited scope of the analysis. We identified some areas where the IPPSS internal event analysis appeared incomplete. (Two of the more important areas were discussed previously; namely, the component cooling water system pipe break and steam generator tube rupture with a stuck open secondary safety valve.) Although we believe our revised estimates reflect a state-of-the-art level of completeness for PRA's, there is no guarantee that our review, which was based largely on our own PRA experience, was complete in an absolute sense.

Q.26 Mr. Israel, what core-melt frequencies for internal events are used in the NRC Staff's response to Commission Question 1?

A.26 The estimated core-melt frequencies (for internal events) used in the Staff's response to Commission Question 1 are presented in Table 2, as a function of plant damage state.

Under Containment Bypass Prior to Core Melt, the estimated frequency of interfacing LOCA (4×10^{-7}) is a compromise between the original IPPSS values (5×10^{-7}, Table 8.3-9 Event 24 and Table 8.3-10 Event 15), the initial Sandia reestimate (3 to 5×10^{-7}, draft Sandia report) and the final Sandia estimate (2×10^{-7}). The estimates are sensitive to the different models used to describe an interfacing LOCA, an event which has not occurred. The NRC Staff has not performed a separate review to differentiate between the various results because of the small difference among the estimates. So, we are using an estimate somewhat biased to the high side, for purposes of developing a risk estimate.

The estimated core-melt frequency for a steam generator tube rupture event with a stuck-open secondary relief valve was developed by the NRC Staff as discussed in Testimony for Board Question 2.2.1. The NRC Staff's evaluation considered multiple tube ruptures, while Sandia's evaluation considered only a single tube rupture. The multiple tube rupture events yielded higher estimated core-melt frequencies.

Table 2 Estimated Core Melt Frequencies For Internal Events

Plant Damage State	IP2	IP3
Containment Bypass Prior to Core Melt		
Interfacing LOCA	4×10^{-7}*	4×10^{-7}
SGTR	2×10^{-6}	2×10^{-6}
Early Core Melt without Containment Cooling	6×10^{-7}	6×10^{-7}
Early Core Melt with Containment Cooling	1.2×10^{-4}	1.8×10^{-4}
Late Core Melt with Containment Cooling	1×10^{-4}	1×10^{-4}

*Frequencies are events per reactor-year.

The estimated core-melt frequencies for Early Core Melt without Containment Cooling, Early Core Melt with Containment Cooling, and Late Core Melt with Containment Cooling were obtained from Sandia's reevaluation of the dominant sequences presented in IPPSS (Table 5.2-1 and Table 5.2-2 of the Sandia report, NUREG/CR-2934). Core melt sequences with containment cooling available generally involve LOCA's. The Sandia estimates for these categories considered additional aspects not considered in IPPSS in some of the sequencies. The Staff believes that the Sandia analyses for these categories are a better representation of the plants (within the scope of the study); however, the core-melt frequencies may be somewhat high because of the LOCA's frequencies used. The LOCA's frequencies used are from the IPPSS and represent point estimates that are higher than those used in previous PRAs. Because the plant damage states (core-melt with containment cooling) are not dominant contributors to offsite consequences, as shown in J. Meyer's testimony, the NRC Staff did not try to resolve what LOCA's frequencies are most appropriate for risk assessments.

I cannot confirm that the estimates of core-melt frequencies presented in Table 2 are correct in an absolute sense because of uncertainties associated with completeness, data, and modeling. The overall core-melt frequencies for internally initiated events appear to be consistent with other studies within the scope of the analysis performed and appear to be reasonably developed within the state-of-the-art based on the Sandia review of IPPSS.

Q.27 Does this conclude your testimony?

A.27 Yes

SANFORD L. ISRAEL

Professional Qualifications

I am a Risk Analyst in the Reliability and Risk Assessment Branch, Division of Safety Technology, Office of Nuclear Reactor Regulation. I am responsible for evaluating the reliability of nuclear power plants, identifying dominant risk sequences associated with plant operation, and assessing the relative importance of safety issues and proposed plant modifications.

I received a Bachelor of Science Degree in 1958 and a Master of Science Degree in 1959 from MIT. Both these degrees were in Mechanical Engineering.

From 1960 to 1966, I was an engineer with Nuclear Development Associates (later known as United Nuclear Corporation) where I was initially involved in test programs related to two-phase flow and hydrogen thermal conductivity. Subsequently, I was responsible for the thermal-hydraulic design of fuel for light water reactors.

From 1966 to 1974, I was Manager of the Thermal-Hydraulic Section at United Nuclear Corporation (later known as Gulf United Nuclear Fuels Corporation) where I supervised test programs, computer code development, and analysis related to the thermal-hydraulic design of light water reactor fuel.

In June 1974, I accepted employment with the Atomic Energy Commission (now the Nuclear Regulatory Commission) in the Reactor Systems Branch where I was responsible for reviewing various safety systems and analyses in the Sequoyah, North Anna, Floating Nuclear, and Alan S. Barton plants. In 1976, I was appointed Section Leader in the Reactor Systems Branch where I supervised the activities of several professionals who reviewed systems important to safety for conformance to the regulations, Standard Review Plan, and guidelines.

In 1979, I served on the Bulletins and Orders Task Force which developed and implemented recommendations based on concerns that were identified in the TMI-2 accident.

In May 1980, I joined the Reliability and Risk Assessment Branch where I have developed risk perspectives for several safety issues such as light~~ing~~ning strikes and the necessity of PORV's and have developed a position paper on the National Reliability Evaluation Program.

Jack W. Hickman

Educational and Professional Qualifications

Jack W. Hickman is Supervisor of the Nuclear Fuel Cycle Systems Safety Division of Sandia National Laboratories in Albuquerque, New Mexico. In this position, he is responsible for the performance of a variety of reliability and probabilistic risk assessment (PRA) programs under sponsorship of the U.S. Nuclear Regulatory Commission (NRC). Past and current programs include responsibility of performance of the risk assessments of light water reactor power plants in the Reactor Safety Study Methodology Applications Program (RSSMAP) and the Interim Reliability Evaluation Program (IREP). In addition, Mr. Hickman is or has been responsible for programs involving the use of risk assessment to address generic and plant specific issues before the NRC. Generic issues have included underground siting, auxiliary feedwater availability, DC power system reliability, and station blackout frequency. He is also currently responsible for a risk based evaluation of the issues identified in the Systematic Evaluation Program and the NRC probabilistic risk assessment training program and occasionally lectures on fault tree analysis and PRA for George Washington University. He serves as Chairman of the Technical Writing Group preparing the Industry/NRC PRA Procedures Guide and has responsbility for the IREP PRA Procedures Guide. He occasionally consults for the Advisory Committee on Reactor Safeguards (ACRS) on the subject of probabilistic risk assessments.

Mr. Hickman received his MS Degree from the University of New Mexico and BS Degree from Oklahoma State University, both in electrical engineering.

LIST OF PUBLICATIONS

1) Dominant Accident Sequences for an Ice Condenser PWR Plant, S.V. Asselin, J.W. Hickman, et al., ANS Winter Meeting, November 1978.

2) System Event Tree Analyses for Determining Accident Sequences that Dominate Risk in LWR Power Plants, S.V. Asselin, J.W. Hickman, et al., ANS Topical Meeting on Probabilistic Analysis of Nuclear Reactor Safety, May 1978.

3) The Reactor Safety Study Methodology Applications Program: Sequoyah #1 PWR Power Plant, NUREG/CR-1659, SAND-80-1897, February 1981.

4) Development and Organization of the Industry/NRC PRA Procedures Guide, International ANS/ENS Topical Conference on Probabilistic Risk Assessment, Hickman.

5) PRA Procedures Guide, Review Draft, NUREG/CR-2300, April 1982.

6) An Assessment of Auxiliary Feedwater Systems, M. Taylor, D. Carlson, M. Cunningham, S. Asselin, J. Hickman, G. Kolb, ANS Transactions, Vol. 33, 1979.

Gregory J. Kolb

Educational and Professional Qualifications

Gregory J. Kolb is a member of the Nuclear Fuel Cycle Systems Safety Division of Sandia National Laboratories in Albuquerque, New Mexico. In this position, he is responsible for the performance and review of nuclear power plant systems reliability analyses which are a part of several research and technical assistance programs funded by the Nuclear Regulatory Commission (NRC). Mr. Kolb has acted as systems analysis team leader for several nuclear power plant probabilistic risk assessments (PRA). Most recently he was the principal investigator for the Arkansas Nuclear One risk assessment as part of the Interim Reliability Evaluation Program. Prior to this assignment, he acted as team leader for the Oconee and Calvert Cliffs PRA as part of the Reactor Safety Study Methodology Applications Program. In addition, he was part of the Crystal River PRA analysis team and was one of the principal reviewers of the Zion PRA. Besides PRA activities, Mr. Kolb has been involved in the technical review of the "Rogovin Study" analysis of the accident at Three Mile Island, and a program which investigated the reliability of several nuclear power plant auxiliary feedwater systems. He has published several papers and reports in the field of PRA.

Mr. Kolb received a BS degree in Engineering from California State University Northridge in 1975 and a MS degree in Nuclear Engineering from the University of Arizona in 1977.

LIST OF PUBLICATIONS

1) The Reactor Safety Study Methodology Applications Program - Oconee Results. S.W. Hatch, G. Kolb - ANS Transactions, Vol. 38, 1981.

2) LWR Core Meltdown Accident Sequencer Phenomenology, P. Cybulskis, R. Wooton, G. Kolb, ANS Transactions, Vol. 41, 1982.

3) Reactor Safety Study Methodology Applicators Program: Calvert Cliffs #2 PWR Power Plant, S. Hatch, G. Kolb, R. Wooton, P. Cybolskis. NUREG/CR-1659, May 1982.

4) Reactor Safety Study Methodology Applications Program: Oconee #3 PWR Power Plant, G. Kolb, S. Hatch, P. Cybulskis, R. Wooton, revised May 1981, NUREG/CR-1659.

5) Interim Reliability Evaluation Program Analysis of the Arkansas Nuclear One-Unit 1 Nuclear Power Plant, G. Kolb, NUREG/CR-2787, June 1982

6) Insights from the Arkansas Nuclear One Unit 1 IREP Analysis, G. Kolb, Proceedings of the International ANS/ENS Topical Meetings on PRA, September 20-24, 1981, Port Chester, NY.

7) Systemic Event Tree Methodology Employed in the Interim Reliability Evaluation Program, G. Kolb, proceedings of the International ANS/ENS Topical Meeting on PRA, September 20-24, 1981, Port Chester, NY.

8) An Assessment of Auxiliary Feedwater Systems, M. Taylor, D. Carlson, G. Kolb, M. Cunningham, J. Hickman, S. Asselin, ANS Transactions, Vol. 33, 1979.

9) Arkansas Nuclear 1, Unit 1, Risk Analysis Results, by G. Kolb and D. Kunsman, International Meeting on Thermal Nuclear Reactor Safety, August 29 - September 2, 1982, Chicago.

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PROFESSIONAL QUALIFICATIONS OF

ROBERT G. EASTERLING

Robert G. Easterling has been a staff member in the Reliability Department of Sandia Laboratories, Albuquerque, since August 1967, except for January to June 1974 when he was a visiting lecturer in the Department of Statistics, University of Wisconsin, Madison, and from June 1975 to June 1977 when he held the position of Statistical Adviser in the Applied Statistics Group of the Nuclear Regulatory Commission. He received his B.S. in mathematics and his M.S. and Ph.D. in statistics from Oklahoma State University in 1964, 1965, and 1967, respectively.

He is a Fellow of the American Statistical Association and has served in various organizational positions including president of the Albuquerque chapter. He is editor of the applied statistics journal, TECHNOMETRICS, and has written articles which appear in various statistical, reliability, and quality control journals and conference proceedings.

His activities at the NRC and Sandia have included consulting and research in statistical data analysis and in the application of statistical techniques to reliability and risk assessment. Publication and presentations in the area of nuclear risk assessment include:

"Probabilistic Analysis of Common Mode Failures." Proceedings of ANS Topical Meeting on Probabilistic Analysis of Nuclear Reactor Safety, May 1978.

"Statistical Problems in Nuclear Regulation," with R. H. Moore, Annual Meeting of the American Statistical Association, August 1978.

"Some Statistical Aspects of Uncertainty Analysis," 1978 ANS Annual Meeting.

Review of Anatomy of Risk, by W. D. Rowe, TECHNOMETRICS, May, 1980, p. 278, 279.

"Statistical Problems in the Assessment of Nuclear Risks," Annual Meeting of the American Statistical Association, August 1980.

"Comments on the Bayesian Method for Estimating Reactor Core Melt Frequency." Nuclear Science and Engineering, 1980, p. 202.

"Discussion of Conover/Iman Paper (Small Sample Sensitivity Analysis Techniques for Computer Models, with an Application to Risk Assessment)." Communications in Statistics, 1980.

"Some Observations on: Reliability Problems in Power Generator Systems." To appear in Proceedings of the 1981 DOE Statistical Symposium.

ALAN D. SWAIN, PH.D.

Dr. Swain is a member of the technical staff in the Statistics, Computing, and Human Factors Division at Sandia National Laboratories, Albuquerque, New Mexico, where he provides the technical direction for the Human Factors Group. In addition to his Sandia responsibilities, he is a regular lecturer at the University of Wisconsin - Extension, and he lectures annually in Europe.

He has been active in the nuclear weapons field since 1954 and in the nuclear power field since 1968. He has advised government authorities in England, Scotland, France, Germany, Denmark, Norway, Sweden, Finland, Italy, and South Africa on methods to reduce serious human errors in the operation of nuclear power plants (NPPs) and on methods to quantitatively assess the influence of human errors in these plants. In 1979 he met with the Swedish Commission on Evaluating Nuclear Power to advise them of the kinds and relative costs of human factors improvements in Swedish NPPs that could materially reduce the risk of human-induced failures in these plants. Before and after the Three-Mile Island accident he has assisted the Nuclear Regulatory Commission (NRC) by evaluating the impact of potential human errors in responding to possible transient conditions in NPPs.

He was responsible for the human reliability analysis in WASH-1400. The human reliability analysis model employed was THERP*, developed by Dr. Swain in the early 1960's for applications to weapon systems, and is now widely employed for a variety of man-machine systems. In Section 6.1 of Appendix III** the rationale for the high estimates of human failure probabilities in WASH-1400 is stated in terms of the poor human factors practices and design features in NPPs.

In 1975 he followed with a study*** of the Zion NPP in which detailed human factors problems were described (which are characteristic of all presently operating plants) plus suggestions for inexpensive changes in on-site practice, equipment, and operating procedures which would result in substantial improvement in human reliability. Subsequent experience in the Zion and other NPPs indicates that these suggestions have not been acted upon.

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* Swain, A. D., <u>A Method for Performing a Human Factors Reliability Analysis</u>, Monograph SCR-685, Sandia National Laboratories, Albuquerque, NM, Aug. 1963, 62 pp.

** "Human Reliability Analysis", Section 6.1 in <u>Appendix III - Failure Data</u>, of WASH-1400 (NUREG-75/014): <u>Reactor Safety Study - An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants</u>, U. S. Nuclear Regulatory Commission, Wash. D.C., Oct. 1975, pp III-59 to III-69 (written by A. D. Swain and H. E. Guttmann).

***Swain, A. D., <u>Preliminary Human Factors Analysis of Zion Nuclear Power Plant</u>, NUREG76-6503, U. S. Nuclear Regulatory Commission, Wash. D.C., Oct. 1975, 81 pp.

Dr. Swain's major effort in NPP research is the Handbook of Human Reliability Analysis With Emphasis on NPP Applications*, a nearly four-year research effort sponsored by the Probabilistic Analysis Staff, Office of Nuclear Regulatory Research, US NRC. The Handbook consists of models of human performance, estimates of human error probabilities (an uncertainty bounds) for NPP tasks, and a human reliability method and technique to apply the data and models to estimate the influence of human errors on safety and reliability of NPP operations. The models are unique in the field of human behavior in that they can be used to predict a wide variety of human behavior in an applied setting, they are testable, and they are modifiable as better data on human performance in NPPs become available. The Handbook is serving as the method for assessing the influence of human errors in the NRC's Interim Reliability Evaluation Program (IREP), a program to quantitatively assess the risk to the public of a sample of operating US NPPs.

Dr. Swain received his PH.D. in experimental psychology in 1953 from the Ohio State University. In 1950 he participated in the Psychology Corporation's study of the effectiveness of flight simulation, the first quantitative assessment of this training technique. From 1952 to 1958 he was with the American Institutes for Research. His research included applications to maintainability design and techniques and to training and training devices. From 1958 to 1961 he was with Dunlap and Associates, Inc., where he designed training programs, course curricula, and training aids and devices for the U. S. Navy, including the nuclear submarine program. In later work for this company, he designed the human reliability program for the Air Force's Manned Orbiting Laboratory. In 1961 he joined the Reliability Analysis Department at Sandia National Laboratories where most of his work has been in human engineering and human reliability analysis in nuclear weapons and nuclear energy. During this time he was Visiting Professor at the University of New Mexico for three years. Since 1972 he has spent up to one-fourth time lecturing for the Department of Energy, the Nuclear Regulatory Commission, the University of Wisconsin, and various foreign agencies. He is the author of numerous publications, including several chapters in books and his own books.

Dr. Swain is a Fellow of the Human Factors Society, a Senior Member of the American Society for Quality Control, an ASQC certified reliability engineer, and a certified psychologist in the State of New Mexico. He is a member of the Group of Experts on Human Error Data and Assessment of the Committee on the Safety of Nuclear Installations, Organization for Economic Co-Operation and Development, with headquarters in Paris. He meets annually with European experts in human reliability to assess their use of his methods and to advise them on human factors problems in nuclear power plants.

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* Swain, A. D. and Guttmann, H. E., Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications (Draft for Public Review), NUREG/CR-1278, Office of Nuclear Regulatory Research, U. S. Nuclear Regulatory Commission, Wash. D. C., Oct. 1980, approx. 600 pp.

January 1982
A. D. Swain

A LIST OF HUMAN FACTORS PUBLICATIONS OF SANDIA NATIONAL LABORATORIES

Items marked with an asterisk, an "at" sign, a plus sign, or a number sign can be obtained as follows:

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Items with SCTM, SCR, SCDC, SCDR, SLA, or SAND designations are Sandia National Laboratories' reports.

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(*) 3. Rook, L. W., Reduction of Human Error in Industrial Production, SCTM-93-62(14), June 1962, 29 pages.

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